
Safety Evaluation Report
related to the operation of
Watts Bar Nuclear Plant,
Units 1 and 2
Docket Nos. 50-390 and 50-391

Tennessee Valley Authority

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

April 1994



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NUREG-0847
Supplement No. 13

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ABSTRACT

This report supplements the Safety Evaluation Report (SER), NUREG-0847 (June 1982), Supplement No. 1 (September 1982), Supplement No. 2 (January 1984), Supplement No. 3 (January 1985), Supplement No. 4 (March 1985), Supplement No. 5 (November 1990), Supplement No. 6 (April 1991), Supplement No. 7 (September 1991), Supplement No. 8 (January 1992), Supplement No. 9 (June 1992), Supplement No. 10 (October 1992), Supplement No. 11 (April 1993), and Supplement No. 12 (October 1993), issued by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission with respect to the application filed by the Tennessee Valley Authority, as applicant and owner, for licenses to operate the Watts Bar Nuclear Plant, Units 1 and 2 (Docket Nos. 50-390 and 50-391). The facility is located in Rhea County, Tennessee, near the Watts Bar Dam on the Tennessee River. This supplement provides recent information regarding resolution of some of the outstanding and confirmatory items, and proposed license conditions identified in the SER.

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ABBREVIATIONS

AECL	Atomic Energy of Canada Limited
AFD	axial flux difference
AFW	auxiliary feedwater
AMSAC	ATWS mitigation system actuation circuitry
ANSI	American National Standards Institute
ATWS	anticipated transient without scram
BISI	bypassed and inoperable status indication
BNL	Breakhaven National Laboratory
BOP	balance of plant
BTP	branch technical position
CAOC	constant axial offset control
CAP	corrective action program
CATD	Corrective Action Tracking Document
CECC	Central Emergency Command Center
CFR	Code of Federal Regulations
CMAA	Crane Manufacturers Association of America
CMS	Code Management System
CNPP	Corporate Nuclear Performance Plan
COC	Chattanooga Office Complex
COLR	Core Operating Limits Report
CRDM	control rod drive mechanism
CSB	Containment Systems Branch
CSST	common station service transformer
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
EAL	emergency action level
EBS	Emergency Broadcast System
ECCS	emergency core cooling system
ECSP	Employee Concern Special Program (TVA)
ECTG	Employee Concern Task Group (TVA)
EDG	emergency diesel generator
EMI	electromagnetic interference
EMT	emergency medical technician
EOF	emergency operations facility
EPA	Environmental Protection Agency
EPIP	emergency plan implementing procedure
EPZ	emergency planning zone
ERCW	essential raw cooling water
ERF	emergency response facility
ESD	electrostatic discharge
ESF	engineered safety feature
ESFAS	engineered safety features actuation system

FEMA	Federal Emergency Management Agency
FSAR	Final Safety Analysis Report
FWCS	feedwater control system
GDC	general design criterion
H&I	harassment and intimidation
IE	Office of Inspection and Enforcement
IEEE	Institute of Electrical and Electronics Engineers
IFBA	integral fuel burnable absorber
INPO	Institute of Nuclear Power Operations
IPE	individual plant examination
IR	inspection report
JIC	Joint Information Center
LCO	limiting conditions for operation
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LRC	local recovery center
MERT	Medical Emergency Response Team
MSS	median signal selector
NEMA	National Electrical Manufacturers Association
NIS	nuclear instrumentation system
NRC	Nuclear Regulatory Commission
NRN	Office of Nuclear Reactor Regulation
NSRS	Nuclear Safety Review Staff (TVA)
NSSS	nuclear steam supply system
NUREG	report prepared by NRC staff
NUREG/CR	report prepared by NRC contractor
OBE	operating basis earthquake
ODS	operation duty specialist
OIA	Office of Inspector and Auditor
OSC	Operational Support Center
PACAO	Program for Assurance of Completion and Assurance of Quality
PER	problem evaluation report
PLC	programmable logic controller
PORV	pilot-operated relief valve
PROM	programmable read-only memory
QA	quality assurance
QC	quality control
QPTR	quadrant power tilt ratio
QTC	Quality Technology Corporation

RAI	request for additional information
RAOC	relaxed axial offset control
RCCA	rod cluster control assembly
RCP	reactor coolant pump
RCS	reactor coolant system
REP	radiological emergency plan
RERP	radiological emergency response plan
RFI	radiofrequency interference
RG	regulatory guide
RMCC	Radiological Monitoring Control Center
RPS	reactor protection system
RTD	resistance temperature detector
RTP	rated thermal power
RWFS	RCCA bank withdrawal from subcritical
SED	site emergency director
SEOC	State Emergency Operations Center
SER	Safety Evaluation Report
SG	steam generator
SI	safety injection
SOS	shift operations supervisor
SP	special program
SRP	Standard Review Plan
SSER	supplemental Safety Evaluation Report
SSI	soil-structure interaction
TAC	technical assignment control
TEDE	total effective dose equivalent
TI	temporary instruction
TSS	Technical Specifications
TSC	Technical Support Center
TTD	trip time delay
TVA	Tennessee Valley Authority
V&V	verification and validation
WABA	wet annular burnable absorber
WARL	Western Area Radiological Laboratory
WBNPP	Watts Bar Nuclear Performance Plan
WCAP	report prepared by Westinghouse

INTRODUCTION AND DISCUSSION

1.1 Introduction

In June 1982, the Nuclear Regulatory Commission staff (NRC staff or staff) issued a Safety Evaluation Report, NUREG-0847, regarding the application by the Tennessee Valley Authority (TVA or the applicant) for licenses to operate the Watts Bar Nuclear Plant, Units 1 and 2. The Safety Evaluation Report (SER) was followed by SER Supplement No. 1 (SSER 1, September 1982), Supplement No. 2 (SSER 2, January 1984), Supplement No. 3 (SSER 3, January 1985), Supplement No. 4 (SSER 4, March 1985), Supplement No. 5 (SSER 5, November 1990), Supplement No. 6 (SSER 6, April 1991), Supplement No. 7 (SSER 7, September 1991), Supplement No. 8 (SSER 8, January 1992), Supplement No. 9 (SSER 9, June 1992), Supplement No. 10 (SSER 10, October 1992), Supplement No. 11 (SSER 11, April 1993) and Supplement No. 12 (October 1993). As of this date, the staff has completed review of the applicant's Final Safety Analysis Report (FSAR) up to Amendment 78.

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

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

Each of the following sections or appendices of this supplement is numbered the same as the section or appendix of the SER that is being updated, and the discussions are supplementary to, and not in lieu of, the discussion in the SER, unless otherwise noted. Accordingly, Appendix A is a continuation of the chronology of the safety review.¹ Appendix E is a list of principal contributors to this supplement. Appendices B-D and F-Z are not changed by this SSER. In Appendix AA, the staff reprints its supplemental safety evaluation concerning the TVA corrective action program on the Q-List. In Appendix BB, the technical evaluation report on feedwater check valve slam analysis is reproduced.

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

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1.7 Summary of Outstanding Issues

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

<u>Issue²</u>	<u>Status</u>	<u>Section</u>
(1) Potential for liquefaction beneath ERCW pipelines and Class 1E electrical conduit	Resolved (SSER 3)	2.5.4.4
(2) Buckling loads on Class 2 and 3 supports	Resolved (SSER 4)	3.9.3.4
(3) Inservice pump and valve test program (TAC M74801)	Updated (SSER 5)	3.9.6
(4) Qualification of equipment (a) Seismic (TAC M71919) (b) Environmental (TAC M63591)	Resolved (SSER 9) Under review (SER)	3.10 3.11
(5) Preservice inspection program (TAC M63627)	Resolved for Unit 1 (SSER 10 and 12)	5.2.4, 6.6, App. Z
(6) Pressure-temperature limits for Unit 2	On hold	5.3.2, 5.3.3
(7) Model D-3 steam generator preheater tube degradation	Resolved (SSER 4)	5.4.2.2
(8) Branch Technical Position CSB 6-4	Resolved (SSER 3)	6.2.4
(9) H ₂ analysis review	Resolved (SSER 4)	6.2.5
(10) Safety valve sizing analysis (WCAP-7769)	Resolved (SSER 2)	5.2.2

²The TAC (technical assignment control) number that appears in parentheses after the issue title is an internal NRC control number by which the issue is managed through the Workload Information and Scheduling Program (WISP) and by which relevant documents are filed. Documents associated with each TAC number can be located by the NRC document control system, NUDOCs/AD.

<u>Issue</u>	<u>Status</u>	<u>Section</u>
(11) Compliance of proposed design change to the offsite power system to GDC 17 and 18 (TAC M63649)	Resolved (SSER 13)	8.2
(12) Fire-protection program (TAC M63648)	Under review (SER)	9.5.1
(13) Quality classification of diesel generator auxiliary system piping and components (TAC M63638)	Resolved (SSER 5)	9.5.4.1
(14) Diesel generator auxiliary system design deficiencies (TAC M63638)	Resolved (SSER 5)	9.5.4, 9.5.5, 9.5.7
(15) Physical Security Plan (TAC M63657)	Under review (SER)	13.6
(16) Boron-dilution event	Resolved (SSER 4)	15.2.4.4
(17) QA Program (TAC M76972)	Resolved (SSER 13)	17
(18) Seismic classification of cable trays and conduit (TACs R00508, R00516)	Resolved (SSER 8)	3.2.1, 3.10
(19) Seismic design concerns (TAC M79717, M80346):		
(a) Number of OBE events	Resolved (SSER 8)	3.7.3
(b) 1.2 multi-mode factor	Resolved (SSER 9)	3.7.3
(c) Code usage	Resolved (SSER 8)	3.7.3
(d) Conduit damping values	Resolved (SSER 8)	3.7.3
(e) Worst case, critical case, bounding calculations	Resolved (SSER 12)	3.7.3
(f) Mass eccentricities	Resolved (SSER 8)	3.7.2.1.2
(g) Comparison of set A versus set B response	Resolved (SSER 11)	3.7.2.12
(h) Category I(L) piping qualification	Resolved (SSER 8)	3.9.3
(i) Pressure relief devices	Resolved (SSER 7)	3.9.3.3
(j) Structural issues	Resolved (SSER 9)	3.8
(k) Update FSAR per 12/18/90 letter	Resolved (SSER 8)	3.7
(20) Mechanical systems and components (TACs M79718, M80345)		
(a) Feedwater check valve slam	Resolved (SSER 13)	3.9.1
(b) New support stiffness and deflection limits	Resolved (SSER 8)	3.9.3.4
(21) Removal of RTD bypass system (TAC M63599)	Resolved (SSER 8)	4.4.3
(22) Removal of upper head injection system (TAC M77195)	Resolved (SSER 7)	6.3.1

<u>Issue</u>	<u>Status</u>	<u>Section</u>
(23) Containment isolation using closed systems (TAC M63597)	Resolved (SSER 12)	6.2.4
(24) Main steamline break outside containment (TAC M63632)	Under review (SSER 7)	3.11
(25) Health Physics Program (TAC M63647)	Resolved (SSER 10)	12
(26) Regulatory Guide 1.97, Instruments To Follow Course of Accident (TACs M77550, M77551)	Resolved (SSER 9)	7.5.2
(27) Containment sump screen design anomalies (TAC M77845)	Resolved (SSER 9)	6.3.3
(28) Emergency procedure (TAC M77861)	Resolved (SSER 9)	13.5.2.1

1.8 Summary of Confirmatory Issues

SER Section 1.8 identified 42 confirmatory issues for which additional information and documentation were required to confirm preliminary conclusions. Issue 43 was added in SSER 6. This section updates the status of those items for which the confirmatory information has subsequently been provided by the applicant and for which review has been completed by the staff. The completion status of each of the issues is tabulated below, with the relevant document in which the issue was last addressed shown in parentheses. Detailed, up-to-date status information for still-unresolved issues is conveyed in the staff's summaries of the monthly licensing status meetings.

<u>Issue</u>	<u>Status</u>	<u>Section</u>
(1) Design-basis groundwater level for the ERCW pipeline	Resolved (SSER 3)	2.4.8
(2) Material and geometric damping effect in SSI analysis	Resolved (SSER 3)	2.5.4.2
(3) Analysis of sheetpile walls	Resolved (SSER 3)	2.5.4.2
(4) Design differential settlement of piping and electrical components between rock-supported structures	Resolved (SSER 3)	2.5.4.3
(5) Upgrading ERCW system to seismic Category I (TAC M63617)	Resolved (SSER 5)	3.2.1, 3.2.2
(6) Seismic classification of structures, systems, and components important to safety (TAC M63618)	Resolved (SSER 5)	3.2.1
(7) Tornado-missile protection of diesel generator exhaust	Resolved (SSER 2)	3.5.2, 9.5.4.1, 9.5.8

<u>Issue</u>	<u>Status</u>	<u>Section</u>
(8) Steel containment building buckling research program	Resolved (SSER 3)	3.8.1
(9) Pipe support baseplate flexibility and its effects on anchor bolt loads (IE Bulletin 79-02) (TAC M63625)	Resolved (SSER 8)	3.9.3.4
(10) Thermal performance analysis	Resolved (SSER 2)	4.2.2
(11) Cladding collapse	Resolved (SSER 2)	4.2.2
(12) Fuel rod bowing evaluation	Resolved (SSER 2)	4.2.3
(13) Loose-parts monitoring system	Resolved (SSER 3)	4.4.5
(14) Installation of residual heat removal flow alarm	Resolved (SSER 5)	5.4.3
(15) Natural circulation tests (TACs M63603, M79317, M79318)	Resolved (SSER 10)	5.4.3
(16) Atmospheric dump valve testing	Resolved (SSER 2)	5.4.3
(17) Protection against damage to containment from external pressure	Resolved (SSER 3)	6.2.1.1
(18) Designation of containment isolation valves for main and auxiliary feedwater lines and feedwater bypass lines (TAC M63623)	Resolved (SSER 5)	6.2.4
(19) Compliance with GDC 51	Resolved (SSER 4)	6.2.7, App. H
(20) Insulation survey (sump debris)	Resolved (SSER 2)	6.3.3
(21) Safety system setpoint methodology	Resolved (SSER 4)	7.1.3.1
(22) Steam generator water level reference leg	Resolved (SSER 2)	7.2.5.9
(23) Containment sump level measurement	Resolved (SSER 2)	7.3.2
(24) IE Bulletin 80-06	Resolved (SSER 3)	7.3.5
(25) Overpressure protection during low-temperature operation	Resolved (SSER 4)	7.6.5
(26) Availability of offsite circuits	Resolved (SSER 2)	8.2.2.1
(27) Non-safety loads powered from the Class 1E ac distribution system	Resolved (SSER 2)	8.3.1.1

<u>Issue</u>	<u>Status</u>	<u>Section</u>
(28) Low and/or degraded grid voltage condition (TAC M63649)	Resolved (SSER 13)	8.3.1.2
(29) Diesel generator reliability qualification testing (TAC M63649)	Resolved (SSER 7)	8.3.1.6
(30) Diesel generator battery system	Resolved (SSER 2)	8.3.2.4
(31) Thermal overload protective bypass	Resolved (SSER 2)	8.3.3.1.2
(32) Update FSAR on sharing of dc and ac distribution systems (TAC M63649)	Resolved (SSER 13)	8.3.3.2.2
(33) Sharing of raceway systems between units	Resolved (SSER 2)	8.3.3.2
(34) Testing Class 1E power systems	Resolved (SSER 2)	8.3.3.5.2
(35) Evaluation of penetration's capability to withstand failure of overcurrent protection device (TAC M63649)	Resolved (SSER 7)	8.3.3.6
(36) Missile protection for diesel generator vent line (TAC M63639)	Resolved (SSER 5)	9.5.4.2
(37) Component cooling booster pump relocation	Resolved (SSER 5)	9.2.2
(38) Electrical penetrations documentation (TAC M63648)	Under review (SER)	9.5.1.3
(39) Compliance with NUREG/CR-0660 (TAC M63639)	Resolved (SSER 5)	9.5.4.1
(40) No-load, low-load, and testing operations for diesel generator (TAC M63639)	Resolved (SSER 5)	9.5.4.1
(41) Initial test program	Resolved (SSER 3)	14
(42) Submergence of electrical equipment as result of a LOCA (TAC M63649)	Resolved (SSER 13)	8.3.3.1.1
(43) Safety parameter display system (TAC M73723)	Updated (SSER 6)	18.2, App. P

1.9 Summary of Proposed License Conditions

In Section 1.9 of the SER and in SSERs that followed, the staff identified 43 proposed license conditions. Since these documents were issued, the applicant has submitted additional information on some of these items, thereby removing the necessity to impose a condition. The completion status of the proposed license conditions is tabulated below, with the relevant document in which the

issue was last addressed shown in parentheses. Detailed, up-to-date status of still-unresolved issues is conveyed in the staff's summaries of the monthly licensing status meetings.

<u>Proposed Condition</u>	<u>Status</u>	<u>Section</u>
(1) Relief and safety valve testing (II.D.1)	Resolved (SSER 3)	3.9.3.3, 5.2.2
(2) Inservice testing of pumps and valves (TAC M74801)	Resolved (SSER 12)	3.9.6
(3) Detectors for inadequate core cooling (II.F.2) (TACs M77132, M77133)	Resolved (SSER 10)	4.4.8
(4) Inservice Inspection Program (TAC M76881)	Resolved (SSER 12)	5.2.4, 6.6
(5) Installation of reactor coolant vents (II.B.1)	Resolved (SSER 5)	5.4.5
(6) Accident monitoring instrumentation (II.F.1)		
(a) Noble gas monitor (TAC M63645)	Resolved (SSER 5)	11.7.1
(b) Iodine particulate sampling (TAC M63645)	Resolved (SSER 6)	11.7.1
(c) High-range in-containment radiation monitor (TAC M63645)	Resolved (SSER 5)	12.7.2
(d) Containment pressure	Resolved (SSER 5)	6.2.1
(e) Containment water level	Resolved (SSER 5)	6.2.1
(f) Containment hydrogen	Resolved (SSER 5)	6.2.5
(7) Modification to chemical feedlines (TAC M63622)	Resolved (SSER 5)	6.2.4
(8) Containment isolation dependability (II.E.4.2) (TAC M63633)	Resolved (SSER 5)	6.2.4
(9) Hydrogen control measures (NUREG-0694, II.B.7) (TAC M77208)	Resolved (SSER 8)	6.2.5, App. C
(10) Status monitoring system/BISI (TAC M77136, M77137)	Resolved (SSER 7)	7.7.2
(11) Installation of acoustic monitoring system (II.D.3)	Resolved (SSER 5)	7.8.1
(12) Diesel generator reliability qualification testing at normal operating temperature	Resolved (SSER 2)	8.3.1.6
(13) DC monitoring and annunciation (TAC M63649)	Resolved (SSER 13)	8.3.2.2

<u>Proposed Condition</u>	<u>Status</u>	<u>Section</u>
(14) Possible sharing of dc control power to ac switchgear	Resolved (SSER 3)	8.3.3.2.4
(15) Testing of associated circuits	Resolved (SSER 3)	8.3.3.3
(16) Testing of non-Class 1E cables	Resolved (SSER 3)	8.3.3.3
(17) Low-temperature overpressure protection/power supplies for pressurizer relief valves and level indicators (II.G.1) (TAC M63649)	Resolved (SSER 7)	8.3.3.4
(18) Testing of reactor coolant pump breakers	Resolved (SSER 2)	8.3.3.6
(19) Postaccident sampling system (TAC M77543)	Updated (SSER 5)	9.3.2
(20) Fire protection program (TAC M63648)	Under review (SER)	9.5.1.8
(21) Performance testing for communications systems (TAC M63637)	Resolved (SSER 5)	9.5.2
(22) Diesel generator reliability (NUREG/CR-0660) (TAC M63640)	Resolved (SSER 5)	9.5.4.1
(23) Secondary water chemistry monitoring and control program	Resolved (SSER 5)	10.3.4
(24) Primary coolant outside containment (III.D.1.1) (TACs M63646, M77553)	Resolved (SSER 10)	11.7.2
(25) Independent safety engineering group (I.B.1.2) (TAC M63592)	Resolved (SSER 8)	13.4
(26) Use of experienced personnel during startup (TAC M63592)	Resolved (SSER 8)	13.1.3
(27) Emergency preparedness (III.A.1.1, III.A.1.2, III.A.2) (TAC M63656)	Resolved (SSER 13)	13.3
(28) Review of power ascension test procedures and emergency operating procedures by NSSS vendor (I.C.7) (TAC M77861)	Resolved (SSER 10)	13.5.2
(29) Modifications to emergency operating instructions (I.C.8) (TAC M77861)	Resolved (SSER 10)	13.5.2
(30) Report on outage of emergency core cooling system (II.K.3.17)	Resolved (SSER 3)	13.5.3

<u>Proposed Condition</u>	<u>Status</u>	<u>Section</u>
(31) Initial test program (TAC M79872)	Resolved (SSER 7)	14.2
(32) Effect of high-pressure injection for small-break LOCA with no auxiliary feedwater (II.K.2.13)	Resolved (SSER 4)	15.5.1
(33) Voiding in the reactor coolant system (II.K.2.17)	Resolved (SSER 4)	15.5.2
(34) PORV isolation system (II.K.3.1, II.K.3.2) (TAC M63631)	Resolved (SSER 5)	15.5.3
(35) Automatic trip of the reactor coolant pumps during a small-break LOCA (II.K.3.5)	Resolved (SSER 4)	15.5.4
(36) Revised small-break LOCA analysis (II.K.3.30, II.K.3.31) (TAC M77298)	Resolved (SSER 5)	15.5.5
(37) Detailed control room design review (I.D.1) (TAC M63655)	Updated (SSER 6)	18.1
(38) Physical Security Plan (TAC M63657, M83973)	Resolved (SSER 10)	13.6.4
(39) Control of heavy loads (NUREG-0612) (TAC M77560)	Resolved (SSER 13)	9.1.4
(40) Anticipated transients without scram (Generic Letter 83-28, Item 4.3) (TAC M64347)	Resolved (SSER 5)	15.3.6
(41) Steam generator tube rupture (TAC M77569)	Updated (SSER 12)	15.4.3
(42) Loose-parts monitoring system (TAC M77177)	Resolved (SSER 5)	4.4.5
(43) Safety parameter display system (TAC M73723)	Opened (SSER 5)	18.2

1.12 Approved Technical Issues for Incorporation in the License as Exemptions

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

- (1) Seal leakage test instead of full-pressure test (Section 6.2.6, SSER 4) (TAC M63615)
- (2) Criticality monitor (Section 9.1, SSER 5) (TAC M63615)

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

1.13 Implementation of Corrective Action Programs and Special Programs

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

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

In NUREG-1232, Vol. 4, the staff documented its general review of the corrective action programs (CAPs) and special programs (SPs) through which the applicant would effect corrective actions at Watts Bar. When the report was published, some of the CAPs and SPs were in their initial stages of implementation. The staff stated that it will report its review of the implementation of all CAPs and SPs and closeout of open issues in future supplements to the licensing SER, NUREG-0847; accordingly, the staff prepared Temporary Instructions (TIs) 2512/016-043 for the Inspection Manual and adhered to the TIs to perform inspections of the CAPs and SPs. This new section was introduced in SSER 5 and will be updated in subsequent SSERs. The current status of all CAPs and SPs follows. The status described here fully supersedes that described in previous SSERs.

1.13.1 Corrective Action Programs

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

Program review status: Complete: NUREG-1232, Vol. 4; Letter, P. S. Tam (NRC) to D. A. Nauman (TVA), April 25, 1991 (the safety evaluation was reproduced in SSER 7 as Appendix P); supplemental safety evaluation dated April 24, 1992 (Appendix T of SSER 9); letter, P. S. Tam (NRC) to M. O. Medford (TVA), February 14, 1994.

Implementation status: Full implementation expected by June 1994.

NRC inspections: Inspection Reports 50-390, 391/90-09 (June 22, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/90-22 (November 21, 1990); 50-390, 391/90-24 (December 17, 1990); 50-390, 391/90-27 (December 20, 1990); 50-390, 391/90-30 (February 25, 1991); 50-390, 391/91-07 (May 31, 1991); 50-390, 391/91-09 (July 15, 1991); 50-390, 391/91-12 (July 12,

92-01 (March 17, 1992); audit report of June 12, 1992 (Appendix Y of SSER 9); 50-390, 391/92-05 (April 17, 1992); 50-390, 391/92-13 (July 16, 1992); 50-390, 391/92-18 (August 14, 1992); 50-390, 391/92-22 (September 18, 1992); 50-390, 391/92-26 (October 16, 1992); 50-390, 391/92-30 (November 13, 1992); 50-390, 391/92-35 (December 15, 1992); 50-390, 391/92-40 (January 15, 1993); 50-390, 391/93-10 (March 19, 1993); 50-390, 391/93-11 (March 25, 1993); 50-390, 391/93-35 (June 10, 1993); 50-390, 391/93-40 (July 15, 1993); 50-390, 391/93-48 (August 13, 1993); 50-390, 391/93-56 (September 20, 1993); 50-390, 391/93-63 (October 18, 1993); 50-390, 391/93-70 (November 12, 1993); 50-390, 391/93-74 (December 20, 1993); 50-390, 391/93-85 (January 14, 1994); 50-390, 391/93-91 (February 17, 1994); 50-390, 391/94-11 (March 16, 1994); to come.

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

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

Implementation status: Full implementation expected by July 1994.

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

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

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

Implementation status: Full implementation expected by May 1994.

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

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

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

Implementation status: Full implementation expected by July 1994.

NRC inspections: Inspection Reports 50-390, 391/89-05 (May 25, 1989); 50-390, 391/89-07; (July 11, 1989); 50-390, 391/89-14 (December 18, 1989); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/91-31 (January 13, 1992); 50-390, 391/92-02 (March 17, 1992); audit report of May 14, 1992 (Appendix S of SSER 9); 50-390, 391/92-05 (April 17, 1992); 50-390, 391/92-09 (June 29, 1992); 50-390, 391/92-201 (September 21, 1992); 50-390, 391/92-26 (October 16, 1992); 50-390, 391/93-07 (February 19, 1993); 50-390, 391/93-35 (June 10, 1993); 50-390, 391/93-70 (November 12, 1993); 50-390, 391/93-74 (December 20, 1993); 50-390, 391/93-91 (February 17, 1994); 50-390, 391/94-11 (March 16, 1994); to come.

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

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

Implementation status: Full implementation expected by May 1994.

NRC inspections: Inspection Reports 50-390, 391/90-30 (February 25, 1991); 50-390, 391/92-22 (September 18, 1992); 50-390, 391/92-40 (January 15, 1993); 50-390, 391/93-35 (June 10, 1993); 50-390, 391/93-40 (July 15, 1993); 50-390, 391/93-63 (October 18, 1993); 50-390, 391/94-11 (March 16, 1994); to come.

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

Program review status: Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 11, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3.10.

Implementation status: Full implementation expected by July 1994.

NRC inspections: Inspection Reports 50-390, 391/90-05 (May 10, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/90-28 (January 11, 1991); 50-390, 391/91-03 (April 15, 1991); audit report of May 14, 1992 (Appendix S of SSER 9); 50-390, 391/92-201 (September 21, 1992); 50-390, 391/93-07 (February 19, 1993); 50-390, 391/93-79 (March 4, 1994); to come.

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

Program review status: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 7, 1989; NUREG-1232, Vol. 4; review in progress, results to be published in Section 9.5.1 of a future SSER.

Implementation status: Full implementation expected by June 1994.

NRC inspections: To come.

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

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

Implementation status: Full implementation expected by July 1994.

NRC inspections: Inspection Reports 50-390, 391/89-14 (December 18, 1989); 50-390, 391/90-14 (August 3, 1990); 50-390, 391/90-18 (September 20, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/90-28 (January 11, 1991); 50-390, 391/91-03 (April 15, 1991); audit report of May 14, 1992 (Appendix S of SSER 9); 50-390, 391/92-201 (September 21, 1992); 50-390, 391/92-26 (October 16, 1992); 50-390, 391/92-35 (December 15, 1992); 50-390, 391/93-07 (February 19, 1993); 50-390, 391/93-35 (June 10, 1993); 50-390, 391/93-45 (July 20, 1993); 50-390, 391/93-56 (September 20, 1993); 50-390, 391/93-70 (November 12, 1993); 50-390, 391/93-74 (December 20, 1993); 50-390, 391/94-11 (March 16, 1994); to come.

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

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

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

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

(10) Heating, Ventilation, and Air-Conditioning Duct and Duct Supports (TAC R00510; TI 2512/025)

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

Implementation status: Full implementation expected by July 1994.

NRC inspections: Inspection Reports 50-390, 391/89-14 (December 18, 1989); 50-390, 391/90-05 (May 10, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/91-01 (April 4, 1991); 50-390, 391/92-02 (March 17, 1992); audit report of May 14, 1992 (Appendix S of SSER 9); 50-390, 391/92-08 (May 15, 1992); 50-390, 391/92-13 (July 16, 1992); 50-390, 391/92-201 (September 21, 1992); 50-390, 391/93-07 (February 19, 1993); 50-390, 391/93-91 (February 17, 1994); 50-390, 391/94-08 (March 11, 1994); to come.

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

Program review status: Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 8, 1989; NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to O. D. Kingsley (TVA), October 26, 1990 (Appendix K of SSER 6).

Implementation status: Full implementation expected by July 1994.

NRC inspections: Inspection Reports 50-390, 391/90-14 (August 3, 1990); 50-390, 391/90-23 (November 19, 1990); 50-390, 391/91-02 (March 6, 1991); 50-390, 391/91-03 (April 15, 1991); 50-390, 391/91-26 (December 6, 1991); 50-390, 391/93-74 (December 20, 1993); 50-390, 391/94-11 (March 16, 1994); to come.

(12) Prestart Test Program (TAC M71924)

Program review status: Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), October 17, 1989; NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to D. A. Nauman (TVA), March 27, 1991.

Implementation status: Withdrawn by letter (J. H. Garrity (TVA) to NRC, February 13, 1992). Applicant will re-perform preoperational test program per Regulatory Guide 1.68, Revision 2.

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

Program review status: Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), December 8, 1989; NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to M. O. Medford (TVA) June 9, 1992 (Appendix X of SSER 9); letter, P. S. Tam (NRC) to M. O. Medford (TVA), January 12, 1993; letter, F. J. Hebdon (NRC) to M. O. Medford (TVA), August 12, 1993; letter, P. S. Tam (NRC) to O. D. Kingsley (TVA), April 25, 1994.

Implementation status: Full implementation expected by April 1994.

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

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

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

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

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

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

Program review status: Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), November 22, 1989; NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to O. D. Kingsley (TVA), February 11, 1991 (Appendix N of SSER 6); letter, P. S. Tam (NRC) to M. O. Medford (TVA), July 27, 1992, and April 5, 1994.

Implementation status: Full implementation expected by May 1994.

NRC inspections: Inspection Reports 50-390, 391/91-08 (May 30, 1991); 50-390, 391/91-29 (December 27, 1991); 50-390, 391/92-03 (March 16, 1992); 50-390, 391/92-11 (June 12, 1992); 50-390, 391/92-17 (July 22, 1992); 50-390, 391/92-21 (September 18, 1992); 50-390,

391/92-40 (January 15, 1993); 50-390, 391/93-22 (April 25, 1993); 50-390, 391/93-34 (July 9, 1993); 50-390, 391/93-38 (June 24, 1993); to come.

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

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

Implementation status: 100% (certified by letter, J. H. Garrity (TVA) to NRC, December 2, 1991); staff concurrence in SSER 9, Section 3.7.1.

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

(16)(a) Civil Calculation Program (TAC R00514)

Program review status: No program review. A number of civil calculation categories are required by the Design Baseline and Verification Program CAP and constitute parts of the applicant's corrective actions. This program is regarded as complementary to but not part of the Seismic Analysis CAP. Staff efforts consist mainly of audits performed at the site and in the office.

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

NRC audits: Complete: Memorandum (publicly available), T. M. Cheng (NRC) to P. S. Tam, January 23, 1992; letter, P. S. Tam (NRC) to D. A. Nauman (TVA), January 31, 1992; letters, P. S. Tam (NRC) to M. O. Medford (TVA), May 26 and December 18, 1992 and July 2, 1993; 50-390, 391/93-07 (February 19, 1993); letter, P. S. Tam (NRC) to M. O. Medford (TVA), November 26, 1993.

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

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

Implementation status: Full implementation expected by July 1994.

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

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

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

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

NRC inspections: Inspection Reports 50-390, 391/89-04 (August 9, 1989); 50-390, 391/90-04 (May 17, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/91-05 (May 28, 1991); 50-390, 391/91-18 (October 8, 1991); 50-390, 391/91-23 (November 21, 1991); 50-390, 391/91-32 (February 10, 1992); 50-390, 391/92-20 (August 12, 1992); 50-390, 391/92-28 (October 9, 1992); 50-390, 391/93-02 (February 2, 1993); 50-390, 391/93-19 (March 15, 1993); 50-390, 391/93-38 (June 24, 1993); 50-390, 391/93-84 (December 21, 1993); 50-390, 391/94-05 (February 19, 1994); 50-390, 391/94-16 (March 15, 1994); to come.

1.13.2 Special Programs

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

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

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

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

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

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

Implementation status: 100% (certified by letter, W. J. Museler to NRC, December 30, 1993); staff concurrence to come.

NRC inspections: Inspection Report 50-390, 391/93-56 (September 20, 1993); to come.

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

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

Implementation status: Full implementation expected by June 1994.

NRC inspections: To come.

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

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

Implementation status: Full implementation by June 1994.

NRC inspections: Inspection Reports 50-390, 391/93-63 (October 18, 1993; to come.

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

Program review status: Complete: NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to O. D. Kingsley (TVA), February 6, 1991; letter, P. S. Tam (NRC) to TVA Senior Vice President, March 30, 1992 (Appendix U of SSER 9).

Implementation status: 100% (certified by letter, W. Museler (TVA) to NRC, April 2, 1993); staff concurrence in Inspection Report 50-390, 391/93-31 (May 6, 1993).

NRC inspections: Complete: Inspection Reports 50-390, 391/86-24 (February 12, 1987); 50-390, 391/92-05 (April 17, 1992); 50-390, 391/92-09 (June 29, 1992); 50-390, 391/92-27 (September 25, 1992); 50-390, 391/93-31 (May 6, 1993).

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

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

Implementation status: Full implementation expected by June 1994.

NRC inspections: To come.

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

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

Implementation status: 100% (certified by letter, W. J. Museler (TVA) to NRC, August 31, 1993); staff concurrence in Inspection Report 50-390, 391/93-67 (November 1, 1993).

NRC inspections: Complete: Inspection Reports 50-390, 391/90-09 (June 22, 1990); 50-390, 391/90-13 (August 2, 1990); 50-390, 391/93-01 (February 25, 1993); 50-390, 391/93-09 (March 26, 1993); 50-390, 391/93-67 (November 1, 1993).

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

Program review status: Complete: NUREG-1232, Vol. 4; Section 3.6 of SSER 11.

Implementation status: Full implementation expected by May 1994.

NRC inspections: Inspection Reports 50-390, 391/93-85 (January 14, 1994); to come.

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

Program review status: Complete: NUREG-1232, Vol. 4; this program covers areas addressed in Chapter 12 of the SER and SSERs.

Implementation status: Full implementation expected by July 1994.

NRC inspections: To come.

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

Program review status: Complete: NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to TVA Senior Vice President, March 19, 1992; Section 2.5 of SSER 9.

Implementation status: 100% (certified by letter, W. J. Museler (TVA) to NRC, July 27, 1992); staff concurrence in SSER 11, Section 2.5.4.4.

NRC inspections: Complete: Inspection Reports 50-390, 391/89-21 (May 10, 1990); 50-390, 391/89-03 (May 11, 1989); audit report by L. B. Marsh (NRC) (October 10, 1990); audit report, P. S. Tam (NRC) to D. A. Nauman (TVA), January 31, 1992; audit report, P. S. Tam (NRC) to M. O. Medford (TVA), May 26 and December 18, 1992; 50-390, 391/92-45 (February 17, 1993).

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

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

Implementation status: 100% (certified by letter, W. J. Museler (TVA) to NRC, July 24, 1992); staff concurrence in Inspection Report 50-390, 391/93-10 (March 19, 1993).

NRC inspections: Complete: Inspection Reports 50-390, 391/90-19 (October 15, 1990); 50-390, 391/91-08 (May 30, 1991); 50-390, 391/93-10 (March 19, 1993).

1.16 Staff Actions on Quality Technology Company Matters

In May 1985, TVA awarded a contract to Quality Technology Company (QTC) to develop and implement a program for conducting confidential interviews with TVA employees working for the Watts Bar Nuclear Plant. The confidential interviews were conducted with emphasis on the identification of employee concerns dealing with nuclear safety at TVA facilities. After learning of a contract dispute between TVA and QTC, the NRC issued an order on January 30, 1986, which in part ordered TVA to retain QTC employee concern records and ensure NRC would have access to those records in order to photocopy them. To protect the identity of TVA employees, QTC kept the files from TVA, but agreed that NRC could retain a copy of the files on site with the condition that NRC would not reveal the identity of TVA employees who reported the concerns to QTC.

On April 18, 1986, the staff finished copying the files and sent a letter to TVA to rescind the part of Item V(A) of the January 30, 1986, order that prohibits removing the original files from TVA property, thus permitting the original unexpurgated records to be moved to a QTC location in Lebo, Kansas to be stored for 2 years in accordance with arrangements made between TVA and QTC. The part of Item V(A) which prohibited the destruction of the QTC files remains in effect. Item V(B), which required 5-working-day notification of NRC before QTC relinquishes control or custody of the unexpurgated original files, and Item V(C), which required TVA to permit inspection and copying of the unexpurgated original files remain in effect.

As a result of Item V(C) of the order, the staff made three copies of the original unexpurgated QTC files:

- (1) Set A is kept in a locked room in the Watts Bar Resident Inspector's Office, and is being used by region-based and resident inspectors during routine and special inspections, as necessary. Set A is also used for the special inspections being conducted under Temporary Instruction (TI) 2512/15 for the Employee Concern Special Program.
- (2) Set B was shipped to NRC headquarters for recordkeeping purposes and is being stored at the NRC Archival Facility as NRC Job Number 1077.
- (3) Set C was shipped to NRC headquarters for use by the staff. After the staff reviewed Set C (see below), it was destroyed in late 1987.

The NRC Office of Inspector and Auditor (OIA) removed 21 files involving NRC employees from all three sets before any NRC staff reviewed the records. The OIA subsequently returned most of the QTC files or sent a sanitized version of the files to Set C to be reviewed by the staff. The returned files were then most likely destroyed with the Set C files in late 1987. The staff has verified that TVA has been sent any safety concerns that existed in the 21 files that OIA removed, as evidenced by TVA's possession of the sanitized version of these files. The files removed by OIA were never returned to Set A at the Watts Bar Resident Inspector's Office. The OIA documented its investigation findings in a report.

On April 28, 1986, the staff began reviewing (screening) Set C of the QTC files. Procedures had been developed to ensure a consistent review and training of those who reviewed the files. One objective of the review effort

was to protect individual confidentiality, and to identify any additional information on safety-related issues to the TVA Inspector General. In a memorandum dated April 1, 1986, Victor Stello, then Executive Director of Operations, stated to the Commission (publicly available) that

Significant issues raised during the NRC screening effort would be selected for a more comprehensive staff evaluation. Also, specific issues which the NRC staff feels could compromise confidentiality if provided to TVA at any level, would be retained by the staff for NRC follow-up. The TVA resolution program and selected technical issues would be reviewed or inspected by the staff on a sampling basis.

This commitment was fulfilled when the staff issued a safety evaluation on October 6, 1987, as part of the Sequoyah restart effort. On July 25, 1986, the staff completed its screening of 2045 files containing 5237 TVA employee concerns. The staff completed its other followup activities in late August 1986.

The staff prepared a companion ("sanitized" or expurgated) file for each QTC file by deleting (1) any information that could identify individuals, (2) any technical information already in TVA's possession, and (3) information judged to be irrelevant to the particular technical concern. Copies of all companion files were sent to the TVA Inspector General's Office for a second confidentiality review before they were sent to the appropriate TVA organization for resolution. On October 2, 1987, as a result of a Freedom of Information Act request (FOIA-87-623), the staff placed a set of expurgated files in the public domain.

When the NRC staff reviewed Set C, it identified from the 5237 employee concerns 2437 safety-related issues requiring resolution. From the information provided by QTC, TVA already knew about most of these issues and was evaluating and resolving them. Of the 1130 issues classified as NRC items of interest, 126 were potential new issues, 481 were potentially significant issues, 391 were Sequoyah-related issues, and 132 issues were retained for staff review because of confidentiality or other considerations. The issues were separated into about a dozen broad categories, with electrical, welding, QA/QC, and harassment and intimidation (H&I) having the highest concentration of employee concerns.

TVA developed the Employee Concern Special Program (ECSP) in late 1985/early 1986 to address and resolve, among other things, employee concerns identified before February 1, 1986. The scope of the ECSP included the resolution of issues raised during interviews carried out by QTC, the old TVA employee concern program, and the TVA Nuclear Safety Review Staff (NSRS) issues. The ECSP separated the concerns into nine categories and issued a report on each category. The categories were divided into subcategories, and a report was prepared on each subcategory. The subcategories were divided into elements; element reports were written for Sequoyah only. The nine categories of concerns were (1) Construction, (2) Engineering, (3) Operations, (4) Material Control, (5) Welding, (6) Intimidation, Harassment and Wrongdoing, (7) Management and Personnel, (8) Quality Control/Quality Assurance, and (9) Industrial Safety. All of the categories were under the purview of the ECSP except the fifth, Welding, and the sixth, Intimidation, Harassment, and Wrongdoing. The welding concerns were assigned to the TVA Welding Task Group for followup, and the Intimidation, Harassment, and Wrongdoing category was assigned to the TVA

Office of Inspector General. These groups sent their conclusions to the ECSP. TVA transmitted an executive summary report and the nine category reports to the NRC on February 6, 1989.

As part of the ECSP, the TVA Employee Concern Task Group (ECTG) was assigned responsibility to resolve the concerns raised before February 1, 1986, including the QTC-identified concerns. The ECSP recorded issues from the employee concerns for which TVA had not taken corrective actions. The valid issues were identified in the Corrective Action Tracking Documents (CATDs) so the ECSP could track the corrective actions by line organization. The NRC reviewed the program, scope, organization, and methodology used by the ECSP to resolve the issues. The review effort was documented in Inspection Reports (IRs) (50-390, 391/85-49, 85-57, and 85-15, and 50-327, 328/86-08, 86-29, and 86-51). The staff also reviewed the ECSP manual (also known as the ECTG Program Manual). The Program Manual and its implementation were inspected. The inspectors concluded that ECTG members would find acceptable procedures and guidelines in the ECSP for evaluating and correcting the employee concerns that had been raised before February 1, 1986. The staff issued its safety evaluation on the review of the ECSP on October 6, 1986, as a part of the effort to restart Sequoyah Units 1 and 2. The safety evaluation stated that the staff would review the element reports for Sequoyah; and that for other plants, including Watts Bar, the staff would review the subcategory reports.

As a result of problems present in the TVA nuclear program and numerous employee concerns, the staff issued a letter, dated September 17, 1985, pursuant to 10 CFR 50.54(f) requesting that TVA submit information regarding the correction of management problems and site-specific problems. TVA responded with a set of plans: the Corporate Nuclear Performance Plan (Volume 1) issued prior to the restart of Sequoyah, and a Watts Bar Nuclear Performance Plan (Volume 4). The TVA submittals also described the Employee Concern Special Program and the Employee Concern Task Group. The staff reviewed the TVA submittals and issued two reports: NUREG-1232, Volume 1, dated July 1987, addressed corporate and programmatic problems, and NUREG-1232, Volume 4, dated January 1990, addressed problems specific to Watts Bar. NUREG-1232, Volume 1, restated the staff's position that because Sequoyah was the first TVA plant to restart, the staff would review the individual element reports. NUREG-1232, Volume 4, formalized the staff's intention to review the subcategory reports for Watts Bar. There are 107 subcategory reports that are related to Watts Bar. However, the staff withdrew that commitment in Watts Bar SSER 9 (NUREG-0847). The staff concluded that its commitment to review 15 of the subcategory reports had been obviated by its review of the 29 Corrective Action Programs and Special Programs (see details below).

In 1988 and 1989, TVA submitted Corrective Action Programs (CAPs, Section 1.13 of this report) to address 18 broad technical issues. In a meeting on January 18 and 19, 1989, the staff requested a comprehensive listing of the source documents that the 18 CAPs were designed to resolve. The source documents include CATDs, among other documents. TVA responded by a letter on July 13, 1989. In Inspection Reports 50-390, 391/89-14 and 90-05, the staff determined that the CAPs adequately addressed the technical issues identified in the source documents.

The NRC has continued to inspect the ECSP and the associated CATDs to ensure that the program has been implemented successfully. Temporary Instruction (TI) 2512/15, "Inspection of Watts Bar Nuclear Plant Employee Concerns," was

issued on November 11, 1985, to establish an inspection program for the QTC-received concerns. The TI committed the staff to perform at least two inspections, one when the TVA Nuclear Safety Review Staff (NSRS) received responses to about 40 percent of the concerns from the TVA line organization, and another when the NSRS received responses for a majority of the concerns. The staff would review approximately 20 percent of the safety-related concerns and approximately 5 percent of the non-safety-related concerns. One half of these concerns would be reviewed in depth. Approximately 20 percent of the reports completed by TVA would also be reviewed by the staff. Revision 1 (dated October 10, 1991) to the TI committed the staff to review about 20 percent of the safety-related ECSP CATDs. Approximately half of these CATDs would be reviewed in depth to verify that corrective actions have been implemented. The revised TI also required the staff to review the files that had information related to hardware removed to ensure employee confidentiality, and provide the information to Region II management.

The CATDs have been reviewed to assure that the employee concerns which the CATDs address have been corrected. In Inspection Report 50-390, 391/91-19, the staff concluded that the ECSP was being adequately implemented to support the restart of construction at Watts Bar. The inspection addressed the adequacy of the CATDs and the requirements for restart. The staff will continue to inspect TVA and Watts Bar in relation to the ECSP and CATDs. Other inspections carried out by the staff are documented in Inspection Reports 50-390, 391/90-24, 92-43, 93-06, 93-10, 93-16, 93-24, 93-58, 93-65, 93-72, 93-75, 93-83, 94-03, and 94-10. Although more than 20 percent of the CATDs have been reviewed to date, thus satisfying the commitment made in Revision 1 of TI 2512/015, the staff will continue its review. Because of problems discovered in CATD closure, the staff is performing a more rigorous review. On the basis of the adequacy of the CATD closeout, the extent of the review effort will be determined.

The staff reviewed QTC files in 1992. That review was specifically directed at ensuring that any hardware information previously withheld from TVA to protect employee confidentiality was identified and reevaluated by the staff to determine if the specific hardware deficiencies should be released to TVA (as stated in TI 2512/015, Revision 1). The staff's effort resulted in additional information being released on a number of concerns as identified in a letter to TVA dated April 24, 1992. TVA evaluated the additional information and sent the staff the results of those reviews in letters dated June 1, August 17, and November 23, 1992. The NRC staff will assess this material when it performs its "lookback review" (see below).

In a 1993 inspection (Inspection Report 50-390, 391/93-24 mentioned above), the staff discovered that a group of employee concerns did not have a formal program in place that tied the employee concern to the corrective action that resolved the concern. The group of concerns, known as "Class C" concerns, were described as "factual and identify a problem, but corrective action for the problem was initiated before the ECSP evaluation of the issue was undertaken." Because a corrective action had already been identified, the ECSP had taken no further action, even though the concern remained uncorrected. The staff concluded that this was a deficiency in the ECSP and identified it as a programmatic weakness. By letter dated August 20, 1993, TVA agreed to initiate a review of these employee concerns (the Employee Concern Special Program Lookback Review). The scope of the review was wider than just Class C concerns and included Class A, B, D, and E concerns as well. Class A concerns

are those that could not be verified as factual; Class B concerns are those considered valid but that did not require corrective actions; Class D concerns refer to problems that require corrective actions; Class E concerns refer to problems that were identified during the ECTG evaluation. Class D and E concerns are tracked by the CATDs. The TVA lookback review effort is scheduled to be complete in 1994. The staff continues to review this TVA effort through inspections. Inspections documented in IR 50-390, 391/93-75 approved the lookback plan and IR 50-390, 391/93-83 reviewed the initial implementation. The depth of the future review effort will be determined on the basis of the results of the lookback review. In a future SER supplement, the staff will update the results of its review effort.

The staff's effort in compiling this historical account was tracked by TAC M83202.

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

3.9 Mechanical Systems and Components

3.9.1 Special Topics for Mechanical Components

In SSER 6, the staff stated that the applicant performed a nonlinear elastic-plastic analysis of the feedwater system inside the containment in order to evaluate the pressure boundary integrity of the feedwater piping for the water hammer that would occur if the check valve slammed shut following a postulated rupture at the main header in the turbine building. For this analysis, the applicant proposed to use the rules in Appendix F of the ASME Code to develop acceptance criteria for the piping. However, the applicant also assumed that certain supports would fail when the loads in the analysis exceeded the support calculated capacities. The staff considered the applicant's method of analysis an open issue requiring further staff review (Outstanding Issue 20(a)).

A piping analysis in which support failures are postulated is normally not performed. However, the applicant has maintained that it would be difficult to modify certain supports inside the containment because of space limitations. During a site visit in September 1991, the staff confirmed the applicant's contention that space limitations existed for some supports inside the containment. In an August 4, 1992, letter, the applicant stated that "[w]here possible, supports were upgraded in the analysis to maintain structural integrity during the postulated loading scenario." Because of the difficulty in making additional support modifications, and because the pipe break and resulting check valve slam event is a low probability event which creates a very large impulsive load on the piping and pipe supports, the staff accepts the applicant's use of the energy absorption capability of the piping in performing the nonlinear elastic-plastic analysis with the use of the ASME Code, Appendix F-based allowable limits as alternative to the faulted condition allowable limits for this specific load case.

Brookhaven National Laboratory (BNL), the staff's contractor, reviewed the applicant's feedwater check valve slam analysis. A copy of the BNL technical evaluation report is enclosed in this supplement as Appendix BB. BNL concluded that the applicant adequately demonstrated that the feedwater loops meet the selected ASME Code criteria when subjected to the dynamic loads resulting from a check valve slam. On the basis of the conclusions from the BNL review, the staff finds that reasonable assurance exists that the feedwater lines inside the containment will maintain structural integrity when subjected to loads resulting from a postulated rupture of the main header in the turbine building. Therefore, Outstanding Issue 20(a) is resolved.

4 REACTOR

By letters dated October 10, 1991, and August 24, 1992, the applicant proposed changes to the Final Safety Analysis Report (FSAR) to reflect the proposed use of a new fuel assembly design, VANTAGE 5H (V5H). By FSAR Amendments 73 (May 21, 1993) and 80 (January 20, 1994), the applicant formally documented this design change. The new fuel design uses Zircaloy-4 fuel rods and grid spacers, and the staff has approved it for reload applications in a number of nuclear plants. The staff documents its evaluation of V5H for Watts Bar application in Sections 4.2.3, 4.3, and 4.4 below, and in Section 15.2.1 of this report. The staff's evaluation supplements previous evaluations published in the SER and SSERs, unless specifically stated otherwise.

By letter dated April 20, 1993, the applicant responded to the staff's March 15, 1993, request for additional information (RAI) about the Unit 1 draft Technical Specifications (TSs). Three of the applicant's responses are evaluated in Sections 4.2.1, 4.4.4.1, and 4.4.4.2 below.

4.2 Fuel System Design

4.2.1 Description

Question 1 of the staff's March 15, 1993, RAI inquired if the prepressurization of fuel is a design characteristic that should be specified in Section 4.0, "Design Features," of the Watts Bar TSs.

The applicant responded to the question and justified that the initial back-fill pressure is determined during the fuel rod design and is sized to, among other things, preclude flattening the cladding. The detailed fuel rod design establishes such characteristics as pellet size and density, clad-pellet diametral gap, gas plenum size, and helium pre-pressure. The design also considers such effects as fuel density changes, fission gas release, cladding creep, and other physical properties which vary with burnup. As reported in SSER 2, Section 4.2.2, the staff reviewed Westinghouse topical reports for information on internal fuel rod pressurization criteria, and found them acceptable.

The staff reviewed the applicant's response, and agrees that the requested information, already documented in SSER 2, is not key design information that needs to be included in the TSs. This effort was tracked by TAC M76742.

4.2.3 Mechanical Performance

The applicant has adopted the Standard Review Plan (SRP) approach in evaluating the V5H fuel design. This V5H fuel features standard fuel rods, debris filter bottom nozzle, reconstitutable top nozzle, and intermediate grid spacers with mixing vanes. NRC Information Notice 93-82, entitled "Recent Fuel and Core Performance Problems in Operating Reactors," pointed out the industry's experience involving V5H fuel damaged by vibrational fretting wear caused by a flow condition adjacent to the core baffle. The fuel vendor, Westinghouse, proposed short-term and long-term corrective actions. The Watts Bar reactor does not have a mixed-core situation. The applicant informed the

staff that Watts Bar fuel design and core loading have adopted the Westinghouse recommendation of short-term corrective action to address the vibrational fretting wear problem. The staff considers that the applicant's corrective action is acceptable for Watts Bar.

The applicant analyzed stress, strain, rod internal pressure, fatigue, and rod bowing based on approved methodologies for steady-state and transient conditions. The analyses showed that the V5H fuel will perform satisfactorily. The staff considers these analyses acceptable.

The applicant also analyzed the rod cluster control assemblies (RCCAs), control rod drive mechanisms (CRDMs), neutron source assemblies, burnable absorber assemblies, and thimble plug assemblies. The absorber materials used in the RCCAs are boron carbide pellets plus silver-indium-cadmium alloy. The burnable absorbers used are the Westinghouse-designed wet annular burnable absorbers (WABAs). All the RCCA and WABA designs have been approved previously by the staff. Therefore, the staff concludes that the RCCAs, WABAs, and CRDMs are acceptable for Watts Bar.

On the basis of approved mechanical methodologies, the staff concludes that the V5H fuel mechanical design for Watts Bar is acceptable. This effort was tracked by TAC M81887, M81888, M85774, M85775, M88644, and M88645.

In SSER 10, the staff documented its review of FSAR Amendment 65, and raised an issue about the methodology used to demonstrate seismic qualification of the CRDMs that did not appear to have been fully addressed during an earlier staff review of the FSAR. The staff based its concern on an apparent lack of documentation in the FSAR regarding CRDM qualification. Consequently, the staff asked the applicant to describe its basis for CRDM seismic qualification. The staff also asked the applicant to confirm that the revised design-basis seismic-response spectra for Watts Bar were included in the seismic qualification of CRDMs.

In a letter dated June 15, 1993, the applicant transmitted two letters from Westinghouse to TVA which address this issue. In these letters, Westinghouse stated that the CRDM design employed at Watts Bar is the same design that the staff accepted when it reviewed license applications from other Westinghouse plants. In addition, Westinghouse cited several CRDM generic testing programs that the staff accepted when it reviewed other Westinghouse plants, and specifically, when the staff performed a Seismic Qualification Review Team audit at Comanche Peak. Further, Westinghouse confirmed that it had evaluated the referenced tests to ensure plant-specific applicability for demonstrating CRDM seismic qualification at Watts Bar. On the basis of the information submitted in the applicant's June 15, 1993, submittal, the staff's concern regarding the adequacy of documentation is resolved.

With regard to Watts Bar's seismic response spectra, the applicant indicated during a conference call that Westinghouse proprietary report WCAP-13754, dated May 1993, "Qualification of the Reactor Internals, CRDMs, and CRDM Supports for Revised Seismic Spectra and the Addition of a Permanently Attached Head Shield Support Structure," documented an analysis which demonstrated that the design was adequate for Watts Bar's revised seismic response spectra. A portion of this report which summarized the results of the analysis, and which concluded that the CRDMs and supports were adequate to withstand the Watts Bar revised spectrum, was reviewed by the staff at the applicant's office.

Consequently, the staff's concern regarding consideration of the revised seismic spectra is resolved. It is noted that this report contained only a summary of the analysis results. The staff did not review analytical details and supporting engineering calculations which were referenced in the report.

On the basis of this discussion, the staff's specific concerns regarding CRDM seismic qualification at Watts Bar, which were reported in SSER 10, have been adequately resolved. The staff's efforts were tracked by TAC M84249 and M84250.

4.3 Nuclear Design

For the V5H fuel design, the applicant used such approved codes as ARK and PHOENIX-P/ANC to analyze shutdown margin. The analysis showed that the fuel design conforms to shutdown criteria. This is acceptable. The Watts Bar fuel pellet design has a thin layer of coated ZrB_2 on the pellet surface, a feature called "integral fuel burnable absorbers" (IFBAs), to control excessive reactivity during the beginning of the cycle. The IFBAs were approved earlier; hence this design feature is acceptable.

To gain more operating flexibility, the applicant analyzed axial power distribution on the basis of the procedures for constant axial offset control (CAOC) and relaxed axial offset control (RAOC). The hot channel factor, F_q , is maintained within acceptable limits. Additional operating flexibility is acquired by combining RAOC procedure with an F_q surveillance. Since the staff approved these methodologies previously, it concludes that the axial power distribution analysis is acceptable for Watts Bar.

The applicant calculated the fuel temperature coefficient by performing two group calculations using the TURTLE, PALADON, or the ANC code. Since the staff approved these codes earlier, it considers the fuel temperature coefficient analysis acceptable for Watts Bar.

The RCCAs are divided into two types: control groups and shutdown groups. Two criteria have been chosen to design the control groups: (1) the total reactivity worth must be adequate to meet shutdown margin requirements and (2) the total power peaking factor must be low enough to ensure that the power capability requirements are met. The applicant analyzed the control rod worth based on a conservative approach that the highest worth rod is stuck out of the core and the flux is skewed to the bottom of the core. The analysis showed that the two criteria were met, and the available reactivity for shutdown margin is adequate for Watts Bar.

Considering that the applicant used approved methodologies, the staff concludes that the nuclear design use of V5H fuel is acceptable for Watts Bar. This effort was tracked by TAC M81887, M81888, M85774, M85775, M88644, and M88645.

4.4 Thermal-Hydraulic Design

For the V5H fuel design, the applicant used the WRB-1 correlation for departure from nucleate boiling ratio (DNBR) calculations. In the SER, the staff stated that the applicant used the W-3 correlation; WRB-1 supersedes W-3. The

staff approved the WRB-1 correlation earlier; therefore, the applicant's calculated DNBR limit of 1.17 for V5H at Watts Bar is also acceptable.

A maximum rod bow penalty based on the approved methodology was incorporated in the DNBR analysis. See Section 4.4.4.1 for details.

The applicant analyzed the core thermal design using the approved THINC-IV code to determine the conditions in the hot channel and to ensure that the safety-related design bases are not violated. The analysis showed that the DNBR limits were met for steady-state and transient analyses. On the basis of the approved THINC-IV code and conservative results, the staff considers that the thermal design for Watts Bar is acceptable.

Considering that the applicant used approved thermal-hydraulic methodologies for its analysis, the staff concludes that the V5H thermal-hydraulic design is acceptable for Watts Bar. This effort was tracked by TAC M81887, M81888, M85774, M85775, M88644, and M88645.

4.4.4 Operating Abnormalities

4.4.4.1 Fuel Rod Bowing

Question 2 of the March 15, 1993, RAI asked the applicant to identify in the "Basis" section of the TS any plant-specific or generic margin used to offset the reduction in DNBR due to rod bowing, and to incorporate the residual rod bowing penalty into the TS.

By letter dated August 24, 1992 (submittal regarding use of V5H fuel design), the applicant addressed fuel rod bowing issues. The maximum rod bow penalties (<1.5% DNBR) accounted for in the design safety analysis are based on an assembly average burnup of 24,000 MWD/MTU. A 10.7-percent DNBR margin is maintained for the V5H fuel by comparing the DNBR limit of 1.31 to the WRB-1 correlation limit of 1.17. The applicant has incorporated the DNBR margin and residual rod bowing penalty into the "Bases for Safety Limits" section (Section 2.0) of the draft TSs. In the August 24, 1992, letter, the applicant stated that Table 4.1 ("Rod Bow Penalties") of the SER no longer applies; the staff concurs with the applicant in light of discussion in this paragraph.

The staff reviewed the applicant's response and finds it acceptable since an approved method (WCAP-8762-P-A, "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," July 1984) was used, and the DNBR margin and all rod bow penalties have been incorporated into an appropriate section of the TSs. This effort was tracked by TAC M76742.

4.4.4.2 Crud Deposition

Question 3 of the March 15, 1993 RAI reiterates a proposed requirement in the SER and asks the applicant to incorporate appropriate surveillance requirements in the TSs to recognize any rapid crud buildup in the reactor core.

In its April 20, 1993, response, the applicant stated that Westinghouse submitted information detailing how Westinghouse accounts for possible buildup of crud in determining safety limits, and noted means of tracking core operating parameters that might indicate, among other things, a rapid crud buildup. The

applicant also noted that there have never been TS surveillance requirements for any Westinghouse-designed plant that would indicate specific conditions related to flow reductions, power reductions, or temperature excursions that would rule out all possible operating anomalies with the exception of crud buildup. For the current draft TSs, Surveillance Requirement 3.2.2.1 of Limiting Condition for Operation (LCO) 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor (F_{NH}^N)," requires that (F_{NH}^N) be a measure of the maximum total power produced in a fuel rod. The Core Operating Limits Report (COLR, approved by letter of September 20, 1993, to the applicant) gives limits that ensure that the design-basis value of departure from nucleate boiling (DNB) is met for normal operation, operational transients, and transient conditions arising from events of moderate frequency. During power operation, the global power distribution is monitored by LCO 3.2.3, "Axial Flux Difference (AFD)," and LCO 3.2.4, "Quadrant Power Tilt Ratio (QPTR)," which address directly and continuously measured process variables.

The staff reviewed the applicant's response and agrees that the staff's SER statement proposing a TS surveillance requirement to monitor the buildup of crud is addressed through the surveillance of core operating parameters in LCOs 3.2.2, 3.2.3, and 3.2.4. This effort was tracked by TAC M76742.

7 INSTRUMENTATION AND CONTROLS

7.1 Introduction

By letter dated July 10, 1991, the applicant notified the staff of a design change that is planned for Watts Bar Nuclear Plant Unit 1 before fuel loading. The design change involves replacing the existing analogue Foxboro processor control system with Westinghouse Eagle-21 digital process protection equipment. In addition, by letter dated November 5, 1992, the applicant proposed changes that are planned for the FSAR to reflect the Westinghouse Eagle-21 process protection system. The proposed changes were incorporated in the FSAR by Amendment 72.

The proposed Eagle-21 digital process protection equipment upgrades the present reactor protection system (RPS) to include portions that involve micro-processor-based technology. The applicant proposed this upgrade to improve the reliability and accuracy of process data signals, to simplify calibration testing and maintenance of the electronics, and to accommodate future additions to improve control and monitoring instrumentation as they become available. The field instrumentation and cables are not part of the upgrade.

The sections that follow give details and conclusions of the staff's safety evaluation of the applicant's proposed Eagle-21 upgrade at Watts Bar Unit 1, and associated FSAR changes.

The applicant replaced portions of the analogue protection system with the Eagle-21 system in Unit 1 for the first time in 1989. At that time, the applicant replaced 4 of the 14 Foxboro processor protection and control instrumentation cabinets with 4 of the 14 Eagle-21 cabinets as part of an earlier design change that removed the resistance temperature detectors in the primary system bypass piping and manifolds. The staff published its safety evaluation as Appendix R in SSER 8. For the proposed upgrade, the applicant will replace the remaining 10 Foxboro cabinets with Eagle-21 equipment to complete the replacement of all of the Foxboro processor system.

The applicant stated in a letter of October 26, 1992 that the Eagle-21 system proposed for Unit 1 is almost identical to the one that was installed in the Sequoyah Nuclear Plant. The staff approved the Eagle-21 modification at Sequoyah on May 16, 1990. The only difference between the Sequoyah and Watts Bar Eagle-21 modification is that the Eagle-21 equipment at Watts Bar has no environmental allowance modifier (used at Sequoyah to reduce the uncertainty of possible reference leg heatup related to the steam generator low water level trip).

Between December 13 and 17, 1993, the staff inspected the instrumentation and control system of Unit 1 (see Inspection Report 50-390, 391/93-89). One of the purposes of the inspection was to inspect the installation of the Eagle-21 process protection system.

The Eagle-21 process protection system inspection consisted of selective examinations of procedures and representative records, interviews with

personnel, and observations of activities for installing the Eagle-21 process protection system. The staff audited the following documents: (1) seismic qualification testing report, (2) site acceptance testing report, (3) component testing report, (3) loop calibration procedures, (4) configuration control procedures, and (5) training guidance.

The inspection showed that engineering work on the Eagle-21 system was almost complete and that the applicant had incorporated the lessons learned from Sequoyah Eagle-21 system installation into the Watts Bar's Eagle-21 activities. The inspection determined that although no preoperational testing had yet been performed on the Eagle-21 system, functional testing of the Eagle-21 racks had been performed during site acceptance testing and component testing.

7.2 Reactor Trip System

The evaluation that follows supersedes or supplements the evaluations in the SER and SSERs. The staff's efforts were tracked by TAC M81063, M85037, and M85038.

7.2.1 System Description

Westinghouse Electric Corporation designed and manufactured the microprocessor-based Class 1E system, Eagle-21, to replace parts of the existing analog protection and control process instrumentation system. The Eagle-21 system consists of digital modules that perform many functions. All Eagle-21 system is configured to perform the basic functions of signal conditioning for protection and control instrumentation, and includes plant-specific functional modules added to the basic Eagle-21 system.

The Eagle-21 system's hardware consists of three subsystems per rack: loop processor subsystem, input/output subsystem, and tester subsystem. The system is designed to fit into existing analog process racks and to interface with other plant systems in a manner identical to the existing analog equipment. The Eagle-21 system also uses existing field terminals to avoid new cable pulls or splices within the cabinets. In addition, the Eagle-21 system processes the same inputs as the analog system and sends output signals to the reactor protection logic cabinets for initiating a reactor trip and engineered safety features actuation (ESF), and to indicators, recorders, the plant computer, and various control systems. The basic features of the Eagle-21 equipment include:

- (1) automatic surveillance testing capability
- (2) self-calibration (rack only) to reduce/eliminate rack drift and simplify calibration procedures
- (3) self-diagnostic capability to reduce troubleshooting time
- (4) modular design to allow for phased installation into existing process racks and use of existing field terminations
- (5) hardware expansion capability to accommodate functional upgrade and other plant improvements

The Eagle-21 software is arranged in a modular structure with all executable codes contained in a module/subroutine and is programmed in a high-level structured language. The software design implementation has no interrupts or reentries and uses coding standards for high-level and assembly language routines, high-level module logic, and single task programs (no multi-tasking).

The software format is configured in layers, with the main program consisting of general purpose and standard protection functions module layers that can be used in varied applications. The configuration module layer contains plant-specific information which configures the generic functions to project-specific applications. The configuration layer is application-specific and, therefore, generally the only layer requiring additional coding for each project. This configuration permits a significant amount of standardized codes from project to project.

All executable software is supplied in programmable read-only memory (PROM). Plant-adjustable tuning parameters (e.g., adjustable setpoints) are stored in nonvolatile memory for accessible adjustment on site.

The applicant stated that Eagle-21 software implementation at Watts Bar Unit 1 is very similar to the Eagle-21 installation at Sequoyah. The Eagle-21 system at Watts Bar Unit 1 performs few functions that are different from the functions performed by the Sequoyah Eagle-21 software. The Watts Bar Eagle-21 system incorporates all the improvements made since the Sequoyah Eagle-21 software was implemented. Although there are some differences between the Sequoyah and Watts Bar Eagle-21 main program and support routines, the applicant stated that the configuration layer of the Eagle-21 software at Watts Bar Unit 1 is nearly identical to the Eagle-21 software at Sequoyah.

As a part of the Eagle-21 update, the Watts Bar applicant will perform the following:

- (1) Update the electronics in the existing four Eagle-21 cabinets to the electronics used in the new Eagle-21 cabinet.
- (2) Add new steamline-break logic features to reduce the number of spurious actuations of safety injection at low power.
- (3) Reduce the steam generator low-low level trip setpoint to reduce the potential for unnecessary reactor trips due to feedwater operational transients.
- (4) Add a trip time delay designed to reduce unnecessary low-low steam generator water level reactor trips below 50-percent power.
- (5) Add the median signal selector (MSS) system (an analogue system) and eliminate the low-feedwater-flow reactor trip function to reduce the potential interaction between reactor protection and control systems.

The Eagle-21 system is part of the reactor protection system which includes the reactor trip functions and the engineered safety features actuation functions. Therefore, the general design criteria (GDCs), IEEE Standard 279, "Criteria for Protection Systems for Nuclear Power Generating Station" (10 CFR 50.55 a(h)); and the applicable acceptance criteria listed on Table 7-1 of the Standard Review Plan (NUREG-0800) are used as the staff's review guidance. In

addition, ANSI/IEEE Standard 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" are used to evaluate the Eagle-21 system design verification and validation process.

7.2.1.1 Updated Protection Features

New Steamline Protection Feature

The new steamline protection feature is part of the Eagle-21 system. This new steamline protection includes some of the engineered safety features actuation system (ESFAS) signals that exist on later-vintage Westinghouse plants as safety features against steamline breaks. The modification adds two new ESFAS functions to the Eagle-21 system and deletes two existing functions. The added functions are (1) safety injection (SI) and steamline isolation on steamline low pressure and (2) steamline isolation on steamline pressure high negative rate coincident with P-11 pressurizer pressure. The deleted functions are (1) SI and steamline isolation on steamline high flow coincident with P-12 low-low T_{avg} or steamline low pressure and (2) SI on steamline high differential pressure. Compared to the previous steamline protection feature, the new steamline break protection feature is intended to reduce the potential for spurious actuations of SI at low power.

The staff has previously approved a similar new steamline protection feature design for Sequoyah, Units 1 and 2. An amendment to each Sequoyah license was issued on May 16, 1990. The staff finds the Watts Bar steamline protection feature design to be consistent with the Eagle-21 reactor protection function design and, therefore, acceptable.

Reduction of Steam Generator Low-Low Level Trip Setpoint

The steam generator (SG) low-low level trip setpoint is reduced in order to reduce the potential for unnecessary reactor trips due to feedwater operational transients. The applicant stated that reanalysis of various feedwater accident and mass/energy release data demonstrated that the SG low-low level trip setpoint can safely be set at a constant value of 17 percent for all power levels. This constant 17-percent setpoint replaces the current programmed setpoint which varies linearly from 17 percent at 30-percent power to 54 percent at 100-percent power.

The setpoint is changed in the configuration layer of the Eagle-21 software. The staff finds implementation of the setpoint change in the Eagle-21 system to be consistent with the system design and, therefore, acceptable.

Steam Generator Low-Low Trip Time Delay

The steam generator low-low trip time delay (TTD) is part of the Eagle-21 system. The applicant proposed to add TTD to delay the signals that actuate reactor trip and start the auxiliary feedwater (AFW) pumps. This delay reduces the potential for unnecessary reactor trips and AFW actuations caused by the feedwater operational transients. The TTD is based upon the prevailing power level at the time the low-low level trip setpoint is reached and the number of steam generators that are affected. Reanalysis of the various

accidents that are assumed to be mitigated by a reactor trip or AFW actuation in response to an SG low-low level trip shows that TTD can safely be incorporated from 0- to 50-percent power.

The applicant stated that TTD at Watts Bar, Unit 1, is virtually identical to the TTD at Sequoyah. The staff approved the TTD for Sequoyah, Units 1 and 2 (amendment to the Sequoyah license and associated safety evaluation was issued on May 16, 1990). On the basis of the staff's evaluation of the Eagle-21 system at Watts Bar, Unit 1, and previous review of the Eagle-21 system at Sequoyah, the staff finds the incorporation of the TTD feature acceptable. In addition, the applicant has committed to submit the setpoint methodology document referenced in FSAR Amendment 72 (see evaluation in Section 7.1.3.1 of a future SER supplement). The staff will audit the postmodification test procedures and the test results to verify that the TTD design is properly implemented into the reactor protection system. The staff's efforts will be tracked by TAC M81063.

Median Signal Selector

The previous protection system trips the reactor on steam flow/feedwater flow mismatch coincident with steam generator low level, which provides a diverse trip function to the low-low steam generator water level trip. The latter is provided by three independent water level instrument channels for each steam generator upon 2-out-of-3 low-low water level in any steam generator. In the previous design, one of the water level instrument channels also sends a signal to the feedwater control system (FWCS). Thus, a common instrument channel serves both the reactor protection system and the feedwater control system. The diverse steam flow/feedwater flow mismatch trip function protects against potential interaction between the protection system and the control system that could affect the low-low water level trip.

The new median signal selector (MSS) system sends a signal to the FWCS by means of all three channels of steam generator level instrumentation. The MSS will select the median of the three channels as input to the FWCS to control feedwater flow into a steam generator. By selecting the median value, the control system will not be affected by a single failed instrument channel, thus avoiding potential protection system and control system interactions. With the addition of the MSS system, the applicant also proposed to delete the low feedwater flow (steam flow/feedwater flow mismatch) reactor trip function.

By letter dated November 5, 1992, the applicant submitted Topical Report WCAP-12417, "Median Signal Selector for Foxboro Series Process Instrumentation, Application to Deletion of Low Feedwater Flow Reactor Trip," to justify deleting the steam flow/feedwater flow mismatch reactor trip function. In that letter, the applicant stated that the MSS for Watts Bar is almost identical to that of Sequoyah. The topical report also addresses the issues related to the use of a median signal selector system, the hardware configuration, the operating principle, the reliability of the system, the capability for testing, and the adequacy of failure detection within the MSS system.

The staff was concerned that an undetectable failure in the MSS system may cause control and protection system interactions. In response, the applicant stated that Watts Bar's MSS has the capability for on-line testing. The MSS can be tested concurrent with the protection instrument channels providing

input to the unit. The applicant has committed to test the MSS system on a quarterly basis concurrently with the protection channels.

On the basis of the staff's previous review of Topical Report WCAP-12417 and the MSS system testability, the staff finds that the proposed MSS system is acceptable.

7.2.1.2 Watts Bar-Specific Issues

The Eagle-21 system is a modular microprocessor-based upgrade system with small modular variations to meet each plant-specific requirements. The staff reviewed the Eagle-21 generic issues and plant-specific issues for Sequoyah, Zion, and Diablo Canyon (letters to these plants transmitting the reviews are dated May 16, 1990; June 9, 1990; and October 7, 1993, respectively). For the reasons given above, the staff concentrated its evaluation on the Eagle-21 design changes made since Sequoyah, and on specific issues related to Watts Bar. Among the issues reviewed were (1) electromagnetic interference and radiofrequency interference (EMI/RFI), (2) electrostatic discharge (ESD), (3) overcurrent, (4) reliability of the software specific to the Watts Bar Eagle-21 system (this includes verification and validation (V&V) and configuration management), and (5) defense-against-common mode failures.

EMI/RFI Concerns

10 CFR Part 50 (Appendix A, GDCs 2 and 4), requires that safety systems be designed to withstand the effects of natural phenomena and to accommodate the effects of environmental conditions associated with normal operation and postulated accidents. To ensure that these effects will not interfere with the ability of the Eagle-21 system at Watts Bar, Unit 1, to perform its intended safety function(s), the staff reviewed the environmental qualification of the equipment for electromagnetic and radiofrequency interference (EMI/RFI).

EMI/RFI is random noise produced by systems within the operating environment in a nuclear plant. This random noise can affect the safety of the plant since it has the potential to lead to common-cause failure of redundant safety-related equipment that are particularly vulnerable to the noise. In addition, because the effects of EMI/RFI on the safety system are dependent on geometry, field strength of noise source(s), and other site-specific factors, the effects of EMI/RFI are site specific. Nevertheless, the same or similar EMI/RFI qualification methods can be used for difference sites.

Westinghouse qualified the Eagle-21 for EMI/RFI and documented the test and test results in WCAP-11733, "Noise, Fault, Surge and Radio Frequency Interference Test Report for Westinghouse Eagle-21 Process Protection Upgrade System." The staff found, during its previous review of WCAP-11733, that the Eagle-21 system was not sufficiently qualified for expected site EMI/RFI frequencies. WCAP-11733 only covers a frequency range from 20 MHz to 1 GHz and neglects lower frequencies.

In response to the staff's concern that a potential exists for random and unpredictable spurious effects on safety systems due to the ambient low-frequency EMI/RFI in the Eagle-21 environment, the applicant committed to perform tests and site surveys during hot functional testing to ensure that

the Eagle-21 system is qualified for its environment, and to submit the site survey results to the staff.

In addition, the applicant committed to maintain administrative control on the use of walkie-talkies, portable telephones, and temporary equipment in areas that already prohibit the use of walkie-talkies. However, the applicant is not planning to perform any site survey before installing the Eagle-21 system for the following reasons:

- (1) Watts Bar Unit 1 is not an operating plant.
- (2) The construction environment that currently exists at Watts Bar is not representative of an operating plant.
- (3) Eagle-21 system operating experience already exists at both units of Sequoyah.

The applicant's reasons for not performing a preinstallation test at Watts Bar Unit 1 are acceptable, and the commitment to perform postinstallation noise surveys is sufficient to demonstrate that the Eagle-21 is EMI/RFI qualified for the Watts Bar environment as is required by GDCs 2 and 4.

Electrostatic Discharge (ESD)

The staff evaluated electrostatic discharge (ESD) during previous Eagle-21 reviews. In response to the staff's concerns, the applicant stated in its December 27, 1993, letter, that it follows standard practices to minimize ESD when work is performed on the Eagle-21. These practices include the use of ESD mats, ground straps, and antistatic bags. In addition, the applicant stated that all personnel who work with the Eagle-21 system are trained to take precautions against ESD. Considering the applicant's approach to reduce ESD, the staff agrees that ESD precautions at Watts Bar are similar to those reviewed at other plants, and the staff, therefore, finds the applicant's approach acceptable.

Overcurrent

During the previous Eagle-21 review, licensees found that the increased in-rush current created by powering up three Eagle-21 racks simultaneously would exceed the overcurrent protection setpoints. In response to the staff's concerns, the applicant stated that Watts Bar's Eagle-21 power system will not experience overcurrent for the following reasons:

- (1) The Watts Bar Eagle-21 system has an automatic ac power time-sequencing load-distribution system. This load sequencing will limit the inverter loading to well below the overcurrent trip setpoints.
- (2) Unlike the Eagle-21 installations at other plants, Watts Bar did not consolidate Eagle-21 electronics into fewer racks.

On the basis of this explanation, the staff finds that there is reasonable assurance that the Watts Bar Eagle-21 system will not experience overcurrent during system startup, and the applicant's response is, therefore, acceptable.

Software Review: Verification and Validation, Testing and Configuration Management

By letter dated February 26, 1992, the applicant submitted Topical Report WCAP-12374, "Eagle-21 Microprocessor-Based Process Protection System," Revision 1, that provides a detailed technical description of the system. Appendix A to the report "Eagle-21 Replacement Hardware Design, Verification and Validation Plan," describes the Westinghouse-developed comprehensive verification and validation (V&V) program for its Eagle-21 product lines. Westinghouse used this V&V program for Watts Bar's Eagle-21 system. This program is consistent with those for previously approved Eagle-21 installations. Watts Bar's Eagle-21 V&V program includes the experience gained from the Eagle-21 systems at Sequoyah, Zion, Diablo Canyon, Turkey Point, and South Texas.

By letter dated November 8, 1993, the applicant submitted (1) WCAP-13191, "Watts Bar Eagle-21 Process Protection Replacement Hardware Verification and Validation Final Report," and (2) "Eagle-21 Process Protection System Verification and Validation Audit Report, Watts Bar Nuclear Power Plant Unit 1." WCAP-13191 contains the V&V processes that Westinghouse performs for the Watts Bar Eagle-21 system in accordance with Eagle-21 hardware design and the V&V plan. The V&V efforts focused on new, modified, and impacted codes, and the applicant took credit for previously verified and validated codes. As a part of the V&V process, Westinghouse also prepared problem reports whenever any anomalies were discovered during the source code reviews or during testing. The problem reports are of the following types: (1) unit-level problem reports which addressed anomalies specific to a single unit of code; (2) module-level problem reports which addressed anomalies covering entire modules; and (3) generic-problem reports which covered issues that spanned multiple modules. In addition, Westinghouse prepared validation-problem reports to address the problem specific to validation. Westinghouse prepared a total of 13 problem reports consisting of 6 unit-level problem reports, 2 module-level reports, 2 generic-level reports, and 3 validation reports.

The applicant reviewed Westinghouse's V&V plan and principal engineering documents for the Eagle-21 system as they were completed. The applicant also reviewed the V&V process and audited supporting documents to ensure that Westinghouse had developed the Eagle-21 software according to the software development process acceptable to the applicant. For the audit, the applicant conducted a broad review of the total Westinghouse Eagle-21 development process. The results of the audit, submitted with the letter of November 8, 1993, include:

- (1) confirming adequate resolution of the 13 problem reports from the V&V process of Watts Bar's software changes
- (2) confirming that the Westinghouse software development process satisfies ANSI/IEEE-ANS-7-4.3.2-1982
- (3) confirming that Westinghouse resolved the generic Eagle-21 system follow-up issues identified in the staff's Zion Eagle-21 safety evaluation
- (4) ensuring that software errors identified at Sequoyah or at other Eagle-21 installations have been resolved in Watts Bar's system software

- (5) identifying differences between the latest Watts Bar version of the Eagle-21 software and the earlier revision installed at Zion
- (6) evaluating that the Westinghouse software development process is in accordance with the NRC Software Audit Plan
- (7) determining the extent to which Westinghouse identified and resolved "hazards" (i.e., abnormal conditions) to which the Eagle-21 system might be subjected

On the basis of the audit results as documented in the applicant's report, the applicant accepted the Westinghouse V&V process used in the design of Watts Bar's Eagle-21 system.

Software testing is a part of the V&V process that is used for designing and developing the Eagle-21 system. Software testing for system verification consists of structural testing and functional testing. Structural testing ensures all source lines meet the intended design specification. Functional testing exercises the software to verify the expected system functions as specified in the design specification.

The software testing performed during the system validation consists of top-down functional requirement testing and prudency review/testing of the design and its implementation. The top-down functional requirements testing treats the system as a black box. Prudency review/testing, however, requires that the internal structure of the integrated software/hardware system be analyzed and tested in detail.

As stated above, the Eagle-21 software consists of standardized module layers. These standardized module layers are tested as part of the V&V process and used for various applications. However, when Eagle-21 software is updated, as in the case of Watts Bar Eagle-21 software, some parts of the software are retested to reverify and revalidate the modified software. The software is reverified and revalidated by testing affected elements of the main program and supporting software. An impact analysis identifies the software elements (i.e., system, subsystem, module, or unit) that need to be tested and determines whether existing (unchanged), new, or functionally modified codes must be tested. The analysis also determines how much retesting of existing codes must be done to reverify and revalidate the existing codes. The applicant stated that it will submit more information on this analysis later. In addition, the applicant stated during a conference call that Watts Bar's Eagle-21 software upgrades (modifications) are covered by at least one test case. The test results for the original and updated software tests are documented in WCAP-13191.

The staff reviewed these reports and concluded that the V&V program and its implementation as presented by the applicant is consistent with Regulatory Guide 1.152 and IEEE Standard 7-4.3.2, and is, therefore, acceptable. The staff's conclusion is based on (1) the staff's previous acceptance of the generic Westinghouse V&V plan, (2) the similarity between the Watts Bar design and previously approved designs, (3) completion of the Westinghouse action items identified during the applicant's software audit, (4) the results of the applicant's software audit, and (5) the staff's review of the applicant's audit and its results.

The executing software is supplied in PROMs with the tunable constants provided in nonvolatile RAM. All software code and software documentation is controlled by the Code Management System (CMS) at Westinghouse. Westinghouse established CMS as a part of its software management process that develops procedures and standards to control an evolving software system. Westinghouse used the same CMS process for the Watts Bar Eagle-21 codes as for the Sequoyah Eagle-21 codes. The applicant also maintains Eagle-21 configuration control drawings that specify jumper and switch configuration, board and software revision levels, and rack configurations. The instrument technicians at Watts Bar Unit 1 use these drawings to ensure that Eagle-21 boards and racks are restored to their approved design configuration after testing or maintenance.

Software changes, other than field-adjustable parameter changes, are to be made through a TVA-controlled modification process that utilizes Westinghouse as the librarian responsible for the storage and configuration control of the software. In addition, whenever the software is revised, Westinghouse will perform the modified software test as described above. As stated earlier, the software test for modified software had been previously accepted by the staff. The staff finds TVA's configuration management approach consistent with current national standards and, therefore, acceptable.

Defense Against Common-Mode Failure

The single-failure criterion requires that any single failure within the protection system shall not prevent proper protective action at the system level when required. Certain common-mode/common-cause failures are not considered to be single failures for the purpose of the single-failure analysis. However, such common-mode/common-cause failures can prevent the safety system from performing its intended function. For example, a software error may result in loss of multiple protection system channels since these are driven by common software. Common-mode/common-cause failures can also lead to the loss of more than one echelon of defense-in-depth provided by the monitoring, control, and reactor protection functions performed by the digital system. Microprocessor-based digital systems, such as the Eagle-21 system, have a potential for common-mode/common-cause failures in the area of software, hardware, and software-and-hardware interface. Defense against such failures is provided by quality and diversity in the digital system design. Diversity in component design and principles of operation is to be implemented to the extent practical in order to prevent the loss of the protection function due to common-mode failure.

By letter dated November 18, 1993, the applicant submitted WCAP-13869, Revision 1, "Functional Diversity Assessment for the Reactor Protection System/Engineered Safety Features Action System at Watts Bar Units 1 and 2," in response to the staff's concern regarding common-mode failure. This report is similar to those prepared on other plants using the Eagle-21 system. The protection system's responses for the failed Eagle-21 system consist of four categories. These categories are based on the following licensing-basis accidents and events:

Category 1 - Events that do not require Eagle-21 process protection equipment for primary and backup protection

The events in this category are: uncontrolled rod cluster control assembly (RCCA) withdrawal, RCCA ejection, RCCA misoperation, and inadvertent loading of a fuel assembly. The applicant stated that these events are not affected by a common-mode failure to the Eagle-21 system. The applicant stated that although events in this category may receive Eagle-21 process protection system signals, the primary protect system response and backup protect signals are generated by a system other than Eagle-21.

Category 2 - Events that do not require Eagle-21 process protection equipment for primary protection, but that assume Eagle-21 protection system signals for backup

The events in this category are: startup of an inactive reactor coolant pump (RCP), excessive load increase incident, inadvertent operation of the emergency core cooling system, and complete loss of forced reactor coolant flow. The applicant stated that these particular events are unaffected by a common-mode failure of the Eagle-21 system since the primary protection responses are derived through systems other than Eagle-21, or no protection system is required for core or reactor coolant system protection. The backup protection signals are provided by the Eagle-21 process protection system.

Category 3 - Events that require Eagle 21 process protection equipment for primary protection signals but that will receive automatic backup protection from other systems

The applicant stated that plant safety will be affected by a common-mode failure to the Eagle-21 system. However, it stated that backup protection signals would automatically provide the necessary protection functions through systems other than the Eagle-21 process protection system. The applicant also stated that with the exception of rod cluster control assembly (RCCA) withdrawal and main feedwater line break events, all the events in this category have been analyzed by Westinghouse in WCAP-8330, "Westinghouse Anticipated Transients Without Trip Analysis." In addition, the applicant stated that (1) the ATWS (anticipated transient without scram) mitigation system actuation circuitry (AMSAC) at Watts Bar consists of diverse equipment from the Eagle-21 system because it is a system using redundant programmable logic controllers (PLCs) supplied by the Atomic Energy of Canada Limited (AECL), with microchips for the PLCs provided by Sharp Electronics. Thus, the equipment diversity requirements of 10 CFR 50.62 for ATWS mitigation are met for Watts Bar, Unit 1. The applicant stated that:

- (1) The AMSAC system provides the required protection functions for rated thermal power (RTP) above 40 percent (C-20).
- (2) Turbine trip automatically scrams the reactor independent of Eagle-21 for RTP above 50 percent (P-9).
- (3) Generic analyses applicable to Watts Bar performed for ATWS events have demonstrated that the AMSAC system is not required to prevent reactor coolant system damage for RTP below 40 percent.

Below the C-20 power level, AMSAC system is not required to prevent reactor coolant system (RCS) damage. Additionally, with decreasing reactor power, the operator has more time to manually initiate any required protection function. For uncontrolled RCCA withdrawal power events, reactor protection is provided by the nuclear instrumentation system (NIS). For complete loss of load above 50 percent RTP, backup protection is provided by reactor trip on turbine trip, which is processed outside of the Eagle-21 protection system.

The events for which AMSAC was credited for providing the necessary protection functions are:

- (1) loss of normal feedwater
- (2) loss of offsite power to station auxiliaries
- (3) major rupture of a main feedwater pipe (below the C-20 setpoint with operator action required).

Category 4 – Licensing basis events that require Eagle-21 process protection equipment for primary and backup protection signals

These events receive both primary and backup protection signals from the Eagle-21 process protection system for some aspects of protection system response assumed in the safety analysis. The licensing-basis events that require Eagle-21 for primary and backup protection include:

- (1) loss of forced reactor coolant flow (for single reactor coolant pump locked rotor)
- (2) accidental depressurization of the reactor coolant system
- (3) loss-of-coolant accident (small- and large-break LOCA);
- (4) steamline break events
- (5) steam generator tube rupture

The applicant stated that alarms, indications, and controls are available to the operator, independent of the postulated failures in the Eagle-21 system, which provide the necessary information and controls for manual actions to provide the necessary protection system functions in the event of a common-mode failure in the Eagle-21 process protection system.

On the basis of the staff's evaluation of the applicant's defense-in-depth analysis, the staff finds that the applicant has provided adequate diversity and defense against postulated common-mode failures in the Eagle-21 system design, and should a common-mode failure of the Eagle 21 system occur, there exist appropriate diverse means to mitigate licensing-basis events.

7.2.6 Conclusions

On the basis of the staff's review of the information submitted by the applicant, including Amendment 72 concerning Eagle-21 reactor protection system design and associated instrumentation and control system modifications, and the similarities of the design and changes to those previously reviewed and

accepted, the staff concludes that the system meets the requirements of GDCs 2 and 4 for environmental qualification, the guidelines of Regulatory Guide 1.152 and IEEE Standard 7-4.3.2 for safety-related digital system design, and the requirements of 10 CFR 50.62 for ATWS mitigation system diversity for the reactor protection system. The staff, therefore, finds the Watts Bar Unit 1 Eagle-21 system acceptable.

7.3 Engineered Safety Features Actuation System

See Section 7.2.

7.3.1 System Description

See Section 7.2.1.

7.3.1.1 Updated Protection Features

See Section 7.2.1.1.

7.3.1.2 Watts Bar-Specific Issues

See Section 7.2.1.2.

7.3.6 Conclusions

See Section 7.2.6.

7.7 Control Systems Not Required for Safety

7.7.2 Bypassed and Inoperable Status Indication System

In SSER 7, the staff published its evaluation of the bypassed and inoperable status indication (BISI) system, and found it acceptable. By letter dated February 18, 1994, the applicant submitted new information resulting from its generic evaluation of a Sequoyah problem evaluation report (PER) dated September 16, 1992. That PER noted approximately 68 components that were not included in Sequoyah's BISI system, and no justification was documented for omitting them.

TVA's generic evaluation of the Sequoyah PER led the applicant to reevaluate the Watts Bar BISI monitoring device list, and on the basis of this reevaluation, TVA excluded 60 components from that list. The applicant's basis for excluding these components from the Watts Bar BISI monitoring device list is that these components will not be rendered inoperable more than once a year. The applicant reviewed plant procedures for performing periodic surveillance tests, inspections, maintenance, or other routine activities that would result in a component being unable to operate as required, and concluded that the components removed from the list comply with Position C.3(b) of RG 1.47, "Bypass and Inoperable Status Indication for Nuclear Power Plants," Position C.3(b),

The staff finds that the Watts Bar BISI system still meets all pertinent criteria noted in SSER 7, and concludes that the February 18, 1994 revision is acceptable. This effort was tracked by TACs M89026 and M89027.

8 ELECTRICAL POWER SYSTEMS

The staff published its safety evaluation on Chapter 8 of the applicant's Final Safety Analysis Report (FSAR) in the SER and SSERs 2, 3, 7, and 10. However, the applicant submitted numerous FSAR amendments and letters to change parts of Chapter 8 of the FSAR. These changes very often led to (1) invalidation of sections of the staff's previously published evaluation or (2) the need to supplement portions of the staff's previous evaluation. Having reviewed the FSAR up to Amendment 65, the staff issued a request for additional information (RAI) dated June 20, 1991, summarizing all the SER issues that remained open and additional issues that were raised by the applicant's numerous submittals since publication of the SER. In addition to the normal in-office review, the staff performed a site review on August 7 and 8, 1991, and a site audit from January 26 through 28, 1993. The next sections report the resolution of these open issues. The staff's efforts (reported below) were tracked by TAC M63649.

8.2 Offsite Electric Power Systems

8.2.1 Compliance With GDC 5

The following new material supplements the information in the SER pertaining to compliance with General Design Criterion (GDC) 5, "Sharing of Structures, Systems, and Components."

Grid Stability After Loss of Both Units

In Section 8.2.1 of the SER, the staff concluded that the applicant had complied with the requirements of GDC 5, "Sharing of Structures, Systems, and Components," with respect to sharing of circuits of the preferred power system. The staff based this conclusion in part on the assumption that for a design-basis event in one unit causing its trip and a single-failure trip of the remaining unit, the offsite system would remain stable enough that sufficient capacity would be available to safely shut down both units. In Section 8.2.2 of the SER, the staff stated that the applicant's grid stability analysis indicated that loss of both units itself will not cause grid instability. However, subsequent review of the applicant's analysis of grid stability (presented in Section 8.2.2 of FSAR Amendment 63) indicated that the grid will remain stable for loss of one unit but not for loss of both units.

The staff told the applicant about this concern in the RAI dated June 20, 1991. The applicant revised the FSAR by Amendment 71 to state that the grid will remain stable despite the loss of both units. The staff finds the revision acceptable, and confirms its evaluation in the SER.

8.2.2 Compliance With GDC 17

The material that follows revised the SER discussion editorially, but does not change the original conclusion:

Except as noted below, the applicant has complied with the requirements of GDC 17, "Electric Power Systems," with respect to the offsite power system's

(1) capacity and capability to permit structures, systems, and components important to safety to function; (2) provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit or loss of power from the onsite electric power supplies; (3) physical independence of circuits; and (4) availability of circuits.

The Watts Bar plant is linked to the electric grid system by six 161-kV transmission lines that terminate on an existing 161-kV switchyard (Watts Bar hydro plant switchyard) 1.5 miles from the plant. The 161-kV lines enter the switchyard by way of a number of physically separate and independent rights of way. In addition, five hydro-generators feed the switchyard. The four steam-driven generators (mentioned in the SER) are no longer in service.

The 161-kV switchyard consists of circuit breakers, disconnect switches, transformers, buses, and associated equipment arranged so that each incoming or outgoing transmission line can be connected to one or both main buses through circuit breakers. Switchyard protective relaying includes transmission line protective relays and switchyard bus differential relays. These relays are backed up by switchyard bus breakup relays and by switchyard circuit breaker failure relays.

Offsite power from the switchyard to the onsite Class 1E distribution system comes from two independent immediate-access circuits. Each of these two circuits is routed from the switchyard through a 161-kV transmission line and a 161-to-6.9-kV transformer (common station service transformers C and D) to the onsite Class 1E distribution system. The onsite Class 1E distribution system consists, in part, of two redundant and independent 6.9-kV shutdown boards in each unit, each capable of being fed from either of the above-described offsite circuits. Automatic fast transfers occur on 161-kV line or transformer faults.

Offsite power is normally supplied from the main generator through a 22.5-to-6.9-kV transformer (unit station service transformer) and 6.9-kV switchgear (unit start board) to the onsite non-Class 1E distribution system. For any unit generator trip, offsite power to the non-Class 1E system is automatically transferred from the normal supply to the two 161-to-6.9-kV transformers (common station service transformers A and B). Automatic fast transfers of non-Class 1E 6.9-kV boards between these two transformers occur on 161-kV line or transformer faults. Slow transfers occur on undervoltage conditions.

In its grid stability analysis, the applicant stated that the loss of the largest capacity generating unit connected to the grid, loss of the largest load from the grid, loss of the most critical transmission line, or loss of both Watts Bar units will not cause grid instability.

8.2.2.1 Availability of Offsite Power Circuits

The following new material supplements the information in the SER and SSER 2.

Design-Basis Information for the Watts Bar Hydro Plant Switchyard

In the SER, the staff concluded that the offsite power system circuits at the Watts Bar hydro plant switchyard comply with GDC 17 and were acceptable pending documentation in the FSAR of the additional information submitted by

letter dated October 9, 1981. In SSER 2, the staff reported that the additional information was documented in FSAR Amendment 48 and that this item was resolved. However, in Section 8.1.4 of FSAR Amendment 52, the applicant documented the physical independence of the two offsite circuits, but specifically excluded the Watts Bar hydro plant 161-kV switchyard from that discussion.

The staff told the applicant about this concern in the RAI dated June 20, 1991. The applicant incorporated in FSAR Amendment 71 that the two offsite power circuits are physically independent except for sharing the hydro switchyard. The staff reviewed the revision and finds that it continues to support the staff's original conclusion in SSER 2. Therefore, this concern is resolved.

8.2.2.2 Minimizing the Probability of Losing All AC Power

The following new material supplements the information in the SER, SSER 2, and SSER 3.

(1) Automatic Transfer Between the Offsite Power Sources and the Onsite Power Source for Degraded Grid Voltage Conditions

As discussed in Item 1 of Section 8.2 of SSER 3, Figure 8.3.5a of the FSAR was inconsistent with the staff's understanding of the design of the automatic transfer of Class 1E shutdown boards used during a degraded grid voltage incident. The staff originally concluded that the Watts Bar design provided an immediate transfer from the normal offsite source to the preferred offsite source on a degraded grid condition before transferring to the EDG. Because a degraded grid voltage condition could affect all offsite sources, an automatic transfer to the other offsite sources before transferring to the EDG could possibly damage equipment or delay response of the safety systems during accident conditions. By FSAR Amendment 52, the applicant clarified the FSAR to indicate that automatic transfer is directly to the diesel generator for a degraded grid voltage condition. The staff finds this direct transfer to the diesel generator acceptable.

(2) Automatic Transfer of Loads Between Preferred Offsite Circuits

In SSER 2, the staff stated that the testing requirement for the automatic transfer of loads and the design which prevents a faulted or overloaded bus from being automatically transferred was to be reviewed with the Technical Specifications. With respect to the automatic transfer from the first preferred power source to the second alternate preferred power source and automatic transfer of non-faulted buses, the applicant committed by letter dated April 17, 1985, to rack out the second alternate power circuit breaker on each of four Class 1E 6.9-kV buses. Thus, the review of testing requirements was no longer necessary.

Subsequently, the design was changed to utilize automatic transfers of Class 1E and other buses between offsite sources as described in other sections of this report. The applicant will rely on coordination of protective devices to ensure that faulted or overloaded buses are not transferred from one source to the other. The staff considers this item to be acceptably resolved without technical specification requirements for surveillance of the automatic transfer/overloaded bus transfer prevention circuitry. The proposed requirement in SSER 2 is withdrawn.

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

The applicant did not describe proposed testing for the automatic transfer from the offsite normal source to the preferred offsite source in the FSAR. The staff's acceptance of the new design is addressed below. The staff told the applicant about this concern in the RAI dated June 20, 1991.

During the site review of August 7 and 8, 1991, and subsequently in a letter dated September 13, 1991, the applicant stated that the automatic transfer from the normal A and B transformers to the preferred C and D transformers would be eliminated by a design change notice (M-12051-A). This new design was subsequently described in FSAR Amendment 71.

Common station service transformers C and D are each fed from separate and independent 161-kV lines from the Watts Bar Hydro Station on separate towers or poles. Each transformer has two 6.9-kV secondary windings with automatic tap changers. Each secondary winding is the normal feed for a separate Class 1E 6.9-kV shutdown board and the alternate feed for one of the other three shutdown boards. Automatic transfer from the normal feed to the alternate feed occurs only on a 161-kV line or transformer fault. Each shutdown board can be fed from a non-Class 1E 6.9-kV unit board as a maintenance source during periods when both units are in cold shutdown.

Staff review of the new design indicates that this aspect of the offsite power system fulfills the requirements for two independent, available offsite sources of GDC 17 and is acceptable. Therefore, this concern is resolved.

(4) The Use of the Second Alternate Offsite Circuit

In Section 8.3.1.1 of FSAR Amendment 63, the applicant indicated that the supply breaker to a 6.9-kV shutdown board from the second alternate preferred offsite source (one of four possible sources – the other sources being a 6.9-kV unit board, the other preferred offsite source, and an emergency diesel generator) would normally be "racked out," but implied that this second supply breaker may be used during some modes of plant operation. The staff told the applicant about this concern in the RAI dated June 20, 1991.

During the site review of August 7 and 8, 1991, and subsequently in a letter dated September 13, 1991, the applicant stated that the second supply breaker would not be racked out because the design had been changed. The applicant subsequently described the new design in FSAR Amendment 71. Review and acceptance of the new design is addressed in Section 8.2.2.2(3) above.

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

The physical separation and protection provided between any one of the main or common station service transformers and the two preferred offsite circuits had not been clearly described or analyzed in the FSAR. The staff told the applicant about this concern in the RAI dated June 20, 1991.

During the site review of August 7 and 8, 1991, and subsequently in a letter dated September 13, 1991, the applicant stated that the offsite circuits from the common station service transformers C and D are separately routed

underground. Drawing 71N200-1 (Revision 13) shows the underground routing. The applicant incorporated the proposed draft information in FSAR Amendments 70 and 71. Therefore, this concern is resolved.

(6) Separation of Associated Circuits

As discussed Section 8.2(3) of SSER 3, the 125-V dc control power for the off-site power circuits is supplied from the onsite Class 1E power system. The control power cables are treated as associated circuits and are routed in separate raceways. The staff found that this routing complied with the independence requirements of GDC 17 for offsite circuits and was acceptable.

By letter dated July 31, 1990, the applicant proposed to change Sections 8.1.5.2, 8.1.5.3, 8.2.1.6, 8.3.1.4.1, and 8.3.1.4.3 of the FSAR. The proposed revision to Sections 8.2.1.6 and 8.3.1.4.3 changed the allowable separation between the 125-V dc control power cables that are associated with the two independent offsite power circuits. The allowable separation was changed from "routed in separate raceways" to "separated such that they do not touch." The staff told the applicant about this concern in the RAI dated June 20, 1991.

During the site review of August 7 and 8, 1991, and subsequently in a letter dated September 13, 1991, the applicant agreed to go back to routing these cables in separate raceways. The applicant also indicated that any exception to this separation (i.e., separate raceways), when identified, will be justified and documented in the separation/isolation design criteria. The applicant incorporated this information in FSAR Amendment 71. Since this complies with the independence requirements of GDC 17, the staff's conclusion in SSER 3 is confirmed.

(7) Design-Basis Requirements for Offsite Circuit Control Power

In Section 8.2.1.8 of FSAR Amendment 63, the applicant stated that the design of the control power feeders to switchgear associated with common transformers C and D and 6.9-kV shutdown boards A-A and B-B ensures compliance with GDC 17, i.e., a loss of control power will not result in a loss of offsite power. On the basis of the information presented in the FSAR, it was not clear how the design that used the Class 1E vital dc system for control power for offsite circuits conformed to the following requirement of GDC 17:

Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

The staff told the applicant about this concern in the RAI dated June 20, 1991.

During the site review of August 7 and 8, 1991, and subsequently in a letter dated September 13, 1991, the applicant stated that it would revise the FSAR to state that (1) train A of one division supplies C transformer control power while train B of the other division supplies the D transformer control power and (2) coordinated breakers installed as part of the fire protection system provide protection between onsite and offsite circuits. The applicant incorporated the proposed information in FSAR Amendment 71. On the basis of its

review of the revised information, the staff finds that the design conforms to GDC 17; therefore, this concern is resolved.

8.2.2.3 Compliance With GDC 17 for the Duration of Offsite System Contingencies

The following new material supplements the information in the SER, SSER 2, and SSER 3.

In FSAR Amendment 63, the applicant proposed a new design for the offsite power system to be enabled only during offsite system contingencies when the supply of power is inadequate (i.e., loss of voltage on either 6.9-kV start bus A or B or for a 161-kV transmission system contingency). During system contingencies, an automatic load shedding scheme would be manually enabled and a number of alternate offsite supply breakers to balance-of-plant (BOP) boards or panels would be tripped and locked out. For a trip of both units, the load shedding scheme would be initiated to trip off part of the BOP loads.

Information presented in FSAR Amendment 63 to the FSAR indicated that implementing of the proposed new design would reestablish the offsite power system's compliance with the requirements of GDC 17 by supplying adequate capacity and capability following offsite system contingencies.

The staff told the applicant about the need for the new descriptive information in the RAI dated June 20, 1991.

During the site review of August 7 and 8, 1991, and subsequently in a letter dated September 13, 1991, the applicant stated that it would revise the FSAR to describe the circuits that will be used for load shed. Subsequently, in FSAR Amendment 71, the applicant described the load-shedding scheme which reduces loads from common station service transformers A and B.

Common station service transformers A and B each receive power from one of the two separate 161-kV lines from the 161-kV hydro switchyard which supply common station service transformers C and D. On tripping of the main unit's generator (except faults that cause the generator's output breaker to open), the BOP loads fast transfer from the unit station service transformers to offsite power through common station service transformers A and B.

When both units trip and there is a 161-kV supply contingency, selected non-Class 1E loads are shed from the station loads. The load-shed scheme is automatically initiated when both generators are tripped and the 161-kV contingency switch is in the contingency position or when both generators are tripped and an undervoltage condition occurs on a 6.9-kV start board.

The two redundant load-shedding circuits are located in 6.9-kV start board A (associated with common station service transformer A) and 6.9-kV start board B (associated with common station service transformer B). Each load-shed breaker receives a trip from each of the redundant load-shed circuits which are powered from separate 250-V dc batteries and battery boards.

The staff found that this aspect of the offsite power system fulfills the requirements of GDC 17 by providing adequate capacity and capability and is acceptable. Therefore, this concern is resolved.

8.2.2.4 Minimizing the Probability of a Two-Unit Trip Following a One-Unit Trip

The following new material supplements the information in the SER.

In FSAR Amendment 63, the applicant stated that upon any single unit trip, power from the offsite system is transferred from the normal circuits (via the unit transformers) to the preferred circuits (via the common transformers). The applicant also stated that after transfer, one train of each unit will be connected to the preferred circuits. The staff was concerned that on trip of one unit, the other unit would be vulnerable to trip because its source of offsite power was being changed from the normal circuit (which has not lost its capability as a source of offsite power) to its preferred source via the common transformers. The staff told the applicant about this concern in the RAI dated June 20, 1991.

During the site review of August 7 and 8, 1991, the applicant stated that when the design is changed to eliminate automatic transfer between offsite circuits, this item will no longer be applicable. In FSAR Amendment 71, the applicant described the design of the transfer of power sources on trip on a unit's main generator.

The normal source of power to each unit's 6.9-kV shutdown board is from either common station service transformer (CSST) C or D which is not affected by a unit trip. If a loss of voltage from either CSST C or D occurs, the affected shutdown boards are subsequently loaded onto their respective emergency diesel generators.

The normal source of power to each unit's 6.9-kV unit boards and reactor coolant pump boards is from a unit's main generator via one of the two unit station service transformers. Alternate supplies for these boards are available from the common station service transformers A or B. An automatic fast transfer occurs on a trip of the unit. Thus, a trip of one unit does not affect the other unit. Therefore, this design change resolves the staff's concern.

8.2.4 Evaluation Findings

In the SER, the staff opened Outstanding Issue 11 to track various open issues in Section 8.2. The open issues were resolved as stated above. Hence, Outstanding Issue 11 is closed.

8.3 Onsite Power System

8.3.1 Onsite AC Power System Compliance With GDC 17

In the SER, the staff indicated that each of the emergency diesel generators (EDGs) had a continuous rating, 2000-hour rating, and 2-hour rating of 4400 kW, 4750 kW, and 5225 kW, respectively. Subsequently, in Section 8.3.1.1 of FSAR Amendment 52, the applicant stated that the diesel generator rating was 4400 kW continuous, or 4840 kW for 2 hours out of 24 at a power factor of 0.8. The staff told the applicant about its concern regarding the changed EDG ratings in the RAI dated June 20, 1991.

In a letter dated September 13, 1991, the applicant stated that the FSAR would be revised to correctly reflect the EDG ratings, including the cold and hot

dead-load pickup ratings. Subsequently, the applicant incorporated the correct ratings by Amendment 75. Therefore, this concern is resolved, and the conclusion in the SER is confirmed.

8.3.1.1 Non-Safety Loads Powered From the Class 1E AC Distribution System

In the SER and SSER 2, the staff concluded that the design for offsite power, which provided a dedicated transformer for each preferred offsite circuit, was capable of supplying all connected loads. The staff found that the design conformed to GDC 17 and was acceptable as was described in Amendment 48 to the FSAR.

Subsequently, the applicant made additional design changes. These changes, as discussed in previous sections of this report and through Amendment 75 to the FSAR, have added capability by providing not only separate transformers for each offsite preferred source, but also by providing separate transformers for Class 1E and most other loads (BOP loads). Although some load-shedding is provided for the BOP loads if both main generators trip, and a 161-kV transmission system problem occurs (see Section 8.2.2.3 above), no credit is taken for this feature in system adequacy analyses. The staff concludes that the new design is acceptable with regard to the capability of the offsite system to supply all connected loads. The conclusion in the SER and SSER 2 is confirmed.

8.3.1.1.1 Diesel Generator Capacity

The following new material supplements the information in the SER and SSER 2.

In Section 8.3.1.1 of FSAR Amendment 63, the applicant stated that the worst-case loading for the diesel generators occurs for a simultaneous loss of off-site power and a loss-of-coolant accident on the associated unit. Results of analysis which demonstrated that adequate margin exists, in all cases, between worst-case loading and diesel capacity had not been presented in the FSAR. The staff told the applicant about its concern in the RAI dated June 20, 1991.

During the site review of August 7 and 8, 1991, the applicant stated that it would provide some margin but would not commit to the extent of the margin. Loading will not exceed the design rating of the machines. The same process used for Sequoyah and Browns Ferry will be used at Watts Bar for running an actual load test (simulated loss of offsite power and loss-of-coolant accident (LOCA)) with loads to within 3 percent of actual load. The applicant stated that the auto-connected loads would equal the 2-hour rating, but the transient load on the diesel is such that the load sequence is being changed to accommodate transients. After 2 hours, loads that are not needed, such as non-safety loads would be shed manually or automatically.

In FSAR Amendment 75, the applicant compared the worst-case loading to the EDG ratings. The loads were less than the 2-hour ratings, but exceeded the continuous rating by about 2 percent for EDG 2B-B. Since the loading values are based on predicted loads, the loads do not exceed the 2-hour ratings, and the applicant's Loss-of-Offsite-Power Procedure AOI-35 directs the operator to check for EDG overload and contains guidance for reducing loads. This is acceptable, and the staff's concern is resolved.

8.3.1.2 Low and/or Degraded Grid Voltage Condition

The new material supplements the information contained in the SER, SSER 2, and SSER 7.

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

In the SER and SSER 2, the staff stated that the Watts Bar design was in conformance with Position B-2 of Branch Technical Position (BTP) PSB-1 and was, therefore, acceptable. After reviewing the FSAR through Amendment 63, the staff reconfirmed this conclusion. However, with regard to technical specifications requirements specified by Position B-2 of BTP PSB-1, it was not clear that the proposed design using an inverse time-delay-type relay could reliably be shown to not trip above a maximum voltage of 4860 volts but would trip below some minimum voltage level which has not been specified in the FSAR. To clarify this issue, the staff sought additional information in its RAI dated June 20, 1991. Resolution of the staff's concerns is as follows:

- Reliability data which demonstrate that an inverse-time-type relay will not trip at voltages above the 70-percent voltage level of 4860 volts

During the site review on August 7 and 8, 1991, the applicant stated that only the load shed and diesel generator start relays are of the inverse-time type. The degraded voltage and loss of voltage relays are solid-state-type relays. The diesel generators are in full compliance with the position in Regulatory Guide 1.9 on frequency and voltage decreases, i.e., at no time during the loading sequence does the frequency or voltage decrease to less than 95 percent of nominal and 75 percent of nominal, respectively. An accuracy calculation for the load-shedding relays was performed, and 74 percent was the maximum upper limit. The results of the accuracy calculation will be used to support the setpoint specified in the Technical Specifications.

- Technical Specifications limits for maximum and minimum limits that will be in the Technical Specifications

During the site review of August 7 and 8, 1991, the applicant stated that the setpoint for the load-shed relays was 4860 volts with a 5-second time delay to close contacts on complete loss of voltage. The staff will require the applicant to include the setpoint in the Technical Specifications for the load-shed relays and similar minimum limits for the diesel-start relays. Maximum limits for the load-shed relays will be included in the Technical Specifications based on the results of the applicant's analysis. At 75-percent voltage, the load-shed relays should not actuate to strip loads. Thus, the Technical Specifications should require, for example, that the capability of the relays not to trip when subjected to a voltage of 75 percent for 30 seconds be periodically demonstrated.

These items have now been included in the draft Technical Specifications. Therefore, the staff's concerns are resolved.

(2) Compliance With BTP PSB-1

In the SER, the staff stated that it would verify the adequacy of the applicant's analysis regarding compliance with BTP PSB-1 once the preoperational test was completed.

The staff had confirmed that a preoperational test had shown that the Watts Bar design conforms with BTP PSB-1 (see Inspection Report 50-390/84-90, dated February 11, 1985). Hence, Confirmatory Issue 28 was resolved. However, due to design changes, the results obtained from the previous test are no longer valid and the applicant is performing preoperational tests again. The staff will review this issue when it inspects the applicant's preoperational test program.

8.3.1.7 Possible Interconnection Between Redundant Divisions Through the Normal and Alternate Power to the Battery Charger

In the SER, the staff looked for possible interconnections for components that can be transferred between redundant divisions. For each component that can be transferred, the applicant documented the commitment to keep the alternate feed breaker at the 480-volt shutdown boards open, to install an alarm in the control room when this alternate feed breaker is closed, and to install an alarm in the control room when the manual transfer switch is in the alternate position. On the basis of this commitment, the staff concluded that the applicant's transfer design offered sufficient independence between redundant divisions, assured correct system alignments, conformed to the requirements of GDC 17, and was acceptable.

However, by FSAR Amendment 56 (Section 8.3.1.1), the applicant stated that the alternate feed breakers at the 480-volt shutdown boards would normally be kept open except for three particular cases where the power supply alignment is not important. Also, the applicant stated that the alternate feed breakers would be verified open in accordance with the Technical Specifications (i.e., no control room alarms will be installed on the alternate feed breakers or on the manual transfer switches).

The staff told the applicant about its concerns related to manually transferring loads between redundant divisions in the RAI dated June 20, 1991.

During the site review of August 7 and 8, 1991, the applicant stated that when a battery charger or inverter loads or both are transferred to the other redundant division, the design would not be in compliance with the requirements of GDC 17 because the diesel generator does not have sufficient capacity to supply these additional loads. Surveillance requirements were provided in the draft Technical Specifications for monitoring the position of these supply breakers every 7 days. Therefore, for the battery charger and inverter loads, this item is considered acceptably resolved.

For the remaining loads, the applicant stated that it would prepare an analysis or supply additional information with acceptance criteria which demonstrates that the remaining shared loads fulfill the requirements when supplied from the alternate source.

In a September 13, 1991, letter, the applicant submitted a draft to the FSAR which reflected the acceptance criteria for alternate feeders. The applicant incorporated the proposed draft in Amendment 71 to the FSAR. The applicant also submitted a discussion and calculations which demonstrated that alternate sources have sufficient capacity when supplying shared loads (except battery chargers and inverters mentioned above). Therefore, the staff's concerns are resolved.

8.3.1.10 No-Load Operation of the Diesel Generator

The following new material supplements the information in the SER.

Section 8.3.1.1 of FSAR Amendment 55 indicates that the diesel generator manufacturer recommends that the diesels should not be run for extended periods of time at less than 50 percent of continuous rated load. The FSAR also states that diesel engines have been tested for no-load operation for 4 hours. After 4 hours of operation at less than 30-percent load, the diesel generator will be run at a minimum of 50-percent load for at least 30 minutes. Also, after an accident situation when the diesel generator has run for an extended period of time at low load or no load, the load is to be gradually increased until the exhaust smoke is approximately twice as dense as normal. The increasing load is then stopped until the smoke clears. This procedure is repeated until full load can be carried with a clear exhaust. For all situations, the applicant has loads continuously available to the operator that exceed 50 percent of the continuous rated load.

The staff was concerned that the diesel generator may not have sufficient capability to supply required power within the specified accident analysis time limits following an accident, prolonged operation of the diesel generators at no load, and a loss of offsite power. The staff told the applicant about these concerns in the RAI dated June 20, 1991. The concerns were resolved as follows:

- (1) The manufacturer's test results that verify the capability of the diesel generator to accept design-basis loads within the specified accident analysis time limits following 4 hours of no-load operation

During the site review on August 7 and 8, 1991, and subsequently in a letter dated September 13, 1991, the applicant stated that the diesel generator ratings are not degraded as a result of no-load operation and if the condition warrants, the diesel generator may be loaded as if it had just been started. Although the applicant does not have a manufacturer's test to demonstrate this capability of the diesel generator, previous industry testing has indicated that only after months of no- or low-load operation did damage to turbochargers occur. Therefore, the only industry recommendation is that diesel generators be loaded before shutdown following extended no-load operation.

In addition, the applicant stated that concurrent loss of offsite power with a LOCA is the design basis for the plant. Thus, the need for the diesel generator after its prolonged operation at no load due to an accident is bounded by the current design basis. This response is acceptable and this issue resolved.

- (2) The accident analysis results which demonstrate that sufficient time is available to load the diesel generator slowly as described in the FSAR after 4 or more hours of operation of the diesel generator at no load

In the letter dated September 13, 1991, the applicant stated that the specific loading information would be deleted from the FSAR. This was done in Amendment 67. The applicant also stated that the EDG will be automatically loaded with accident loads after an accident, and delayed loss of offsite power, if the safety injection (SI) signal has not been reset or with nonaccident loads if the SI signal has been reset. This information resolves this issue.

- (3) A description of the administrative procedures for assuring that the diesel engine will never be expected to provide power to accident loads after the diesel engine has operated for more than 4 hours under no-load conditions

During the site review on August 7 and 8, 1991, and subsequently in a letter dated September 13, 1991, the applicant stated that operating procedures have certain requirements to prohibit operating the EDG below 60-percent load except for an emergency. This response resolves this issue.

- (4) During accident conditions, it is a standard technical specifications requirement that onsite and offsite power circuits not be paralleled. For this condition of operation (i.e., the offsite circuits are supplying loads required to safely shut the reactor down following an accident), describe how the available loads will be connected to the diesel generator so it will not be operated at no load for prolonged periods of time. Also describe the circuit design that automatically makes available the onsite power source in the event of a loss of offsite power.

During the surveillance testing when the EDG is parallel to the offsite power, the operator is monitoring/controlling the EDG loading. How the EDG responds during the loss-of-offsite-power (LOOP) and LOOP/LOCA scenarios while parallel to the offsite grid is addressed in Section 8.3.1.12 herein. In the September 13, 1991, letter, the applicant stated that the loads would automatically be supplied power via the load sequencer once the EDG is separated from the offsite power system. On the basis of this response and those provided for concerns 1 through 3 above, this issue is resolved.

8.3.1.11 Test and Inspection of the Vital Power System

The following new material supplements the information in the SER.

In Section 8.3.1.1 of FSAR Amendment 63, the applicant stated that several equipment characteristics and components would be tested before placing the vital ac system in operation, but would not be subsequently tested during plant operation. In order to complete the staff's review, a substantiating analysis was required which would demonstrate that there would be reasonable assurance that the associated equipment would maintain its capability to perform its design-basis safety function over the operational life of the plant without further testing of the characteristics and components. The staff told the applicant about this concern in the RAI dated June 20, 1991. Resolution of the staff's concerns follows:

- (1) The stability of output voltage and frequency of each inverter

During the site review on August 7 and 8, 1991, and subsequently in a letter dated September 13, 1991, the applicant stated that it verifies the voltage and frequency every 18 months during load testing.

- (2) The capability of the inverter to deliver 100 percent of its output while operating on either the normal or emergency supplies

During the site review on August 7 and 8, 1991, and subsequently in a letter dated September 13, 1991, the applicant stated that every 18 months, the inverter is loaded to 20 kW (100% of rated load) at 1.0 power factor and

transfers to the emergency 125-V dc supply while maintaining voltage and frequency.

(3) Calibration of the panel and board-mounted instruments

During the site review on August 7 and 8, 1991, and subsequently in a letter dated September 13, 1991, the applicant stated that it calibrates the panel and board-mounted instruments every 18 months.

(4) Proper trip operation of circuit breakers on each instrument power board

During the site review on August 7 and 8, 1991, the applicant stated that all of the breakers are not tested, since Class 1E boards are not required to be tested. Subsequently in a letter dated September 13, 1991, the applicant stated that it will test circuit breakers on each instrument power board using test sets to simulate fault currents. A draft markup of the FSAR was submitted.

(5) Fuse verification with respect to size and type

During the site review on August 7 and 8, 1991, and subsequently in a letter dated September 13, 1991, the applicant stated that plant Administrative Instruction 10.10 provides for control of fuses along with the master fuse report drawing series 45B6000.

The applicant incorporated the proposed draft discussed in concern 4 (above) in FSAR Amendment 71. Therefore, the staff's concerns are resolved as described above in concerns 1 through 5.

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

The following new material supplements the information in the SER.

In FSAR Amendment 63, the applicant stated that for test purposes, a diesel generator may be manually paralleled with the normal or preferred power source. While offsite and onsite sources are paralleled, it was also indicated that both a loss of offsite power and a safety injection signal would be required to automatically override the manual controls and establish the appropriate alignment.

When there is a safety injection signal alone without a loss of offsite power or when there is a loss of offsite power by itself without a safety injection signal, the staff was concerned that the capability and independence of offsite and onsite sources may be compromised when they are paralleled during testing. The staff told the applicant about this concern in the RAI dated June 20, 1991, specifically asking the applicant to demonstrate in accordance with the requirements of GDC 17 that appropriate provisions have been included in the design to minimize the probability of losing both offsite and onsite power supplies given an accident or loss of offsite power during testing with the offsite and onsite sources paralleled.

During the site review on August 7 and 8, 1991, the applicant stated that loss of the offsite power supply would cause the instantaneous overcurrent relay to trip the EDG output circuit breaker, the loss of voltage relays to trip the

supply breaker and loads, and subsequently the diesel generator load sequencer to load the shutdown board with the non-LOCA loads. If an accident signal was initiated during testing of the standby supply, the parallel connection would be maintained unless loss of offsite power also occurs. Should a LOCA and a LOOP event occur when the diesel generator was parallel with the grid under test, the same sequence of events would take place as loss of offsite power, except the diesel generator sequencer would load the accident loads. Only one diesel generator will be in the test mode at any given time.

In a letter dated August 5, 1993, the applicant reiterated what was stated above and submitted specific details pertaining to how an EDG being tested in parallel with offsite power would respond during LOOP and LOOP-with-LOCA scenarios. As stated above, during a LOOP event, the diesel generator's output breaker would trip on overcurrent. The instantaneous overcurrent relay would reset automatically and as soon as load shedding was completed, the EDG's output breaker would close since the diesel generator is already running at rated speed. For a LOOP concurrent with a LOCA, the same scenario applies (with a different load sequence) with the EDG loading being initiated before the design-basis time (10 seconds).

Subsequently, the applicant stated in a phone call initiated per 10 CFR 50.9 that this information was wrong. Further, the applicant stated that it was considering a design change. In a letter dated February 7, 1994, the applicant discussed the design change which modifies the EDG control circuitry so that the output breaker for the EDG that is being tested parallel with the grid will trip on a safety injection signal. This, in turn, will realign the EDG to its emergency mode and override the test controls. For a LOOP condition, the EDG's output breaker will trip on a fault signal associated with the normal offsite power which, in turn, de-energizes the associated 6.9-kV shutdown board leading to the load-shedding sequence.

The staff's review of the applicant's letter dated February 7, 1994, raised questions. The staff transmitted these questions to the applicant in a letter dated March 28, 1994. The staff will review the applicant's response under TACs M89109 and M89110, and will report its evaluation in an SER.

8.3.1.13 The Use of an Idle Start Switch for Diesel Generators

The following new material supplements the information in the SER.

In Section 8.3.1.1 of FSAR Amendment 57, the applicant stated that an idle start switch would start and run the diesel engine at idle speed for periods of unloaded operation. During this type of operation, any emergency signal will cause the engine to go to full speed and complete the emergency start sequence.

The applicant did not present detailed design and testing information in the FSAR demonstrating that the idle start switch will not degrade below an acceptable level the capability of safety systems to perform their safety functions. The staff told the applicant about this concern in the RAI dated June 20, 1991.

During the site review of August 7 and 8, 1991, the applicant stated that a local idle start switch is provided by the diesel manufacturer to start and run the engine at idle speed for extended durations of unloaded operation.

Use of the local idle start switch is enabled by a permissive signal from the main control room. During idle operation, any emergency start signal will disable the idle start circuitry and will command the engine to go to full speed and complete the emergency start sequence.

FSAR Figure 8.3.25 provides the design details for the idle start switch. Also, the information was incorporated into the FSAR by Amendment 71. The applicant stated that the circuitry for bypassing the idle start switch on an accident signal would be periodically tested per the Technical Specifications as part of EDG testing. Therefore, this concern is resolved.

8.3.1.14 Master Fuse List Program

The following new material supplements the information in the SER.

The staff published its evaluation of the applicant's master fuse list program in Appendix U of SSER 9.

8.3.2 Onsite DC System Compliance With GDC 17

8.3.2.2 DC System Monitoring and Annunciation

As discussed in SSER 3, the staff determined that the following items had been omitted from the design of the Watts Bar diesel generator dc monitoring and annunciation system. Proposed License Condition 13 was kept open to track completion of these issues:

- (1) Battery circuit input current is not monitored.
- (2) DC bus undervoltage is not alarmed in the control room.

In Section 8.3.1.1 of FSAR Amendment 63, the main control room alarm for battery charger output breaker open was removed. In addition to the two items above, justification was required which demonstrates that the diesel generator dc system is adequately monitored without the battery charger output breaker being alarmed in the main control room.

The staff told the applicant about these concerns in the RAI dated June 20, 1991.

During the site review of August 7 and 8, 1991, and subsequently in a letter dated September 13, 1991, the applicant stated that the battery circuit input current is monitored locally. The dc bus undervoltage is monitored indirectly through the battery charger low output voltage alarm in conjunction with the battery discharge alarm. The battery charger output breaker open is alarmed by charger output current alarm or by the battery discharge alarm. On the basis of this information, the staff's concerns are considered acceptably resolved, and proposed License Condition 13 is deleted.

8.3.2.4 Diesel Generator Battery System

In the SER and SSER 2, on the basis of information presented through FSAR Amendment 48, the staff concluded that the diesel generator 125-V dc control power system was acceptable. However, after subsequent revisions to the FSAR documented by Amendments 57, 63, and 65, it was no longer clear that the design continued to be in compliance with the applicable GDCs, regulatory

guides, and IEEE standards. The staff told the applicant about this concern in the RAI dated June 20, 1991.

During the site review of August 7 and 8, 1991, the applicant stated that compliance of the diesel generator battery system with applicable GDCs, regulatory guides, and IEEE standards is stated in FSAR Section 8.3.1.1. The applicant incorporated this information in FSAR Amendment 70. This resolves the staff's concerns.

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

In the SER, on the basis of information presented through FSAR Amendment 48, the staff concluded that the powering of non-safety loads from the dc distribution system would not degrade the Class 1E systems below an acceptable level and was, therefore, acceptable. This conclusion was based in part on the applicant's statement in FSAR Section 8.3.2.1.1 that the batteries have the capacity to supply all connected loads (Class 1E and non-Class 1E) for a minimum of 2 hours, and that the batteries will be tested periodically in accordance with the Technical Specifications to ensure this capacity.

Subsequent review of information presented in FSAR Section 8.3.2.1.1 through Amendment 65 revealed that batteries may not have sufficient capacity to supply all connected (or connectable) loads as originally concluded in the SER. In order to clarify or establish the minimum design-basis requirements for battery capacity with margin and in order to determine required administrative procedures to ensure nonessential loads do not use any of this minimum required design-basis capacity, the staff sought additional information from the applicant in the RAI dated June 20, 1991. The staff's concerns are resolved as follows:

- (1) As a design-basis requirement, define the load currents each battery is expected to supply for specified time periods for all modes of plant operation (i.e., battery duty cycle as defined in IEEE Standard 485). An accident with loss of offsite power, loss of offsite power without an accident, and loss of both offsite and onsite ac sources without an accident (i.e., station blackout) should be included in the duty cycle.

During the site review of August 7 and 8, 1991, and subsequently in a letter dated September 13, 1991, the applicant stated that the batteries have been sized for loss of both offsite and onsite ac sources without an accident for 2 hours, an accident with loss of offsite power and chargers plus a single failure for 30 minutes, and full-load rejection with no accident or loss of power.

The station blackout load profile has been changed to 4 hours with manual load shedding after 30 minutes into the event. Battery sizing for this load profile has been completed. The NRC documented its station blackout review in a letter dated March 18, 1993. The staff submitted supplement to that SER in a letter dated September 9, 1993. This issue is resolved.

- (2) Define the design margin that will be maintained for the capacity of the battery and/or the size of loads included in the battery duty cycle.

During the site review of August 7 and 8, 1991, subsequently in a letter dated September 13, 1991, the applicant stated that margin would be greater than or

equal to zero. The current margin is 5 percent. The battery loading calculation assumes the vital inverters are fully loaded, although the inverter loading calculation indicates actual loading to be 60 to 90 percent. This response adequately resolves this issue.

- (3) Describe how the battery capacity with margin will be maintained and periodically verified over the life of the plant.

During the site review of August 7 and 8, 1991, and subsequently in a letter dated September 13, 1991, the applicant stated that battery performance would be verified periodically by test. On the basis of this response, this item is resolved.

- (4) Describe the additional capacity that has been provided in each battery for nonessential loads.

During the site review of August 7 and 8, 1991, and subsequently in a letter dated September 13, 1991, the applicant stated that nonessential loads are not separated from safety loads in the sizing calculations. On the basis of this response the staff's concern is resolved.

- (5) Describe the administrative procedures and/or other provisions that will be implemented to ensure nonessential loads do not use any of the capacity required for essential loads for all modes of plant operation.

During the site review of August 7 and 8, 1991, and subsequently in a letter dated September 13, 1991, the applicant stated that nonessential loads are not separated from safety loads in the sizing calculations. On the basis of this response the staff's concern is resolved.

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

The following new material supplements information in the SER.

In Section 8.3.2.2 of FSAR Amendment 63, the applicant stated that each essential load supplied power from the Class 1E dc distribution system boards has a redundant supply which is electrically separate from its first supply. The supply cables are routed so as to provide complete physical separation from the two supplies to each load. The staff was concerned that, when the redundant supply is used, the system alignment may be such that the design-basis requirements, which includes four independent power trains, will not be maintained. The staff told the applicant about this concern in Section 8.3.1.7.1 of the RAI dated June 20, 1991.

During the site review of August 7 and 8, 1991, the applicant stated that an alternate feeder analysis would precede the integrated tests of the safety systems. The analysis would identify the restrictions and limitations for each alternate feeder and the results would be incorporated into the Technical Specifications.

The applicant also stated that it would submit acceptance criteria for the future analysis that would be ready by January 1, 1993. In a letter dated September 13, 1991, the applicant stated that loads would only be transferred

between power sources within the same train in the opposite unit. This information was incorporated into the FSAR by Amendment 71.

In the letter dated September 13, 1991, the applicant also stated that analyses would be performed to indicate that the batteries' capacities would be sufficient with the alternate feeders energized during an accident in one unit and loss of battery chargers for 30 minutes. The applicant stated that station blackouts were also being analyzed and that limitations for the use of alternate feeders, such as load-shedding, would be included in the Technical Specifications.

The applicant submitted Calculation WBN EEB-MS-TI11-003, which analyzed the capacities of the 125-V dc vital batteries and chargers. An evaluation was included which demonstrated that adequate capacity was available while supplying alternate feeders for an accident scenario, but not for station blackout conditions. Therefore, the issue of adequate battery/charger capacity was considered closed for a LOCA with a loss-of-charge scenario, but would be reviewed separately for station blackout conditions. Subsequently, the applicant submitted Revision 35 of WBN EEB-MS-TI11-003, which demonstrated that the vital batteries have adequate capacity while supplying the alternate feeders also during station blackout conditions.

Verification of correct breaker alignments every 7 days has been included in the plant's draft Technical Specifications. An example (SOI-211.01, Revision 1) of a procedure to control the use of the alternate feeders was submitted for the staff's review. The review generated questions, and the staff transmitted these to the applicant in a letter dated March 28, 1994. The staff will review the applicant's response under TACs M89109 and M89110, and will report its evaluation in an SER supplement.

8.3.2.7 The Fifth Vital Battery System

The following new material supplements the information in the SER.

By FSAR Amendment 58, the applicant documented its proposed to add a fifth vital battery system to be used as a temporary replacement for any one of the four 125-V dc vital batteries during their testing, maintenance, and outages with no loss of system reliability under any mode of operation.

In FSAR Section 8.3.2.1.1, the applicant stated that the four channels of the vital dc power system are electrically and physically separated so that a single failure in one channel will not cause a failure in another channel.

When one of the four batteries of the electrically and physically separated vital dc power system described above has been temporarily replaced by the fifth battery, it was not clear, based on the information presented, that the four channels will remain electrically and physically separate so that a single failure in one channel will not cause a failure in another channel. In order to clarify the design's compliance with the single-failure criterion when the fifth battery is being used, the staff requested additional information in a letter dated June 20, 1991.

In the letter dated September 13, 1991, the applicant responded, referencing appropriate FSAR sections. Briefly, the fifth battery is located separately from the other batteries, with physical separation the same as that provided

for the other batteries. It has its own dedicated heating and ventilation system with redundant intake and exhaust fans with each set powered from separate auxiliary electrical trains. Also the applicant stated that electrical separation is provided by mechanically interlocked transfer switches with break-before-make contacts in addition to normally open breakers in each primary battery board. The limiting conditions of operation (when it is substituted for another vital battery) and surveillance of the fifth battery are encompassed by the Technical Specifications. On the basis of this information, these concerns are resolved.

8.3.2.8 Reenergizing the Battery Charger From the Onsite Power Sources Manually Versus Automatically Immediately Following a Loss of Offsite Power

The following new material supplements the information in the SER.

FSAR Section 8.3.1 implies that the battery chargers are Class 1E loads that are automatically reconnected to the onsite power supply following a loss of offsite power with or without an accident signal. However, FSAR Section 8.3.2 implied that the chargers may be reconnected manually 30 minutes following a loss of offsite power. The staff told the applicant about this concern in the RAI dated June 20, 1991.

During the site review of August 7 and 8, 1991, and subsequently in a letter dated September 13, 1991, the applicant stated that the chargers are not shed on a loss of offsite power and are reenergized when the EDGs are connected to the shutdown boards. In FSAR Amendment 70, the applicant removed the implication that the chargers needed to be reconnected. This concern is resolved.

8.3.3 Common Electrical Features and Requirements

8.3.3.1 Compliance With GDCs 2 and 4

8.3.3.1.1 Submerged Electrical Equipment as a Result of a Loss-of-Coolant Accident

In the SER and SSER 3, the staff stated that the design for the automatic deenergizing of loads as a result of a loss-of-coolant accident (LOCA) would be verified as part of the staff's site visit/drawing review. This activity was tracked by Confirmatory Issue 42. By letter dated January 16, 1985, the applicant submitted electrical schematic drawings for each valve identified to be automatically deenergized by a LOCA signal. On the basis of a review of drawing 45W760-30-8, Revision 10, the staff concluded that the design for power removal was acceptable pending resolution of surveillance requirements that were to be reviewed with and included in the Technical Specifications.

In the SER, the staff similarly stated that the design using power lockout or protective devices was acceptable pending resolution of surveillance requirements that were to be reviewed with and included in the Technical Specifications.

Subsequently, as part of a technical specifications improvement program, the staff concluded that such surveillance requirements were not appropriate for inclusion in technical specifications, but should be reviewed with and included in the FSAR as licensing commitments. In order to initiate its review

of the required surveillance, the staff asked the applicant for additional information in its RAI dated June 20, 1991.

During the site review of August 7 and 8, 1991, and in subsequently in a letter dated September 13, 1991, the applicant stated that it would revise the FSAR to state that the components listed in Table 8.3-28 are automatically deenergized by the accident signal, and that the accident signal must be reset to remove the automatic trip signal from each component. Testing to ensure the operability of the circuitry used for automatic deenergization is contained in technical surveillance requirements in the Technical Requirements Manual. Acceptance criteria for this test is that all components are automatically deenergized on a simulated accident signal. This test is performed every 18 months. For equipment automatically deenergized in the event of a LOCA, the described surveillance is acceptable.

In FSAR Amendment 71, the applicant incorporated this information. The components listed in FSAR Table 8.3-28 did not agree with those listed in the associated list in the Technical Requirements Manual because the FSAR was limited to only listing the major components to avoid excessive detail. This is acceptable, and fully resolves Confirmatory Issue 42.

During the site review of August 7 and 8, 1991, the applicant stated that it would provide an additional response for power lockout provisions and proper operation of protective devices. The applicant stated that power lockout provisions are not used to protect Class 1E buses due to submergence. This statement conflicted with the information in the Technical Requirements Manual pertaining to the power lockout circuitry for valves 1-FCV-74-1 and 1-FCV-74-9. Subsequent information submitted by the applicant indicates that these two valves do have power lockout provisions as do valves 1-FCV-74-2 and 1-FCV-74-8, which will also be included in the Technical Requirements Manual which allows a 7-day LCO (limiting condition for operation) for the lockout circuitry and provides monthly verification that the valves are deenergized. This is acceptable.

Further discussions with the applicant have indicated that protective device testing provisions are encompassed by technical surveillance requirements in the Technical Requirements Manual for isolation devices and containment conductor overcurrent protection devices. The staff reviewed those requirements and finds them acceptable. Therefore, the issue pertaining to testing of protective devices is resolved.

8.3.3.1.3 Failure Analysis of Circuits Associated With Cables and Cable Splices Unqualified for Submergence

The following new material supplements information in the SER and SSER 3.

Amendment 63 removed the following phrase from the FSAR: "Cables and cable splices are qualified for submergence."

With the removal of this phrase, the staff concludes that cables and splices used to supply Class 1E or non-Class 1E equipment that may be submerged following a LOCA are not qualified for submergence. The staff was concerned that the circuits associated with these unqualified cables and cable splices were not included in the failure analysis to demonstrate the ability of the electrical power system to withstand failure of submerged electrical components

from the postulated LOCA flood levels inside the containment. In order to resolve this concern, the staff asked the applicant to submit additional information clarifying that circuits associated with these unqualified cables and splices were included in the failure analysis; this request was made in the RAI dated June 20, 1991.

During the site review of August 7 and 8, 1991, and in subsequent discussions, the applicant indicated that cables and cable splices that are required to operate during and after a LOCA will not be routed below the flood level unless they are protected in some way, such as being located in conduit or a junction box. Also the electrical power system will be able to withstand the failure of submerged components (including cables) from the postulated LOCA flood levels.

A revision to the submergence calculation was scheduled to be performed upon the completion of the analysis of the auxiliary power system with the new common station service transformers (CSSTs) C and D. The calculation was revised to identify those cables that are routed below the flood level and receive power from a Class 1E board. The revision addressed the loading effect on the Class 1E boards due to the cables that are submerged and energized. If any of the cables are required to be energized for component operation, they will be rerouted above the flood level. For any of the Class 1E boards determined to be overloaded as a result of the additional loading from unqualified submerged cables, such corrective action will be initiated as tripping the load for an accident or changing the rating of the protective device.

Subsequently, the applicant submitted submergence calculation WBN-EEB-MS-TI08-009 for the staff's review. This calculation took into account the loading effects on Class 1E power supplies from submerged components. On the basis of the information in that calculation, this issue is resolved. The acceptability of submerged components will also be taken into account when the staff reviews the applicant's equipment qualification (10 CFR 50.49) program, which is tracked by Outstanding Issue 4(b).

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

The following new material supplements information in the SER.

Section 8.3.1.2.3 of FSAR Amendment 63 indicates that Class 1E cables routed underground between the auxiliary building, the diesel generator building, and the intake pumping station have waterproof splices in the potentially submersible sections of the duct runs. Position 9 of Regulatory Guide 1.75 states that cable splices in raceways should be prohibited. The basis for Position 9 further states that if cable splices exist, the resulting design should be justified by analysis and that the analyses should be submitted as part of the safety analysis report. In order to evaluate the use of cable splices in raceways, the staff asked the applicant for additional justifying analyses which will demonstrate compliance with the requirements of GDCs 2, 4, and 17; this request was made in the RAI dated June 20, 1991.

During the site review of August 7 and 8, 1991, and subsequently in a letter dated September 13, 1991, the applicant stated that splices are included in manholes of the underground duct run. The Watts Bar design basis does not permit splices to be installed in raceways. The applicant stated that the

FSAR would be changed to state that splices are not allowed to be installed in raceways.

Contrary to this, the applicant, in a response dated December 17, 1993, to issues raised in Inspection Report Nos. 50-390/93-74 and 50-391/93-74, dated December 20, 1993, described two methods of splicing cables in open cable trays allowed by Watts Bar Standard Drawing SD-E12.5.9. The staff's prohibition against splices in raceways is centered on the prevention of fires caused by improper splices. If splices are used in raceways that are part of the raceway system (not used for device terminations), then an analysis justifying their use should be made and documented in the FSAR as recommended by Revision 1 to RG 1.75. The staff told the applicant about this concern in a letter dated March 28, 1994. The staff will track resolution of this issue by TACs M89109 and M89110, and will convey its evaluation in an SER supplement.

8.3.3.2 Compliance With GDC 5

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

In the SER the staff stated that the dc system for Unit 1 supplies power to vital buses I and II for Unit 2, and the dc system for Unit 2 supplies power to vital buses III and IV for Unit 1. This was wrong; the Unit 1 dc system supplies Unit 1 buses I and II and the Unit 2 dc system supplies Unit 2 buses III and IV. Because the dc distribution system design provides for normal and alternate supplies to system boards and some components, some sharing of the dc system can occur in certain modes of operation. See Section 8.3.2.5.1 herein for a discussion of the acceptability of this design feature.

Also in the SER, the staff concluded (based on results of an analysis submitted in a letter dated January 7, 1982, imposition of appropriate Technical Specifications for shared dc systems, and other information documented in the FSAR) that there was reasonable assurance that the sharing would not significantly impair the ability of the dc system to perform its safety function, meets the requirements of GDC 5, and was acceptable. However, after further review of the results of the analysis submitted in the letter dated January 7, 1982, and information documented in the FSAR, it was not clear that the sharing of raceways would not significantly impair the ability of the dc system from performing its safety function for any given single failure in accordance with the requirements of GDCs 5 and 17. The staff was concerned that a single failure of the shared train A (or train B) raceway may cause failure of two dc vital battery systems. The staff told the applicant about these two concerns in the RAI dated June 20, 1991. The staff's concerns were resolved as follows:

- (1) During the site review of August 7 and 8, 1991, and subsequently in a letter dated September 13, 1991, the applicant stated that the design is capable of the safe shutdown in one unit and the mitigation of an accident in the other unit for loss of battery combinations I and III or II and IV.

The applicant further indicated that failure of the A train power to the battery chargers can cause loss of battery systems I and III, and that failure of the B train power to the battery chargers can cause loss of battery systems II and IV. The complete failure of the two dc channels

would not occur until 30 minutes into an accident. Also the complete loss of two dc channels would lead to a reactor trip and the plant entering into a reactor trip recovery procedure. Also cables with channel/train designations are allowed only in raceways having the same channel/train designation. On the basis of this information, this concern is resolved.

- (2) During the site review of August 7 and 8, 1991, and subsequently in a letter dated September 13, 1991, the applicant stated that it would revise the FSAR to document the results of the vital 125-V dc system with respect to compliance with Position C.2 of Regulatory Guide (RG) 1.81. Operations personnel indicated that some sharing, at least at the system level, may be required. The applicant stated that it would reconsider its response for compliance to RG 1.81. Information documenting the applicant's position was subsequently submitted in FSAR Amendment 71. Therefore, this concern is resolved.

8.3.3.2.2 Sharing of AC Distribution Systems and Standby Power Supplies Between Units 1 and 2

In the SER, the staff stated that (1) sharing of onsite ac and dc systems had not been adequately described or analyzed in FSAR Section 8.3; (2) the applicant, in a letter dated January 7, 1982, had submitted the required description and analyses; and (3) the design met, on the basis of the information presented in the letter dated January 7, 1982, the guidelines of RG 1.81, and was acceptable pending revision of the FSAR that reflects requirements of the shared safety systems.

In SSER 3, the staff stated that information presented in Amendment 48 and in a letter dated January 17, 1984, was consistent with information presented in the applicant's letter dated January 7, 1982, and was acceptable pending confirmation that the information in the January 17, 1984, letter was also incorporated into the FSAR. The staff stated that Confirmatory Issue 32 remained open to track this effort.

The staff reviewed information submitted in the FSAR through Amendment 63 with respect to the letters dated January 7, 1982, and January 17, 1984. As a result of this review, the staff noted a number of discrepancies. The staff told the applicant of its concerns in the RAI dated June 20, 1991.

During the site review of August 7 and 8, 1991, and subsequently in a letter dated September 13, 1991, the applicant submitted a written response which stated that for a loss of offsite power, a design-basis accident in one unit and a full load rejection in the other unit, one division of engineered safety features (ESF) equipment was sufficient to safely shut down both units, even with a complete failure of the other train in both units. Procedures ensure that test and maintenance activities will not be performed if the redundant counterpart equipment is inoperable. Also, each unit has a bypass and inoperable status panel which receives inputs from each shared system. On the basis of the applicant's response, the staff's concerns and Confirmatory Issue 32 are resolved.

8.3.3.2.4 Possible Sharing of DC Control Power to AC Switchgear

See Section 8.3.2.5.1 for information that supplements the discussions in the SER and SSER 3.

8.3.3.3 Physical Independence (Compliance With GDC 17)

(3) Separation Criteria Between Class 1E and Non-Class 1E Circuits

In the SER, the staff stated that surveillance requirements for protective devices will be reviewed with the Technical Specifications. Since issuing the SER, the staff, as part of its technical specifications improvement program, has established that surveillance requirements for the subject protective devices that are to be used to ensure independence of circuits at Watts Bar should be included in, and should be evaluated as part of, the FSAR. In the RAI dated June 20, 1991, the staff asked the applicant to submit a description of the surveillance requirements for protective devices.

During the site review of August 7 and 8, 1991, and subsequently in a letter dated September 13, 1991, the applicant stated that at least 10 percent of the electrically operated circuit breakers and molded-case circuit breakers actuated by a fault current and installed as isolation devices will be tested every 18 months and 100 percent of the breakers will have the recommended maintenance performed within a 60-month period. For any breaker found inoperable, an additional 10 percent of that type of breaker will be tested until no more failures are found or until all electrically operated circuit breakers of that type have been functionally tested. In FSAR Amendment 71, the applicant incorporated this information. Therefore, this issue is resolved.

Also during the site review of August 7 and 8, 1991, the applicant stated that it would describe the surveillance requirements for the testing of protective devices used to protect Class 1E circuits from failure of non-Class 1E circuits. Testing includes a 300-percent current test at initial installation on the thermal breakers and 15x rating or more for magnetic breakers at initial installation. The dc breakers are not tested, since the manufacturer tests them. After installation, the breakers are tested to NEMA (National Electrical Manufacturers Association) Standards (Publication/AB2-1990, "Procedures for Field Inspection and Performance Verification of Molded Case Circuit Breakers Used in Commercial and Industrial Applications,") which only tests opening and closing of the breaker. During the site review, the staff stated that this level of testing was not considered sufficient to ensure the operability of the protective devices. In the letter dated September 13, 1991, the applicant stated that subsequently it would submit material regarding testing. In the proposed Technical Requirements Manual (submitted as part of the Technical Specifications review effort, TAC M76742), the applicant has included surveillance requirements (functional testing) for circuit breakers which are used as isolation devices. On the basis of the review and acceptance of that document, this issue is resolved.

(5) Separation Between Open Cable Trays and Conduits

The following new material supplements information in the SER.

In the SER, the staff stated that separation between conduits and open top cable trays was not described in the FSAR or in additional information submitted by the applicant. Currently, FSAR Section 8.3.1.4.2 states that there is no established minimum separation between open top non-Class 1E cable trays and conduits containing redundant cables, and that credit is taken for fire-resistant cable coating installed before October 18, 1984, together with adequate circuit protective device(s) as meeting the intent of RG 1.75. Coating is not used after October 18, 1984, on cables that conform to meet IEEE Standard 383-1974, "IEEE Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations." Further, there is no discussion in the current FSAR of separation distances between Class 1E open cable trays and conduits.

In NRC Inspection Report Nos. 50-390/93-74 and 50-391/93-74, dated December 20, 1993, the staff raised concerns related to the minimum separation distance between divisional open cable trays and conduits as described in Watts Bar General Design Criteria WB-DC-30-4, "Separation/Isolation." Appendix C of that document gives the rationale (unsupported by analysis/test) for distances down to 1 inch when the cable tray is not covered. The appendix states that the metal conduit (twice as thick as a cable tray cover) is a sufficient heat shield/sink to protect cables in the conduit against the physical energy associated with a fault in an open cable tray located as close as 1 inch to the conduit. The appendix further states that the conduit thickness and the lack of sufficient oxygen needed to support combustion inside the conduit ensure that damage to cables in cable trays as close as 1 inch is unlikely if a fault should occur inside the conduit. Credit is also taken in the appendix for the fire detection/suppression systems to minimize the propagation of a fire, for the use of fire-retardant material in specific cases, for certain cable passing vertical flame tests, and for the protection provided by primary breakers.

Staff guidance from RG 1.75 states that if the minimum separation distance (much greater than an inch) cannot be maintained, the redundant circuits should be run in solid enclosed raceways (enclosed cable trays, conduits, etc.) that qualify as barriers, or that other barriers should be provided with a minimum separation of 1 inch between the enclosed raceways and between the barriers and raceways.

A comparison between RG 1.75 and WB-DC-30-4 revealed several differences, such as the use of a cable tray cover allowed by WB-DC-30-4 versus a completely enclosed tray recommended by RG 1.75. Also the use of a barrier without an additional 1-inch air gap is allowed by WB-DC-30-4. As noted above, the appendix to WB-DC-30-4 allows exceptions (such as no tray cover required between a cable tray and a conduit for separations down to an inch) to the separation requirements based on a case-by-case analysis without supporting test results; that also deviates from RG 1.75.

In a response dated December 17, 1993, to the inspection report, the applicant referred to IEEE Standard 384-1992, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits," as providing guidance for separation distances between open cable trays and conduits. Although the staff has not formally endorsed this revision to the IEEE standard, the staff's review indicates that it does provide guidance (with limiting assumptions) for minimum separation distances based on actual, credible test results. Unfortunately,

as noted in the applicant's letter, WB-DC-30-4 allows separation distances less than that supported by the IEEE standard.

Because of the differences noted between WB-DC-30-4 and RG 1.75 pertaining to the separation between open cable tray and conduits, the staff will review the applicant's case-by-case justification (supported by analysis/test) for deviations from RG 1.75 with further current industry guidance contained in IEEE Standard 384-1992 and its supporting documentation. The staff told the applicant about this concern in a letter dated March 28, 1994. The staff will track resolution of this issue by TACs M89109 and M 89110, and will convey its evaluation in an SER supplement.

8.3.3.5 Compliance With GDC 18

8.3.3.5.1 Compliance With Regulatory Guides 1.108 and 1.118

In the SER, the staff stated that the applicant, by letter dated October 16, 1981, had stated that the FSAR will be revised to indicate full compliance with Regulatory Guide 1.118 ("Periodic Testing of Electric Power and Protection Systems"). The staff found this commitment for full compliance acceptable. Subsequently in FSAR Amendment 63, the applicant stated that the Watts Bar design complies with all of the positions of Regulatory Guide 1.118, Revision 2, except for Position C.6(a).

Position C.6(a) allows (as an exception to Section 6.4(5) of IEEE Standard 338-1977) the use of temporary jumper wires when portable test equipment is used during testing. For the temporary jumper wires to be acceptable, the position requires that the safety system equipment undergoing test be provided with facilities specifically designed for connection of this portable test equipment and that these facilities must be considered part of the safety system undergoing test and must meet all the requirements of IEEE Standard 338-1977.

In justification of this exception, the applicant documented in the FSAR that, where feasible, test switches or other necessary equipment will be installed permanently to minimize the use of temporary jumpers in testing. By RAI dated June 20, 1991, the staff sought additional information to complete this review. The staff's review of the additional information follows:

- The staff asked the applicant to identify each safety system component where temporary jumpers are used that do not meet Position C.6(a) of RG 1.118. During the site review of August 7 and 8, 1991, the applicant stated that it will identify, by procedure, each safety system component where temporary jumpers are utilized in testing. This resolves the staff's concern.
- The staff asked the applicant to demonstrate that the use of each jumper identified above will not compromise the design-basis safety function of the system component being tested. During the site review of August 7 and 8, 1991, the applicant stated that it will (through the 10 CFR 50.59 process) analyze each identified jumper to ensure that it will not compromise the design basis of the system component being tested. This resolves the staff's concern.

The staff asked the applicant to make a design commitment which indicates that test switches or other necessary permanent equipment installed to minimize the use of temporary jumpers in testing will meet all the requirements of IEEE Standard 338-1977. During the site review of August 7 and 8, 1991, the applicant stated that it would revise the FSAR as proposed in a draft markup to indicate compliance with IEEE Standard 338-1977. The applicant subsequently incorporated the proposed draft in FSAR Amendment 71. Therefore, the staff's concern is resolved.

In the FSAR, the applicant identified an exception to Position C.2.a(2) of Regulatory Guide 1.108. Position C.2.a(2) requires that the emergency loads be sequenced onto the diesel generator unit with each load operating at its full load rating (i.e., each pump operating at full flow). The applicant stated that as part of preoperational testing, loads would be operated at full flow while being sequenced on the diesel generator, but that, during 18-month periodic testing, loads would be operated at miniflow while being sequenced on the diesel generator. Although miniflow produces a loading of less than full load, the staff stated in the SER that the use of miniflow loads during periodic testing was acceptable.

Because testing at full load is not practical, the staff continues to find the use of miniflow loads during periodic testing acceptable; however, the staff is concerned that this acceptability for the use of miniflow loads may be misinterpreted to mean that a successful test at miniflow translates, without further analysis or evaluation, to a successful demonstration of the diesel generator's capability to sequence loads at full flow. The staff told the applicant about this concern in the RAI dated June 20, 1991.

During the site review of August 7 and 8, 1991, the applicant stated that it would submit a written response with regard to modeling, trending, analysis, and/or other methodology performed on the miniflow test data to periodically demonstrate the capability of the diesel generator to sequence the full-flow design-basis loads. In a letter dated September 13, 1991, the applicant stated that a baseline for each EDG's dynamic response during sequential loading would be developed during preoperational testing. A trending program will be established to compare the 18-month periodic pump miniflow test data to the data used for the pumps in the baseline. Adverse variations in EDG loading response noted during the comparisons or taken from other trending analyses will be analyzed with justification for the acceptability of the adverse response provided or corrective actions to restore the EDG to acceptable status implemented. The applicant stated in discussions that future consideration would be given to the following:

- (1) A computer model of the EDG dynamic loading response should be developed and validated against preoperational data.
- (2) This model should be used as a reference for future modifications and tests.
- (3) This model should account for the time difference in miniflow versus full-flow pump response.

On the basis of this information, this issue is resolved.

In a letter dated February 7, 1994, the applicant stated that the FSAR would be revised to describe compliance with RG 1.9, Revision 3, "Selection, Design, Qualification, Testing, and Reliability of Diesel Generator Units Used As On-site Electric Power Systems at Nuclear Power Plants." As part of the FSAR revision, the applicant is deleting compliance statements for RG 1.108 which has been withdrawn by the staff. Since this, in effect, negates the staff's previous review conclusions pertaining to RG 1.108 and its guidance for diesel generator testing, the applicant's new compliance with RG 1.9, Revision 3 (and an exception thereto), is considered an open item. The staff has told the applicant about this concern in a letter dated March 28, 1994. The staff will track resolution of this issue by TACs M89109 and M89110, and will convey its evaluation in an SER supplement.

8.3.3.5.2 Testing of One of Two Class 1E Power Systems Versus One of Four Systems

In the SER, the staff stated that the applicant, by letter dated October 9, 1981, documented the commitment that only one of the four power trains of the plant (Unit 1 and 2) would be tested at one time. The staff concluded that this commitment was acceptable pending its documentation in the FSAR. In SSER 2, the staff stated that the applicant had supplied the required documentation by FSAR Amendment 48. The staff, in its review of FSAR Amendment 63 could not locate the specific commitment which states that only one train would be tested at one time. The staff told the applicant about this concern in the RAI dated June 20, 1991.

During the site review of August 7 and 8, 1991, the applicant stated that the FSAR commitment to test only one train at a time appears on pages 8.3-9 and 8.3-12 of the FSAR. The operations engineers at the site review pointed out that the applicant's procedures state that only one train (two EDGs - one per unit) should be tested at one time and that this is inconsistent with the FSAR statement that only one diesel will ever be in the test mode at any one time. In Amendment 71, the applicant revised the FSAR to state that only one EDG will be in the test mode at any one time, unless both units are in cold shut-down. This is acceptable and this issue is resolved.

8.3.3.6 Compliance With GDC 50

In FSAR Section 8.1.5.3, the applicant stated, as an exception to the testability requirements of RG 1.63, that, in lieu of the testing of fuses by resistance measurement, a fuse inspection and maintenance program would be established. Because justification was not presented in the FSAR for this exception, the staff stated in the SER that periodic resistance measurement of fuses and their terminal connections would be required as part of the Technical Specifications.

By letter dated September 15, 1982, and FSAR Amendment 55, the applicant submitted information as to why periodic resistance measurement is not practical and provided justification for the adequacy of the proposed inspection and maintenance program in lieu of resistance measurement testing. However, the applicant, by FSAR Amendment 63, removed this justification. Therefore, on the basis of information then presented in the FSAR, the staff was unable to find the applicant's proposed inspection and maintenance program acceptable.

In regard to the statement in the SER that periodic resistance measurement of fuses and their terminal connections would be required as part of the Technical Specifications, the staff has subsequently concluded as part of the standard technical specifications improvement program that this type of testing requirement should not appear in the plant's Technical Specifications but should be included instead in the FSAR as a design-basis requirement for the plant. Periodic resistance measurement of fuses and their terminal connections was, therefore, not required as part of the Technical Specifications as indicated in the SER, but was addressed as part of the staff's review of the FSAR. The staff told the applicant of its concerns pertaining to the adequacy of the proposed inspection and maintenance program in the RAI dated June 20, 1991.

During the site review of August 7 and 8, 1991, and subsequently in a letter dated September 13, 1991, the applicant stated that it would put back the following justification that had been removed:

Fuse manufacturers have also stated that fuses do not deteriorate with service life. Service temperatures above the rated temperature, current surges, and unusual cycling conditions all reduce the fuses service life, i.e., the fuse becomes more protective. Under no conditions will a fuse become less protective during its service life.

The subsequent inclusion of this statement in the FSAR by Amendment 70 resolves this concern.

Failure of the fuse because of aging was discussed. The applicant stated that fuses would not age at the same rate and that their failure would not cause common-mode failure of the redundant circuits. Fuses will trip randomly and will be identified by periodic monitoring of equipment or by loss of power due to their failure. This was not considered to be an issue.

9 AUXILIARY SYSTEMS

9.1 Fuel Storage Facility

9.1.4 Fuel Handling System

In Section 9.1.4 of SSER 3, the staff stated that it was reviewing the applicant's response to NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." The applicant replied by letters dated February 6 and March 20, 1984. To ensure completion of Phase I of NUREG-0612 guidelines in a timely manner (pending completion of the staff's review of the applicant's response), the staff stated that the following license condition (proposed License Condition No. 39) will be placed in the operating license:

The applicant will meet the guidelines of Sections 5.1.1 and 5.1.3 of NUREG-0612 (Phase I) before the first refueling outage.

By letter dated July 28, 1993, the applicant revised its response to the guidelines of NUREG-0612. This latest submittal superseded the applicant's responses of February 6 and March 20, 1984. The revised response reflects the substantial redesign and procurement of lifting devices that has occurred during the time since the 1984 letters.

Phase I of NUREG-0612 covers the following areas:

- (1) definition of safe load paths
- (2) development of load-handling procedures
- (3) periodic inspection and testing of cranes
- (4) qualifications, training, and specified conduct of operators
- (5) special lifting devices that should satisfy the guidelines of ANSI N14.6.6 (American National Standards Institute standard)
- (6) lifting devices not specially designed that should be installed and used in accordance with the guidelines of ANSI B30.9
- (7) design of cranes to ANSI B30.2 or CMAA-80 (Crane Manufacturers Association of America standard)

Phase II addresses the need for electrical interlocks/mechanical stops or, alternatively, single-failure-proof cranes or load-drop analyses in the spent fuel area, the containment, and in other areas containing safety-related equipment. The staff has reviewed the applicant's response and concludes that the applicant has satisfied the guidelines of Phase I. The staff, therefore, concludes that proposed License Condition No. 39 is no longer necessary.

In Generic Letter 85-11 (Completion of Phase II of "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612) dated June 28, 1985, the staff stated that satisfaction of the Phase I guidelines ensures that the potential for a heavy

load drop is extremely small. The staff, therefore, concluded (see Comanche Peak SSER 12, NUREG-0797) that the guidelines of Phase I adequately provide the intended level of protection against load-drop accidents and that a detailed review of the Phase II response was not necessary. However, the staff has performed a cursory review of the applicant's response to Phase II of NUREG-0612 and concludes that the Phase II guidelines have been adequately addressed.

As stated above, proposed License Condition No. 39 is deleted. The staff concludes that the heavy-load-handling systems conform to the guidelines of NUREG-0612, and are, therefore, acceptable.

13 CONDUCT OF OPERATIONS

13.3 Emergency Preparedness

13.3.1 Introduction

The applicant originally submitted the Watts Bar Nuclear Plant Radiological Emergency Plan (REP) on January 28, 1982. The staff's review of that plan is discussed in the SER (June 1982). Subsequently, the applicant halted licensing activities at Watts Bar and withdrew the Watts Bar REP. On February 12, 1993, the applicant resubmitted the Watts Bar Nuclear Plant Radiological Emergency Plan, Revision 0, and Implementing Procedures as Appendix C to the TVA Radiological Emergency Plan, Revision 15. Because more than a year had elapsed between submittals, the staff performed a complete review of the REP.

The staff reviewed REP and subsequent revisions (1 through 5) against the requirements of 10 CFR 50.47(b) and (d) and Appendix E to 10 CFR Part 50. In conducting the review, the staff used the guidance criteria in NUREG-0654/FEMA-REP-1, Revision 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," dated November 1980, along with NUMARC/NESP-007, "Methodology for Development of Emergency Action Levels" (Revision 2, January 1992), and Supplement 1 to NUREG-0737, "Clarification of TMI Action Plan Requirements - Requirements for Emergency Response Capability." NUREG-0654/FEMA-REP-1 has been endorsed by Regulatory Guide 1.101, Revision 2, "Emergency Planning and Preparedness for Nuclear Power Reactors," dated October 1981. NUMARC/NESP-007 has been endorsed by Regulatory Guide 1.101, Revision 3, dated August 1992. This review was done in accordance with NUREG-0800, Revision 2, July 1981, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 13.3, "Emergency Planning."

This safety evaluation adheres to the format of Part II of NUREG-0654 in that each of the planning standards is listed and is followed by a summary of the applicable portions of the plan that relate to that specific standard. It supersedes the safety evaluation in Section 13.3 of the SER.

13.3.2 The Emergency Plan

13.3.2.1 Assignment of Responsibility (Organizational Control)

The planning standard requires that primary responsibilities for emergency response by the licensee of the nuclear facility, and by State and local organizations within the emergency planning zones have been assigned, the emergency responsibilities of the various supporting organizations have been specifically established, and each principal response organization has staff to respond and to augment its initial response on a continuous basis.

The shift operations supervisor (SOS) on duty or an assistant shift operations supervisor acting as the SOS at Watts Bar is designated as the site emergency director (SED) until relieved by the plant manager or his/her alternate. Upon detection of a known or suspected emergency, the SOS determines the classification of the emergency, assumes the duties of the SED, and implements the

plan. The responsibilities of major organizations for supporting the Watts Bar plant are summarized in Figure 2-1 (Revision 11) of the TVA REP. There is a 24-hour-a-day communication capability between the station and the Federal, State and local response organizations to ensure rapid transmittal of accurate notification information and assessment data.

Responsibility for overall performance of the site emergency response organization is vested in the SED, who is responsible for the overall direction of the plant emergency organization and for initiating long-term 24-hour-per-day accident mitigation operations. Qualified members of the station staff, who report directly to the station, have been assigned specific responsibilities for the major elements of the emergency response.

The applicant has established a Central Emergency Command Center (CECC) in Chattanooga, Tennessee. The NRC approved TVA's centralized emergency management concept on March 19, 1981. Emergency facilities and equipment are discussed in Section 13.3.2.8 of this safety evaluation. The stated purpose of the CECC and its staff is to provide the facilities and manpower for evaluating, coordinating, and directing the overall activities involved in coping with a radiological emergency at any TVA facility. The CECC is designed to serve as the central point for information collection, assessment, and transfer during an emergency. The CECC has direct communication links with the State emergency response centers, other TVA emergency response organizations, the plant sites, the Joint Information Center, and offsite Federal and State organizations. Within the CECC, provisions have been made for the NRC response team.

The CECC Director has overall responsibility and authority for ensuring adequate TVA response to affected State and local governments in protecting the health and safety of the public. The State of Tennessee Multi-Jurisdictional Radiological Emergency Response Plan (RERP) for Watts Bar, as well as the plans for those portions of States within the 50-mile ingestion pathway, are referenced in Appendix E to the TVA REP. These plans provide for the coordinated response of the State of Tennessee and local governments (McMinn, Meigs, and Rhea counties for Watts Bar) within the 10-mile plume exposure emergency planning zone (EPZ) as well as the States and local governments within the 50-mile ingestion pathway.

Arrangements have been made for requesting and effectively using offsite assistance at Watts Bar. Updated written agreements with appropriate agencies and organizations are maintained annually by either the Watts Bar Nuclear Security Services for offsite law enforcement support or by Watts Bar emergency preparedness planning or TVA corporate emergency planning for all other support. Offsite supports includes police departments, ambulance services, fire-fighting support, and hospitals near Watts Bar, as well as the Institute for Nuclear Power Operations and the Department Of Energy Radiation Assistance Center/Training Site in Oak Ridge, Tennessee. Section 16.5 (Revision 7) of the TVA REP lists agreements or contracts maintained for services of outside organizations during an emergency.

13.3.2.2 Onsite Emergency Organization

The planning standard requires that (1) the responsibilities of the onshift plant staff for emergency response are clearly defined, (2) adequate staffing to provide initial facility accident response in key functional areas is

maintained at all times, (3) timely augmentation of response capabilities is available, and (4) the working relationships among various onsite response activities and offsite support and response activities are specified.

The onsite emergency organization of plant staff personnel and the responsibilities and duties of the normal staff complement are specified in Figure 3-1, page 12 (Revision 10), of the TVA REP and Figures 1-C and 2-C, pages C-202 and C-203 (Revision 0), of the Watts Bar REP.

The shift operations supervisor assumes the responsibilities of the SED until relieved by the plant manager or a designated alternate. The authorities and responsibilities of the SED have been clearly specified, including those authorities and responsibilities that cannot be delegated. Once the CECC is staffed, the decision to make protective action recommendations to authorities responsible for offsite emergency measures is transferred to the CECC Director. The SED can immediately and unilaterally declare an emergency and make offsite notifications.

Station staff emergency assignments have been made and the relationship between the emergency organization and normal staff complement is shown in the Watts Bar REP. Positions, titles, and qualifications of shift and plant staff personnel both on and off the site who are assigned major emergency functional duties, are listed. Figures 1-C and 2-C of the Watts Bar REP show minimum onshift staffing levels during emergencies and augmentation of shift staffing that conform to the guidance of Table B-1 of NUREG-0654 and Supplement 1 of NUREG-0737.

TVA management personnel will supply support services utilizing the manpower and equipment resources of the TVA organization. Watts Bar Emergency Plan Implementing Procedure 13 (EPIP-13) (Revision 2), "Termination of the Emergency and Recovery," provides a long-term emergency organization framework, which is to be headed by the Director, CECC. Working relationships between and among the TVA staff, station staff, governmental and private sector organizations, and technical and/or engineering contractor groups have been specified along with services to be provided. Other organizations capable of augmenting the planned onsite response have also been identified.

13.3.2.3 Emergency Response Support and Resources

The planning standard requires that arrangements for requesting and effectively using assistance resources have been made, arrangements to accommodate State and local staff at the licensee's near-site emergency operations facility have been made, and other organizations capable of augmenting the planned response have been identified.

The CECC Director is the TVA person authorized to request Federal assistance. TVA has developed agreements with the Department of Energy (Oak Ridge Operations Office and Savannah River Office) to provide experienced specialists and/or specialized equipment to help during radiological emergencies. The CECC Director implements these agreements as necessary during the course of an emergency. TVA sends a CECC representative to the State Emergency Operations Center (SEOC) in Nashville, Tennessee, to interpret technical aspects of the emergency condition and keep the CECC informed about State problems, requests, and actions. Arrangements are in place to accommodate State and local staff at the CECC.

Procedures have been established for communications with State and local response organizations, including initial notification of the declaration of the emergency. When the emergency plan is activated, the SOS on duty in the reactor control room notifies the operation duty specialist (ODS) in Chattanooga, who notifies Tennessee State authorities, the TVA Information Office, and the Director of the CECC, and in case of a General Emergency, also makes a direct parallel notification to the local county emergency operation centers. Provisions have been made and procedures are in place to periodically send plant and response status reports to the State.

Provisions have been made at the CECC and at the Watts Bar Technical Support Center (TSC) for the NRC response team. Additionally, a local recovery center (LRC) has been predesignated at Watts Bar for use by offsite TVA and NRC personnel. The NRC may use the LRC during the event as an area near the site for assessment and assistance; it has the capability to communicate with the TSC and offsite locations. It is located so that personnel have access to necessary drawings and documents. Meteorological information is also available in the LRC. Emergency facilities and supplies are discussed in Section 13.3.2.8, which gives additional information concerning the LRC.

Each offsite agency that potentially will be called upon to participate in the plan has concurred with the responsibilities assigned its agency in the plan by executing a Letter of Agreement with TVA. The applicant maintains an agreement with the Institute of Nuclear Power Operations (INPO), a consortium of nuclear utilities and other nuclear industries, to obtain any necessary support available from the industry during an emergency.

The applicant has made provisions to support the NRC portion of the Federal response and the State of Tennessee has made provisions in its RERP for the support of other Federal responders. The Federal Emergency Management Agency (FEMA) will review the State of Tennessee RERP and make findings and determinations concerning the adequacy of that plan to the NRC before NRC authorizes operation above 5 percent of rated power. This item will be tracked by TAC M89154 and will be addressed in a supplement to the SER.

13.3.2.4 Emergency Classification System

The planning standard requires that a standard emergency classification and action level scheme, the bases of which includes facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by facility licensees for making determinations about minimum initial offsite response measures.

The Watts Bar REP, submitted on January 28, 1982, contained an emergency action level (EAL) scheme based on Appendix 1 of NUREG-0654/FEMA-REP-1. By letter dated February 12, 1993, as supplemented by letters dated August 16 and September 22, 1993, TVA proposed changes to the emergency plan for Watts Bar. Specifically, Revision 4 to the Watts Bar REP incorporated revised EALs based on NUMARC/NESP-007, Revision 2, January 1992, "Methodology for Development of Emergency Action Levels." In Revision 3 to Regulatory Guide 1.101, "Emergency Planning and Preparedness for Nuclear Power Reactors," the staff endorsed NUMARC/NESP-007, Revision 2, as an acceptable method for licensees to meet the requirements of 10 CFR 50.47(b)(4) and Appendix E to 10 CFR Part 50. The staff relied upon the guidance in NUMARC/NESP-007 as the basis for its review of EAL changes at Watts Bar.

The staff reviewed the EAL changes in Revision 4 to the Watts Bar REP against the requirements in 10 CFR 50.47(b)(4) and Appendix E to 10 CFR Part 50. In accordance with Section IV.B. of Appendix E, the applicant, the State of Tennessee, and the local county government authorities discussed the proposed EALs and agreed upon them. In a letter to TVA dated February 19, 1993, the Tennessee Emergency Management Agency stated that the State of Tennessee concurs with the EALs proposed by NUMARC for the Watts Bar plant.

In 10 CFR 50.47(b)(4), the NRC specifies that onsite emergency plans must meet the following standard: "A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee...." Watts Bar EALs are used for recognizing and declaring the emergency class and are based upon specific measurable values or observable conditions. The specific instrument readings and parameters used as EALs for Watts Bar are detailed within the in-plant procedures.

In 10 CFR Part 50, Appendix E, Section IV.C, the Commission specifies that "emergency action levels (based not only on onsite and offsite radiation monitoring information but also on readings from a number of sensors that indicate a potential emergency, such as pressure in containment and the response of the emergency core cooling system) for notification of offsite agencies shall be described....The emergency classes defined shall include: (1) notification of unusual event, (2) alert, (3) site area emergency, and (4) general emergency." At Watts Bar, the licensee uses the standard emergency classifications of (1) Notification of Unusual Event, (2) Alert, (3) Site Area Emergency, and (4) General Emergency.

The applicant formatted its EALs into seven separate tables titled as follows:

- (1) Fission Product Barrier Matrix
- (2) System Degradation
- (3) Loss of Power
- (4) Hazards and SED Judgement
- (5) Destructive Phenomena
- (6) Shutdown System Degradation
- (7) Radiological

Eighty-nine (89) discrete EALs that correspond to the example EALs in NUMARC/NESP-007 are categorized in a top-down fashion within these tables. The applicant has submitted adequate technical justification for any departures from or omissions to the example EALs in NUMARC/NESP-007.

On the basis of its review of the proposed EAL changes in Revision 4 to the Watts Bar emergency plan and the supplementary information submitted in letters dated August 16 and September 22, 1993, the staff concludes that the revised EALs are consistent with the guidance in NUMARC/NESP-007 and, therefore, conform to the requirements of 10 CFR 50.47(b)(4) and Appendix E to 10 CFR Part 50.

13.3.2.5 Notification Methods and Procedures

The planning standard requires that procedures be established for the licensee to notify State and local response organizations about an emergency and for all response organizations to notify emergency personnel. The planning

standard also requires that the content of the initial and followup messages to response organizations and to the public be established, and that means to provide early notification and clear instructions to the populace within the plume EPZ be established.

Procedures have been established for notifying State and local response organizations in case of an emergency. The SED has been given the authority and responsibility to initiate prompt notification to these agencies. The SED at Watts Bar notifies the ODS, a permanently manned 24-hour-a-day position at the TVA Chattanooga Office Complex (COC), via a dedicated ringdown system or commercial phones. After being notified of an emergency by the SED as shown in Figure 5-1 (Revision 7) of the TVA REP, the ODS is responsible for making initial notification and sending information to the State of Tennessee within 15 minutes of the declaration of the event. For the initial classification at the General Emergency level (when the CECC is not operational), the ODS notifies both the State of Tennessee and the local government agencies (McMinn, Meigs, and Rhea counties for Watts Bar) within 15 minutes and passes along protective action recommendations developed by the SED. In addition, the ODS initiates notification of the appropriate TVA offsite emergency personnel by means of the TVA emergency paging system. The SED also has the capability to notify the State of Tennessee directly.

Beginning at the Alert classification level, the State of Tennessee is advised hourly or more often, as necessary, about appropriate plant status and environmental conditions.

The prompt notification system at Watts Bar consists of fixed sirens and tone-alert radios. The system is designed to provide an alert signal and information message via the Emergency Broadcast System (EBS) within 15 minutes of the decision made by government authorities to notify the population within 10 miles of the plant. The fixed-siren component consists of 99 electromechanical sirens. The sirens are activated by the Tennessee Emergency Management Agency (TEMA). A backup activation system is located in Rhea County. The siren system is activated on a monthly basis by TEMA as a regularly scheduled test. A silent test is conducted every two weeks to test the radio link to the sirens. A "growl" test is performed by TVA on a quarterly basis. This test consists of each siren being activated individually by a portable test unit.

The tone-alert radio component consists of radios which operate on county frequencies and are activated by McMinn, Meigs, and Rhea counties. The radios are located in the schools in the Watts Bar EPZ. Preventive maintenance for the sirens and tone-alert radios is performed by TVA on an annual basis commensurate with the manufacturer's recommendations. Unscheduled maintenance is performed as needed.

TVA submitted the "Evaluation and Analysis of the Alert and Notification System, FEMA REP-10 Report," for Watts Bar to FEMA for review on April 26, 1993. In addition to this review, FEMA will evaluate the adequacy of the system during actual system tests, and the system will be upgraded, if necessary, in response to any identified deficiencies. This item will be tracked by TAC M89154 and will be addressed in a supplement to this SER before NRC authorizes operation above 5 percent of rated power.

13.3.2.6 Emergency Communications

The planning standard requires that provisions exist for prompt communications among principal response organizations to emergency personnel and to the public.

The station communications systems are designed to provide secure, redundant and diverse communications to all essential onsite and offsite locations during normal operations and under accident conditions. Within-station systems include a public address system, two-way radios, a private automatic exchange, and a sound-powered telephone system. Offsite systems include both commercial and leased telephone lines, a microwave system, and a two-way-radio system. A Bell Telephone ring-down system is designated as the primary means of communication between plant and offsite emergency control centers.

These telephones, plus other systems located within the plant, operate 24 hours a day. In emergency situations, the SED will communicate directly with the TVA ODS who is responsible for providing initial notification to the appropriate State emergency organizations. In the event of a General Emergency, the ODS is required to also notify the appropriate local response agencies. The offsite warning points for the State and local governments are staffed 24 hours a day. The applicant has committed to revise the CECC EIPs for Watts Bar to include the phone numbers of the local government organizations.

Communications between emergency response facilities (control room, technical support center, and operational support center) and CECC are tested monthly in accordance with the "Quality Assurance Record of Periodic Testing of Communications Used in Emergencies." The Watts Bar EIP-12, "Emergency Equipment and Supplies," will be revised to include the monthly testing of communication equipment between emergency response facilities (ERFs). Communications between the ERFs and the NRC headquarters and region are tested every month. The plant medical alarm can be activated through the plant telephone system. A coordinated communication link exists for fixed and mobile medical support facilities. A communications drill of the entire system is conducted at least once each calendar year in conjunction with the annual emergency preparedness exercise.

13.3.2.7 Public Education and Information

The planning standard requires that information be made available to the public periodically basis on how the public will be notified and what its initial actions should be in an emergency (e.g., listening to a local broadcast station and remaining indoors). The principal points of contact with the news media for dissemination of information during an emergency (including the physical location or locations) are established in advance, and procedures are established for coordinated dissemination of information to the public.

The applicant, in coordination with appropriate State agencies, annually sends all residents within the 10-mile EPZ of each TVA nuclear plant (including Watts Bar) a calendar containing information about radiation, respiratory protection, sheltering, evacuation routes, special needs of the handicapped, and radioprotective drugs. The calendar tells the public what it should do in the event of an emergency and also lists contacts for gaining additional information, including local civil defense offices and TVA's toll-free Citizens

Action Line. Mailing lists for the public in the EPZ are updated annually to ensure thorough, accurate distribution of the emergency information.

TVA holds training sessions for the media at each TVA nuclear plant at least once a year to acquaint reporters with the basics of reactor operations, the health effects of radiation, emergency planning, points of contact for information during an emergency, nuclear waste management, operator training, and other pertinent subjects. TVA conducted such a meeting at Watts Bar on July 15, 1993.

Until the CECC is activated, the communications organization at Watts Bar will coordinate public information with other primary response agencies. Once the CECC is activated and staffed, the responsibility for coordinating public information with other agencies will shift to the CECC information staff. Upon activation and staffing of the Joint Information Center (JIC), the responsibility for coordinating public information will shift from the CECC to the JIC staff when and if offsite agencies are also operational at the JIC. Watts Bar has both a near-site JIC, located in the Watts Bar Training Center, and a primary JIC, located in the TVA-COC in Chattanooga, Tennessee. The near-site JIC, also designated as the local JIC, is a facility at which Watts Bar communications staff can respond to the news media present at the site and where staff can coordinate with offsite agencies in presenting emergency news briefings and responding to telephone inquiries from the public.

The Vice President, Communications, directs the activities of the emergency public information personnel at three locations: (1) the plant site where communications staff responds to the news media present at the site; (2) the CECC in the Chattanooga Office Complex (COC) where staff will develop news releases and coordinate the releases with offsite agencies; and (3) the local Joint Information Center (JIC) near the plant site where staff will coordinate with the offsite agencies in presenting emergency news briefings and responding to telephone inquiries from the public. The emergency public information organization will have sufficient staff at every location to maintain operations 24 hours a day.

In an emergency, the Manager of Power serves as the chief spokesperson for TVA at press briefings in the JIC at the TVA-COC. The CECC Information Officer is responsible for coordinating TVA news releases with the TVA centers and the State as required.

For rumor control, emergency information responsibilities are handled by three teams in the JIC at the TVA-COC. A trained media relations team will respond to telephone inquiries from the news media and at media briefings, and a trained team will respond to telephone inquiries from the public. Additionally, a trained team will monitor news media coverage. Information activities will be coordinated with offsite agencies at the JIC.

13.3.2.8 Emergency Facilities and Equipment

The planning standard requires that adequate emergency facilities and equipment to support the emergency response be maintained.

Emergency facilities needed to support an emergency response at Watts Bar include a Technical Support Center (TSC) and an Operational Support Center

(OSC). Each will be activated at the Alert or higher emergency classifications. A specific area (adjacent to the relay room) in the control building at elevation 755 feet within the protected area is designated as the TSC. The room has capabilities to communicate inside and outside the plant. The room is shielded to ensure occupancy during an emergency and, consistent with the control room, is designed to be continuously habitable during radiological emergencies. Ventilation and air conditioning for this facility have redundant or backup power systems. Data are available to the TSC staff to enable it to assess the consequences of the event and assist the control room personnel in mitigating the accident.

The OSC is located in the service building at elevation 713 feet adjacent to the radiation control laboratory within the protected area. It contains emergency team briefing areas, and there is additional space in the adjacent hallway and adjoining rooms for staging, briefing, and dispatching of maintenance teams. An alternate OSC in the conference room in the plant office building has additional space in the adjacent plant assembly room. The OSC has communications with the Technical Support Center and the control room.

Under the TVA CECC concept, a near-site emergency operations facility (EOF) is not required. TVA has established a Local Recovery Center (LRC) in Classroom 5 of the Watts Bar Training Center outside the protected area for use by NRC responders. Figure 6-C of the Watts Bar REP shows the location of the LRC in the Watts Bar Training Center. The LRC has telephone communication capabilities as well as such equipment as facsimile and copy machines, plant-specific drawings, manuals, and procedures which are located in the nearby operations training area, and computer terminals are available to allow LRC personnel access to both meteorological and dose rate information. On March 19, 1981, the NRC approved the TVA centralized emergency management concept with certain provisions. The Watts Bar LRC, as discussed above, conforms to those provisions. Activation time for the TSC, OSC, and CECC is approximately 60 minutes following the declaration of an Alert or higher classification.

Provisions have been made to inspect, inventory, and operationally check emergency equipment/instruments at least once each calendar quarter and following a drill, exercise, or declaration of an emergency. Identification of emergency kits, by general category, and their contents is included in appendices to Watts Bar EPIP-12, "Emergency Equipment and Supplies," Revision 3. Sufficient reserves of instruments are to be available to replace those removed from emergency kits for calibration or repair. Equipment will be calibrated at intervals recommended by the supplier. The applicant has made provisions for meteorological instrumentation and procedures which conform to the guidance in Supplement 1 of NUREG-0737. Provisions have also been made to obtain representative current meteorological information from other sources.

13.3.2.9 Accident Assessment

The planning standard requires that adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition be in use.

The applicant has identified plant system and effluent values characteristic of a spectrum of off-normal conditions and accidents, and has identified the plant values and other information which correspond to the example initiating conditions of NUMARC/NESP-007, "Methodology for Development of Emergency

Action Levels." These are identified in the Watts Bar REP and in EPIP-1, "Emergency Plan Classification Flowchart" (Revision 5), which specifies the kinds of instruments being used and their capabilities. The adequacy of the Watts Bar classification methodology is evaluated in Section 13.3.2.4 of this report.

Onsite capability and resources are in place to provide initial and continuing assessment throughout the course of an accident. These include process, effluent, and area monitors that read out in the control room, postaccident sampling capability, and containment monitoring.

The applicant is prepared to assess the consequences of potential or actual releases off site. Watts Bar has a van equipped to monitor the environment for radioactivity. The vehicle will be dispatched for any emergency situation of the Alert, Site Area, or General Emergency class within 30 minutes. An additional sampling team can be at the plant within 2 hours of notification. A third team will be dispatched from the Western Area Radiological Laboratory (WARL) in Muscle Shoals, Alabama. Any of the teams may be transported by helicopter or airplane. Additionally, TVA has aquatic monitoring teams located at Knoxville and Chattanooga, Tennessee, and Muscle Shoals, Alabama. These teams have boats that can be deployed to obtain samples from the river for subsequent analysis for radioactivity in the laboratories.

The applicant's accident assessment program includes a dose calculational methodology for identifying the potential scope of radiological consequences of emergency situations, including capabilities for dose projection using real-time meteorological information and for dispatch of radiological monitoring teams. A preliminary dose is projected following receipt of measured effluent release data (the source term) and meteorological data. This projection is followed up by a more detailed assessment using computerized dose models. Manual dose assessment methods are available for use should the computer be unavailable. Input to the detailed calculations includes measured source terms, projected future releases, near real-time and forecast meteorological data, and field measurements of exposure rates and airborne radioactivity in the environs around the plant. These calculations for the plume exposure, ingestion, and relocation pathways are in accordance with the "Environmental Protective Agency Manual of Protective Actions for Nuclear Incidents" (EPA 520/1-75-001-A). Field measurements are used to estimate doses and source terms (especially in the case of unmonitored releases), and to verify doses projected using models.

After termination of accidental releases to the atmosphere, integrated doses are calculated to assist in recovery/reentry operations. A combination of data, including results from modeling field exposure rate and air concentration measurements, and laboratory analyses of soil, vegetation, and water samples, are used to assess doses. TVA makes recommendations to TEMA regarding reentry based on doses calculated for exposure to ground contamination, inhalation of resuspended radioactivity, and ingestion of radioactivity in vegetables and milk.

All environmental monitoring activities for Watts Bar will be coordinated through the TEMA Radiological Monitoring Control Center (RMCC) located in the TEMA Operations Center at Alcoa, Tennessee. Environmental monitoring data will be shared between the State of Tennessee and TVA. A TVA mobile radio-analytical laboratory can be dispatched to Watts Bar to become the central

point for receipt of samples and for detailed field analysis. Samples obtained by the sampling teams may be returned to the Western Area Radiological Laboratory (WARL), which has the capability to perform further quantitative and qualitative analysis. The mobile laboratory and WARL are available at all times and can be operated 24 hours a day.

13.3.2.10 Protective Response

The planning standard requires that a range of protective actions be developed for the plume exposure pathway emergency planning zone (EPZ) for emergency workers and the public. Guidelines for the choice of protective actions during an emergency consistent with Federal guidance are developed and in place, and protective actions for the ingestion exposure pathway EPZ appropriate to the locale have been developed.

The applicant has established the means and time to warn or advise onsite personnel and individuals, contractor and construction personnel, visitors, and other persons who may be in public access areas or passing through the site or within the owner-controlled area. Provisions have been made to evacuate nonessential onsite personnel, for radiological monitoring of persons evacuated from the site, and for evacuation routes and transportation for onsite individuals to suitable offsite location(s). For persons remaining on site, provisions have been made for individual respirator protection, protective clothing, and for the administration of potassium iodide (KI).

In Watts Bar EPIP-4 and EPIP-5, "Site Area Emergency," Revision 5, and "General Emergency," Revision 5, respectively, the SED is required to initiate Watts Bar EPIP-8, "Personnel Accountability and Evacuation," Revision 5, at the Site Area and General Emergency classification levels. Provisions for the accountability of individuals on site within 30 minutes and to maintain accountability continuously thereafter are included in Watts Bar EPIP-8.

The applicant has addressed the development of protective action recommendations to be made to the State of Tennessee and local authorities in the event of a radiological emergency at Watts Bar. The bases for the choice of the recommended protective actions for the plume exposure pathway EPZ are shown in Appendix B, Watts Bar EPIP-5 (Revision 5), "General Emergency." Protective action recommendations are based both on plant conditions and radiation doses.

Prompt notification will be made directly to the offsite authorities responsible for implementing protective actions within the plume exposure pathway EPZ. The TVA REP has a table of recommended protective actions, including shelter and evacuation, with special consideration for children and pregnant women to reduce the total effective dose equivalent (TEDE) and thyroid doses from exposure to an atmospheric release, that is consistent with both the State of Tennessee and U.S. Environmental Protection Agency (EPA) guidelines. The recommended actions contain the statement that officials may implement low-impact protective actions at lower values in keeping with the principle of maintaining radiation exposures as low as reasonably achievable.

The applicant has committed to make recommendations for the ingestion exposure pathway to State and local agencies responsible for the decision to act upon such recommendations. The State of Tennessee Multi-Jurisdictional RERP states that protective actions will be based upon protective action guides developed by the EPA and U.S. Food and Drug Administration.

An evacuation time assessment study within the plume exposure pathway EPZ in accordance with the criteria set forth in Appendix 4 to NUREG-0654 is provided in Appendix 1, Annex H, of the State of Tennessee Multi-Jurisdictional Radiological Emergency Response Plan. This item is being reviewed separately by FEMA, will be tracked by TAC M89154, and will be addressed in a supplement to this SER before NRC authorizes operation above 5 percent of rated power.

13.3.2.11 Radiological Exposure Control

The planning standard requires that means for controlling radiological exposures, in an emergency, be established for emergency workers. The means for controlling radiological exposure shall include exposure guidelines consistent with EPA Emergency Worker and Lifesaving Activity Protective Action Guides.

Implementing procedures have been developed to prevent or minimize exposure to radiation for individuals at the site. These procedures include evacuation, accountability, radiological monitoring, and decontamination of nonessential personnel. Respiratory protective equipment and protective clothing are provided for essential plant personnel who would remain on site. The applicant's REP includes emergency guidelines for doses during an emergency that are consistent with the guidance on dose limits for workers performing emergency services in EPA's "Manual of Protection Action Guides and Protective Actions for Nuclear Incidents," October 5, 1991. Emergency dose limits for onsite personnel will be authorized in the SED, and dose limits for offsite TVA personnel will be authorized by the Radiological Assessment Manager in the CECC. The REP states that personnel in both cases must be made aware of possible consequences of such exposures and must be selected on a voluntary basis unless they are members of an emergency team and have previously consented to receive the exposure.

The staffing of the radiation control laboratory provides a 24-hour-a-day capability to determine doses received by emergency personnel involved in any nuclear incident. As appropriate, thermoluminescence dosimetry badges, multi-range pocket chambers or electronic dosimeters, and multiple badges are issued to emergency team members. Self-reading dosimetry must read frequently to prevent overexposure. Action levels for determining the need for decontamination are specified in TVA Nuclear Power Standard 5-1, "Radiation Protection Plan," Revision 2.

Watts Bar maintains supplies and equipment to establish a temporary decontamination area for the purpose of gross radiological decontamination and in order to evaluate and stabilize the injured. This area, complete with shower and sink, is in the service building, elevation 713 feet. Equipment and materials for decontamination and first aid are kept there. All contaminated personnel will be decontaminated to the limits specified in Watts Bar Radiological Control Instruction 102, "Contamination and Hot Particle Contact," before they are released. Additional clothing is kept on site for people who need it.

The State of Tennessee and local agencies are responsible for implementing actions to protect the health and safety of the public off site. The applicant will recommend protective actions, but the State and local governments are responsible for deciding if any public protective actions are needed and what they should be. Potential choices for reducing public exposure to radioactivity are shelter; evacuation; closing public water supplies; confiscating

crops, food, and dairy products; placing milk animals on uncontaminated stored feed; and administering potassium iodide (KI). The State of Tennessee stores KI at strategic locations (e.g., local health departments), and will distribute it to people within 5 miles of Watts Bar who want to keep it on hand. The State also issues KI to those residents in the 10-mile EPZ who volunteer to come and get it. KI is distributed to institutionalized persons and to State emergency workers.

13.3.2.12 Medical and Public Health Support

The planning standard requires that arrangements be made for medical services for contaminated injured individuals.

The applicant has made arrangements with Rhea County Medical Center and Athens Community Hospital to care for people who receive injury at the site, and to people who are exposed to radiation or to radioactive material. These facilities have emergency plans, staff training programs, and adequate equipment and supplies for receiving radiological contaminated patients.

The applicant has made provisions to locate medical equipment with trauma kits and other specified equipment on site for use by the Medical Emergency Response Team (MERT). First aid is provided by emergency medical technicians (EMTs) and medical supplies and treatment for minor injuries are available. A medical office, staffed by registered nurses, is located in the Watts Bar Training Center. Medical treatment is available during the day and evening shifts.

Arrangements have been made for transporting victims of radiological accidents to medical support facilities. A minimum of one ambulance is available on site. When ambulance transportation is indicated, transport may be provided by the fire protection EMTs on site (using the TVA ambulance) or by an ambulance service that has an agreement with TVA - Rhea County Ambulance. The MERT Team Leader will coordinate any request for offsite ambulance assistance through the SOS.

13.3.2.13 Recovery and Reentry Planning and Postaccident Operations

The planning standard requires that general plans for recovery and reentry be developed.

The decision to downgrade an incident from a higher classification to a lower classification will be made by the SED after consultation with the plant technical and operations staffs and will be coordinated with the CECC Director. All other TVA resources and other government and vendor support will be available through the TVA corporate organization to aid the SED in developing, evaluating, and implementing specific recovery and reentry operations. Procedures and plans shall then be drawn up to implement the most expeditious recovery sequence to return the plant to normal operations. The Senior Vice President, Nuclear Power, will direct the overall recovery effort. The local recovery center is available to provide additional office space near the site for the recovery effort.

The decisions to downgrade protective actions for the public will be made by the appropriate State representatives. The CECC Director will submit information to the appropriate State agencies to facilitate the decision. The CECC

Director, after consultation with the Federal responders, the State, and the SED, will announce that the emergency has terminated and the recovery phase is to be initiated.

Reentry and recovery individual and population dose estimates may be obtained using dose rate measurements or calculations and population distribution based on methodology contained in the CECC EIPs.

13.3.2.14 Exercises and Drills

The planning standard requires that periodic exercises be conducted to evaluate major portions of emergency response capabilities and to develop and maintain key skills. Also, deficiencies identified as a result of exercises or drills must be corrected.

An annual exercise is required for Watts Bar which is to include the mobilization of the applicant's personnel and resources adequate to verify the capability to respond to an accident scenario requiring response, with at least partial participation by the State of Tennessee. An exercise with full participation by State and local authorities is to be conducted at Watts Bar every two years. The scenario for the exercise will be mutually agreed to and rotated each year to ensure that all major elements of the emergency plan are tested over a 5-year period.

In accordance with 10 CFR Part 50, Appendix E, Section IV.F, the TVA REP states that an exercise to test the integrated capability of TVA, State of Tennessee, and local emergency response organizations shall be conducted within two years before the issuance of the first full-power operating license for Watts Bar. This exercise is designed to test a major portion of the basic elements existing within emergency preparedness plans and organizations.

Drills and exercises are conducted regularly to develop and maintain the key skills that are required for emergency response. Drills identified in Section 14.0 (Revision 7), "Drills and Exercises," of the TVA REP may be conducted individually or as part of a REP exercise. Drills, based on emergency conditions, will be held at least annually for response components such as fire, medical, health physics, and communications, to ensure maximum effectiveness of the plan. At the conclusion of each exercise, a critique is to be conducted at which the exercise and its participants are evaluated and comments on areas needing improvement can be obtained from both evaluating observers and participants. EP is to evaluate critique comments, develop a formal report, coordinate corrective actions for deficiencies or items needing improvement, and follow up to ensure completion of corrective actions. EP Instruction Letter, "EP Tracking System," EPIL-5, Revision 1, dated October 1, 1991, describes the mechanism to ensure effective tracking of EP commitments, critique items, routine tests, and exercise and drill requirements.

Drill scenario development and implementation is the responsibility of the organization responsible for the specific drill. Exercise scenario development is the responsibility of EP. Exercise scenario planning and development are coordinated with representatives of appropriate organizations and State agencies.

A full-participation exercise was conducted at Watts Bar on October 6 and 7, 1993. The onsite portion of the exercise is evaluated in NRC Inspection

Report 50-390/93-64, dated November 15, 1993. FEMA's assessment of the off-site portion of the exercise will be included in its findings and determination on offsite preparedness for Watts Bar. This item will be reviewed separately and will be addressed in a supplement to the SER before authorization to exceed 5 percent of rated power. The next exercise of the Watts Bar REP is scheduled for October 1994, and will include the partial participation of offsite agencies.

13.3.2.15 Radiological Emergency Response Training

The planning standard requires that radiological emergency response training be provided to those who may be called on to assist in an emergency.

TVA provides training in emergency procedures to all permanent plant personnel and all non-plant personnel expected to be on site for longer than one week. The training enables each employee to have a working knowledge of the emergency plan and his/her responsibilities and action if an emergency is declared. Personnel with specific duties and responsibilities in the WBN REP program receive instruction in the performance of their duties and responsibilities per the Nuclear Power Training Manual, TRN-30 (Radiological Emergency Preparedness Training) and as required in TVA REP Section 15.0, Training.

Training consists of initial training classes and annual retraining, drills, and activation of the alarms to maintain familiarity with the features of the emergency plan. Training and annual retraining is provided to those off-site agencies who may be involved during an emergency and includes procedures for notification, basic radiation protection, their expected roles, and site access procedures, as applicable. Watts Bar EP provides training to key site responders in the TSC and OSC. Training and annual retraining has been provided to offsite agencies (security, fire, ambulance, and hospital personnel), who may be called to assist in an emergency onsite. Emergency Planning at Corporate provides agreement hospital and ambulance support training. Watts Bar provides training for local law enforcement and security agencies and fire support training with assistance, as needed, from Operations Services at Watts Bar.

Medical Services provides emergency medical care training to its medical personnel, and selected Nuclear Power personnel at Watts Bar. After successful completion of the training, commensurate with their duties, personnel are allowed to fulfill the role of a medical care provider of the Watts Bar Medical Emergency Response Team.

13.3.2.16 Responsibility for the Planning Effort: Development, Periodic Review, and Distribution of Emergency Plans

The planning standard requires that responsibilities for plan development and review and for distribution of emergency plans be established and planners be properly trained.

The development, implementation, and maintenance of the TVA REP is the responsibility of Nuclear Power (NP). The Senior Vice President of NP has delegated the authority for overall program control of the TVA REP to the Manager, Emergency Preparedness.

The WBN REP, which is Appendix C to the TVA REP, and WBN Emergency Plan Implementing Procedures (EPIPs) are formally reviewed annually for adequacy and applicability by both TVA Corporate and Watts Bar EP organizations. Results are documented and changes identified in drills and exercises are taken into account. TVA Corporate EP initiates the review of the REP and TVA Corporate Operation Services initiates the review of the EPIPs. Administration Services and Support at WBN issues controlled revisions and ensures that all holders receive all changes. Each line affected by a change will be marked in the margin or whenever an entire page has been substantially changed or added, a statement to that effect will be made at the bottom of the page. Provisions are in place for updating applicable emergency preparedness related telephone numbers quarterly.

A detailed listing of supporting plans, sources, and procedures - by title and purpose - required to implement the plan are provided in the WBN REP.

The qualifications of TVA personnel responsible for radiological emergency planning include academic training in engineering or science. Several employees have graduate training in nuclear engineering or health physics, and all have at least 2 years' experience in engineering, health physics, or emergency planning. The training of individuals responsible for the planning effort is specified in TVA Nuclear Power Training Manual Procedures, TRN-30, "Radiological Emergency Preparedness Training," Revision 3, dated October 13, 1993.

Nuclear Quality Assurance (NQA) at TVA Corporate, which is not immediately responsible for the emergency preparedness program, audits/reviews the plan annually for compliance with existing regulations and its own internal requirements. NQA is also responsible for offering recommendations on overall plan improvement. The results of audits are documented, reported to appropriate organizational management, and retained in the respective NQA files for a period of 5 years. A key objective of the audit is that the TVA REP is coordinated with appropriate TVA, State and local governments, and non-TVA organizations.

TVA has agreements with outside organizations for radiological emergency support to furnish specific services. Copies of the letters documenting these agreements are forwarded to the Site Nuclear Security Manager or the Corporate or Site Emergency Preparedness Managers. These letters are updated every 2 years by the TVA organization requiring these services.

13.3.3 Conclusion

On the basis of its review against the guidance criteria in NUREG-0654/FEMA-REP-1, Revision 1, Supplement 1 to NUREG-0737, and NUMARC/NESP-007, Revision 2, January 1992, the staff concludes that the Watts Bar Nuclear Plant Radiological Emergency Plan, provides an adequate planning basis for an acceptable state of onsite emergency preparedness in accordance with 10 CFR 50.47(b) and (d) and Appendix E to 10 CFR 50, and that all requirements necessary for fuel loading and operation up to 5 percent of rated power have been satisfied. As a result of this conclusion, and because the applicant's new submittals referenced above supersede the applicant's previous submittals that led to the staff's safety evaluation in the 1982 SER, Proposed License Condition 27 is considered no longer valid and is hereby deleted. As stated above, remaining actions, pertaining only to operation beyond 5-percent power, will be tracked by TAC M89154 and will be evaluated in a supplement to the SER.

Once the review of the FEMA findings and determinations as to whether the State and local emergency response plans are adequate is completed, the staff's evaluation of the overall state of emergency preparedness for the Watts Bar Nuclear Plant will be presented in a supplement to the SER. As stated above, such evaluation will be tracked by TAC M89154.

15 ACCIDENT ANALYSIS

15.2 Normal Operation and Anticipated Transients

15.2.1 Loss-of-Cooling Transients

In FSAR Amendment 73, the applicant used the FACTRAN computer code to calculate transient temperature distribution for a fuel rod. FACTRAN calculates the gap heat transfer between fuel and cladding as well as between cladding and coolant. This code has been widely used for other reactors and was approved recently by the staff. The staff concludes that the use of FACTRAN code for transient heat transfer is acceptable for Watts Bar.

A partial loss-of-coolant-flow accident can be caused by a mechanical or electrical failure in a coolant pump, or by a fault in the power supply to the coolant pumps. The immediate effect of loss-of-coolant flow is a rapid increase in the coolant temperature that can damage fuel upon departure from nucleate boiling (DNB). The applicant analyzed this transient using the LOFTRAN, FACTRAN, and THINC computer codes. The staff has previously approved the use of these three codes. The results showed that the DNB ratio was not violated for this transient. On the basis of the applicant's analysis using approved computer codes, the staff concludes that the transient analysis of partial loss of forced reactor coolant flow is acceptable for Watts Bar. This effort was tracked by TACs M81887, M81888, M85774, and M85775.

15.2.4 Reactivity and Power Distribution Anomalies

15.2.4.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From Zero Power Conditions

By letters dated May 28 and November 5, 1993, the applicant responded to the staff's request for additional information (RAI) dated March 15, 1993, on the Watts Bar Unit 1 draft Technical Specifications (TSs). The applicant's letters addressed, among other matters, Question 7 of the staff's RAI.

Question 7 asked the applicant to make appropriate changes to draft TS Limiting Condition of Operation (LCO) 3.4.6 to meet the SSER 7 assumption that two reactor coolant pumps (RCPs) are operable and in operation in Modes 3 and 4 when rod withdrawal is possible.

The applicant stated that the Watts Bar licensing-basis analysis, "Uncontrolled RCCA Bank Withdrawal From Subcritical (RWFS)," assumes operation in Mode 2. An occurrence in Modes 3, 4 or 5, with two or more RCPs in operation, is bounded by the licensing-basis RWFS analysis. This is based on the analysis assumption that the reactor does not trip until the power range high neutron flux (low setting) setpoint is reached and, at the same time, two sequential RCCA banks having the maximum combined worth at the maximum speed (72 steps/minute) are withdrawn. These conservative assumptions result in the core returning to criticality and generating some heat before trip. Therefore, the primary system flow rate becomes an important consideration as a factor in the departure-from-nucleate-boiling (DNB) evaluation. Consideration of typical plant conditions in Modes 3, 4, and 5 for the RWFS event results in

a much less severe transient than would be expected in Mode 2 with two or more reactor coolant pumps in operation. In these modes, the rod control system is manually controlled. If a single failure were to occur in the rod control system, only one RCCA bank could withdraw at a slower speed than automatic rod withdrawal speed. Although not assumed in the analysis, in Modes 3, 4, and 5, with the rods capable of withdrawal, the source range high neutron flux trip is required operable and would terminate the event by tripping any withdrawn and withdrawing rods before any significant power level could be attained. The slower reactivity insertion rate and earlier trip will prevent the generation of any significant amount of power and subsequent reduction of thermal margin.

By letter dated November 5, 1993, the applicant proposed to modify LCO 3.4.6 and the associated "Bases" section to include a requirement that two reactor coolant pumps should be running whenever the rods are capable of withdrawal in Mode 4. The proposed modification of the LCO resolves the staff's concern, and conforms to the assumption used in SSER 7.

This effort was tracked by TAC M76742.

17 QUALITY ASSURANCE

In SSER 10, the staff incorporated by reference its evaluation of the revisions TVA submitted after SSER 5 was published. Since SSER 10 was issued, TVA has submitted the following additional revisions:

- (1) Letter, M. J. Burzynski to NRC, dated January 20, 1993--NRC review and acceptance in letter, A. F. Gibson to M. O. Medford, March 10, 1993.
- (2) Letter, M. J. Burzynski to NRC, dated April 8, 1993--NRC review and acceptance in letter, A. F. Gibson to M. O. Medford, June 9, 1993.
- (3) Letter, B. S. Schofield to NRC, dated August 20, 1993--NRC review and acceptance in letter, A. F. Gibson to M. O. Medford, October 12, 1993.
- (4) Letter, B. S. Schofield to NRC, dated December 2, 1993--NRC review and acceptance in letter, A. F. Gibson to M. O. Medford, January 6, 1994.

17.4 Conclusion

In SSER 5, the staff evaluated the applicant's Nuclear Quality Assurance (QA) Program, submitted by letters of February 15 and June 5, 1990. The staff concluded that "TVA's QA program description is in compliance with applicable NRC regulations and is acceptable for the operations phase at Watts Bar." However, one issue remained open pending TVA's administrative identification of safety-related features in the FSAR.

The staff reviewed the applicant's correspondence to the NRC dated July 8, 1993 as part of its efforts on the Q-List Corrective Action Program, as well as selected portions of Section 3.2 of the FSAR as updated by Amendment 80. Details of the staff's review are documented in Appendix AA of this supplement. As a result of these reviews, the staff concludes that the applicant had established appropriate programmatic controls for the identification of safety-related features at Watts Bar. This resolves the open issue in SSER 5. The staff's efforts were tracked by TAC M76972.

APPENDIX A

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

NRC Letters and Summaries

June 20, 1991	Letter, P. S. Tam to D. A. Nauman (TVA), requesting additional information on FSAR Chapter 8, Electrical Power Systems.
March 15, 1993	Letter, P. S. Tam to M. O. Medford (TVA), requesting additional information on the draft Technical Specifications and Technical Requirements Manual.
August 4, 1993	Meeting notice by P. S. Tam, regarding meeting of August 10, 1994 to discuss generic changes in the draft Watts Bar Unit 1 Technical Specifications.
August 9, 1993	Meeting notice by P. S. Tam, regarding licensing status meeting of September 3, 1993.
August 11, 1993	Letter, F. J. Hebdon to M. O. Medford (TVA), addressing retention of the condensate storage tank limiting condition for operation in the Watts Bar Unit 1 Technical Specifications.
August 11, 1993	Letter, E. J. Reis to S. A. Smith (Foundation for Global Sustainability), responding to his letter of June 18, 1993, regarding various questions on Watts Bar licensing.
August 12, 1993	Letter, F. J. Hebdon to M. O. Medford (TVA), concluding that traceability of Quality Level II structural steel meets pertinent regulation and guidance documents.
August 13, 1993	Letter, E. W. Merschoff to M. O. Medford (TVA), transmitting summary of meeting in the NRC Region II office on August 10, 1993, regarding status of employee concerns program.
August 17, 1993	Summary by P. S. Tam, regarding meeting of July 19-22, 1993, to discuss open issues in the proof and review version of the Watts Bar Unit 1 Technical Specifications.
August 18, 1993	Letter, E. W. Merschoff to M. O. Medford (TVA), transmitting summary of meeting at the Watts Bar site on August 6, 1993, regarding ongoing and planned activities.

August 20, 1993	Letter, P. S. Tam to M. O. Medford (TVA), requesting additional information on individual plant examination (IPE).
August 20, 1993	Summary by P. S. Tam, regarding meeting of August 10, 1993, to discuss open issues in the proof and review version of the Watts Bar Unit 1 Technical Specifications.
September 1, 1993	Summary by P. S. Tam, regarding meeting of August 3, 1993, to discuss use of U-bolts.
September 9, 1993	Letter, P. S. Tam to M. O. Medford (TVA), transmitting safety evaluation accepting TVA's response to the station blackout regulation, 10 CFR 50.63.
September 9, 1993	Letter, P. S. Tam to M. O. Medford (TVA), informing of partial completion of steam generator tube rupture review, and requesting additional information.
September 9, 1993	Summary by P. S. Tam, regarding licensing status meeting of September 3, 1993.
September 13, 1993	Letter, P. S. Tam to M. O. Medford (TVA), transmitting safety evaluation and position on U-bolt pipe clamp issue identified during the last integrated design inspection.
September 20, 1993	Letter, P. S. Tam to M. O. Medford (TVA), transmitting safety evaluation accepting Core Operating Limits Report for Watts Bar Unit 1.
September 21, 1993	Meeting notice by P. S. Tam, regarding meeting of October 13, 1993, to discuss use of U-bolts.
September 23, 1993	Meeting notice by P. S. Tam, regarding licensing status meeting of October 7, 1993.
September 23, 1993	Meeting notice by P. S. Tam, regarding management meeting at the Watts Bar site on October 19, 1993.
October 12, 1993	Letter, A. F. Gibson to M. O. Medford (TVA), approving Revision 3 of TVA's Nuclear Quality Assurance Plan.
October 13, 1993	Summary by P. S. Tam, regarding licensing status meeting of October 7, 1993.
October 25, 1993	Letter, P. S. Tam to M. O. Medford (TVA), requesting additional information regarding the steel containment vessel response spectra from local pressurization.
October 26, 1993	Letter, F. J. Hebdon to M. O. Medford (TVA), transmitting copies of Supplement 12 of the Watts Bar Safety Evaluation Report.

October 26, 1993 Letter, P. S. Tam to M. O. Medford (TVA), informing of upcoming team inspection of the Program for Assurance of Completion and Assurance of Quality (PACAQ).

November 1, 1993 Summary by P. S. Tam, regarding management meeting of October 19, 1993, at the Watts Bar site.

November 2, 1993 Summary by P. S. Tam, regarding management meeting of October 13, 1993, to discuss use of U-bolts.

November 9, 1993 Meeting notice by P. S. Tam, regarding meeting with the Foundation for Global Sustainability on November 12, 1993.

November 16, 1993 Summary by P. S. Tam, regarding meeting of November 12, 1993, with the Foundation for Global Sustainability.

November 24, 1994 Letter, P. S. Tam to M. O. Medford (TVA), informing of upcoming audit of U-bolt issue on site.

November 26, 1993 Letter, P. S. Tam to M. O. Medford (TVA), transmitting supplemental audit report, thus completing the staff's review of the Civil Calculation Program.

November 30, 1993 Letter, P. S. Tam to M. O. Medford (TVA), transmitting safety evaluation regarding disposition of flaw indications in cladding of the safety injection accumulator tank.

November 30, 1993 Meeting notice by P. S. Tam, regarding licensing status meeting of December 14, 1993.

December 6, 1993 Letter, P.S. Tam to M. O. Medford (TVA), informing TVA that the updated information regarding post-trip review (Generic Letter 83-28) enhances the previously approved program.

December 7, 1993 Letter, P. S. Tam to M. O. Medford (TVA), transmitting copy of environmental assessment related to TVA's request to extend expiration dates of the Watts Bar construction permits.

December 9, 1993 Letter, P. S. Tam to M. O. Medford (TVA), requesting additional information on the Offsite Dose Calculation Manual.

December 14, 1993 Letter, P. S. Tam to M. O. Medford (TVA), transmitting order to extend expiration dates of the Watts Bar construction permits.

December 17, 1993 Summary by P. S. Tam, regarding licensing status meeting of December 14, 1993.

December 28, 1993 Meeting notice by P. S. Tam, regarding licensing status meeting of January 7, 1994.

January 3, 1994 Meeting notice by P. S. Tam, regarding meeting of January 18, 1994, to discuss use of U-bolts.

January 11, 1994 Summary by P. S. Tam, regarding licensing status meeting of January 7, 1994.

January 13, 1994 Letter, A. F. Gibson to M. O. Medford (TVA), transmitting summary Inspection Report Nos. 50-390/93-89 and 50-391/93-89.

January 18, 1994 Letter, E. W. Merschoff to M. O. Medford (TVA), transmitting summary of meeting at the Watts Bar site on January 7, 1994.

January 21, 1994 Letter, P. S. Tam to M. O. Medford (TVA), informing of acceptability of VANTAGE-5H fuel design and details to be published in SSER 13.

January 26, 1994 Meeting notice by P. S. Tam, regarding licensing status meeting of February 9, 1994.

January 27, 1994 Letter, E. W. Merschoff to M. O. Medford (TVA), transmitting summary of meeting conducted at site Watts Bar site on January 20, 1994.

January 31, 1994 Letter, P. S. Tam to M. O. Medford (TVA), informing TVA of reviewer's upcoming visit to TVA's Chattanooga office to review information on emergency preparedness.

February 3, 1994 Summary by P. S. Tam, regarding meeting of January 18, 1994, to address technical issues regarding use of U-bolts.

TVA Letters

September 13, 1991 Letter, J. H. Garrity to NRC, submitting additional information on FSAR Chapter 8, Electric Power Systems.

April 20, 1993 Letter, W. J. Museler to NRC, responding to request for additional information on the draft Technical Specifications.

May 28, 1993 Letter, W. J. Museler to NRC, responding to request for additional information on the draft Technical Specifications.

August 3, 1993 Letter, W. J. Museler to NRC, submitting additional information on accumulator tank qualification.

August 5, 1993 Letter, B. S. Schofield to NRC, responding to Generic Letter 93-04, "Rod Control System Failure and Withdrawal of Rod Cluster Assemblies".

August 5, 1993	Letter, W. J. Museler to NRC, transmitting copy of NUS Corporation study assessing quality assurance program at Watts Bar.
August 6, 1993	Letter, W. J. Museler to NRC, transmitting Amendment 77 to the FSAR.
August 20, 1993	Letter, B. S. Schofield to NRC, transmitting Revision 3 of the TVA Nuclear Quality Assurance Program.
August 31, 1993	Letter, W. J. Museler to NRC, informing NRC that the Microbiologically Induced Corrosion Special Program was fully implemented.
September 1, 1993	Letter, W. J. Museler to NRC, submitting additional information on the Design Baseline Verification Program CAP.
September 1, 1993	Letter, W. J. Museler to NRC revising a commitment in the Watts Bar Nuclear Performance Plan.
September 9, 1993	Letter, B. S. Schofield to NRC, transmitting Revision 0-N of the Physical Security Plan.
September 10, 1993	Letter, W. J. Museler to NRC, committing to submit the inservice inspection program within 6 months after receiving operating license.
September 13, 1993	Letter, W. J. Museler to NRC submitting additional information on containment pressure due to postulated main steamline break accident.
September 15, 1993	Letter, W. J. Museler to NRC, transmitting the revised Fire Protection Program.
September 20, 1993	Letter, B. S. Schofield to NRC, responding to Generic Letter 93-04.
September 22, 1993	Letter, W. J. Museler to NRC, submitting additional information on emergency action levels for the Emergency Preparedness Plan.
September 28, 1993	Letter, W. J. Museler to NRC, submitting corrected information on the Fire Protection Program.
September 30, 1993	Letter, W. J. Museler to NRC, transmitting Revision 6 of the Design Baseline Verification Program CAP.
October 6, 1993	Letter, B. S. Schofield to NRC, transmitting copy of certain annexes of Tennessee Multi-Jurisdictional Radiological Emergency Plan.
October 15, 1993	Letter, W. J. Museler to NRC, responding to Generic Letter 92-01, "Reactor Vessel Structural Integrity."

October 20, 1993	Letter, W. J. Museler to NRC, submitting information on Unit 2 reactor vessel head pressure housing weld examination.
October 25, 1993	Letter, W. J. Museler to NRC, submitting additional information on U-bolt supports.
November 1, 1993	Letter, W. J. Museler to NRC, transmitting revised response to certain portions of Generic Letter 83-28, "Salem ATWS Events."
November 1, 1993	Letter, W. J. Museler to NRC, responding to request for additional information on seismic load for HVAC duct supports.
November 5, 1993	Letter, W. J. Museler to NRC responding to a question by the staff on the draft Technical Specification.
November 8, 1993	Letter, W. J. Museler to NRC, submitting additional information related to the Eagle-21 process protection system.
November 10, 1993	Letter, W. J. Museler to NRC, informing NRC of recent changes to the ECCS evaluation model.
November 10, 1993	Letter, M. O. Medford to NRC, requesting an extension of the expiration dates of the Watts Bar construction permits.
November 18, 1993	Letter, W. J. Museler, transmitting Westinghouse Topical Report WCAP-13869, Revision 1, which provides additional information on the Eagle-21 process protection system.
November 26, 1993	Letter, W. J. Museler to NRC, submitting additional information on cable tray and raceway fire barrier program.
November 30, 1993	Letter, W. J. Museler to NRC, committing to full implementation of station blackout fixes before fuel load of each unit.
December 2, 1993	Letter, B. S. Schofield to NRC, submitting Revision 3 of TVA Nuclear Quality Assurance Plan.
December 17, 1993	Letter, W. J. Museler to NRC, providing TVA's position regarding compliance with Regulatory Guide 1.75.
December 21, 1993	Letter, W. J. Museler to NRC, submitting information on implementation of the Cable Issues CAP.
December 27, 1993	Letter, W. J. Museler to NRC, responding to the staff's questions on the Individual Plant Examination (Generic Letter 88-20.

December 27, 1993	Letter, B. S. Schofield to NRC, transmitting Revision 4 of Topical Report TVA-NPOD89-A, "Nuclear Power Organization Description."
December 27, 1993	Letter, W. J. Museler to NRC, responding to the staff's request for additional information on the Eagle-21 process control system.
December 28, 1993	Letter, W. J. Museler to NRC, submitting Amendment 78 to the FSAR.
December 30, 1993	Letter, W. J. Museler to NRC, submitting Amendment 79 to the FSAR.
December 30, 1993	Letter, W. J. Museler to NRC, submitting additional information on the structural aspects of the design-basis accident spectra.
December 30, 1993	Letter, W. J. Museler to NRC, submitting additional information on seismic load for HVAC duct supports.
December 30, 1993	Letter, W. J. Museler to NRC, informing NRC of complete implementation of the Containment Cooling Special Program.
January 3, 1994	Letter, W. J. Museler to NRC, transmitting Revision 3 of the Cable Issues CAP.
January 7, 1994	Letter, B. S. Schofield to NRC, transmitting Revision 4 of the Topical Report TVA-NQA-PLN89-A, "Nuclear Quality Assurance Program."
January 19, 1994	Letter, W. J. Museler to NRC, responding to Bulletin 90-01, Supplement 1.
January 20, 1994	Letter, W. J. Museler to NRC, requesting an extension to TVA's commitment to submit results of the individual plant examination, external events analysis.
January 28, 1994	Letter, W. J. Museler to NRC, informing of complete implementation of the Q-List CAP.
January 31, 1994	Letter, W. J. Museler to NRC, transmitting Amendment 83 to the FSAR.

APPENDIX E
PRINCIPAL CONTRIBUTORS

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APPENDIX AA*

SUPPLEMENTAL SAFETY EVALUATION: CORRECTIVE ACTION PROGRAM ON THE Q-LIST

*Appendix AA was originally published as an enclosure to a letter dated March 17, 1994, from P. S. Tam (NRC) to O. D. Kingsley (TVA). The enclosure is reproduced in its entirety.



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENCLOSURE

SUPPLEMENTAL SAFETY EVALUATION
BY THE OFFICE OF NUCLEAR REACTOR REGULATION
CORRECTIVE ACTION PROGRAM ON Q-LIST
TENNESSEE VALLEY AUTHORITY
WATTS BAR UNIT 1
DOCKET NO. 50-390

1.0 INTRODUCTION

By letter dated November 7, 1990, TVA responded to the staff's request of August 16, 1990, for supplemental information regarding the Q-List Corrective Action Program (CAP), Revision 3. Additional comments regarding this subject were provided in the NRC's letter to TVA dated January 23, 1991. TVA provided a response to these comments in a letter dated July 8, 1993, along with a response to an issue which was identified in NRC Inspection Reports 50-390, 50-391/90-08 concerning equipment tagging in relation to revised Q-List interfaces and program implementation. The staff has reviewed this response and documented the results below.

2.0 EVALUATION

The following evaluation corresponds, item-by-item, to the staff's comments in a letter to TVA, dated January 23, 1991:

1. This issue involved TVA's submission of Revision 3 of the Q-List CAP Plan to the NRC for review without having first submitted Revision 2 of this document. TVA acknowledged that the staff should have been advised of their decision to withhold the issuance of Revision 2 because extensive revisions of the Q-List CAP Plan were required. Furthermore, TVA stated that it is currently their policy to present CAP revisions affecting scope and/or methodology to the NRC. Based on the staff's review of the information related to this issue, the staff determines that TVA's response is acceptable.
2. This issue involved the resolution of certain attributes associated with the implementation of the Q-List which were delineated in TVA's letter dated November 7, 1990. In particular, items which required confirmation included: (1) the proper understanding and usage of the Q-List by site personnel, (2) development of appropriate procedural controls, (3) the availability of current data in both hardcopy and electronic format, and (4) the establishment of the correlation between the Q-List and Table 3.2.1 of the Final Safety Analysis Report.

Items 1 through 3, along with other programmatic aspects of the Q-List, were examined in detail during the NRC's inspection of the Q-List CAP implementation at the 75-percent completion phase. The results of this inspection effort were documented in NRC Inspection Report Nos. 50-390, 50-391/93-68. Based on the results of this inspection effort, no adverse conditions were identified and the staff concluded that the development and implementation of the new Q-List was satisfactory. With respect to Item (4), the staff reviewed the correlation between the seismic Category I structures identified in FSAR Table 3.2-1 and the inclusion of the structures in the 92QL Q-List (the staff reviewed this site document while performing an inspection, IR 50-390/93-204). No discrepancies were identified as a result of the review and the staff determines that TVA had appropriately addressed all aspects of this item.

3. TVA indicated in its November 7, 1990, response that FSAR Question 260.2 would be updated in a proposed amendment to reflect Q-List changes in Subsection 17.2-1 which resulted from Amendment 57. Subsequent correspondence from TVA to the NRC dated September 24, 1991, outlined TVA's plan to delete the questions-and-responses section from the FSAR since the licensing basis for Watts Bar Nuclear Plant is the FSAR itself. Given this methodology, TVA conveyed its intention to no longer update FSAR Question 260.2. Accordingly, the staff reviewed the current disposition of the items addressed by FSAR Question 260.2 to determine if they were correctly captured in TVA's Q-List CAP. Based on the staff's review of the current Q-List, Q-List General Notes, and applicable site standard practice procedures, the staff determines that the items associated with Question 260.2 had been properly addressed by existing programmatic controls. Therefore this item is closed.
4. This item involved the issuance of Revision 4 of the Q-List CAP Plan without the required dating of the document by the authorizing personnel. In response to this issue, TVA indicated that the form cover sheet example contained in Site Administrative Instruction (SAI) 10.02 currently provides a place for dates corresponding to each approval signature. The staff examined the procedural controls specified in Procedure SAI 10.02 and verified that the subsequent revision of the Q-List CAP Plan had been properly annotated with the required dates. Based on the review of these corrective actions, the staff determined that this item had been properly resolved.

The staff also reviewed TVA's response to a comment in Inspection Reports 50-390, 50-391/90-08, involving potential equipment identification tagging issues. Specifically, the staff noted that design change notice implementation activities might require additional actions including equipment tagging.

Based on review of this item, the staff determined that TVA issued the new 92QL Q-List using Design Change Notices which added component identities. Additionally, for design changes which affected component identifiers, TVA established procedural controls, to ensure that components are tagged/retagged as necessary. The staff determines that these corrective actions adequately resolve the identified comment, and this item is resolved.

3.0 CONCLUSION

Based on the results of the staff's evaluations of the referenced documents, the staff concludes that TVA has fully addressed the staff's concerns as addressed in the above referenced documents.

Principal Contributor: Robert M. Latta

Dated: March 17, 1994

APPENDIX BB

TECHNICAL EVALUATION REPORT FOR WATTS BAR FEEDWATER CHECK VALVE SLAM ANALYSIS

Walter Grossman, Brookhaven National Laboratory

**TECHNICAL EVALUATION REPORT FOR
WATTS BAR FEEDWATER CHECK VALVE SLAM ANALYSIS**

1.0 INTRODUCTION

1.1 Purpose

This Technical Evaluation Report (TER) presents the results and conclusions of Brookhaven National Laboratory's (BNL) review and evaluation of the technical adequacy of TVA's Feedwater Check Valve Slam Analysis for Unit 1 of the Watts Bar Nuclear Plant (WBN).

1.2 Background

Outstanding issue 20(a) in the Watts Bar Safety Evaluation Report, NUREG-0847, Supplement 6 (Reference 1), Section 3.9.1, Special Topics for Mechanical Components, concerns TVA's analysis of the feedwater piping for the water hammer forces resulting from a check valve slam following a postulated rupture at the main header in the turbine building.

TVA stated (Reference 2) that several supports on the feedwater line are loaded beyond their ultimate capacities during the water hammer event. In addition, TVA further stated at a meeting with NRC representatives at the Watts Bar site on September 24-25, 1991 (Reference 3), that strengthening modifications would be difficult to make on several supports because of space limitations. As a result, TVA proposed performing a non-linear elastic-plastic finite element analysis of the feedwater system that includes modeling of support failures in the analysis and the use of the acceptance criteria in the ASME B&PV Code, Section III, Appendix F. Reference 1 concludes "The staff considers this criterion and the applicant's proposed method of analysis an open issue requiring further staff review."

1.3 BNL's Review Scope and Methodology

The BNL scope of work was to review the responses from the applicant and prepare a Technical Evaluation Report (TER) that contains a summary of the work performed by TVA and BNL's evaluations and conclusions concerning the technical adequacy of the TVA analysis in satisfying NRC regulatory requirements.

TVA completed the check valve slam (CVS) analysis of the feedwater piping system and issued a report (Reference 4) containing a summary of the analyses and results. BNL received the report from TVA in October 1992 and initiated a detailed review and evaluation of the methodology, computer models, assumptions, evaluation criteria and results presented in the report. TVA's report consisted of a 7 page enclosure providing a cursory summary of the analyses and results of the feedwater system check valve slam analysis.

**TECHNICAL EVALUATION REPORT FOR
WATTS BAR FEEDWATER CHECK VALVE SLAM ANALYSIS**

1.3 BNL's Review Scope and Methodology (Continued)

After reviewing Reference 4, BNL concluded that the applicant's enclosure lacked sufficient details to evaluate the technical adequacy of the response and prepared a request for additional information (RAI) which was submitted to TVA (Reference 5). A detailed response to BNL's RAI was prepared by TVA (Reference 6) and was received at BNL in January 1993.

BNL reviewed TVA's responses (References 4 and 6) concentrating on evaluating the technical adequacy of the analytical methods, the acceptance criteria, and the reasonableness and acceptability of the results. In its RAI (Reference 5), BNL had requested clarifying information with respect to the following aspects of the analysis:

- 1) source and adequacy of Code sponsored stress-strain tests
- 2) generation of the seismic input time histories
- 3) similarity claims for loops, bypass & wet layup lines
- 4) technical adequacy of the analysis input/output
- 5) verification/qualification of ANSYS nonlinear analysis
- 6) source of physical/mechanical properties vs. temperature
- 7) use of CMTR values
- 8) simultaneous blowdown of piping loops
- 9) differential movement of supports in adjacent structures
- 10) postprocessing & classification of stress results

BNL's methodology for the technical review and evaluation of the analysis was to scan and spot check the responses provided by TVA in References 4 and 6; detailed checking of the full analyses and/or the responses is beyond the BNL scope of review.

BNL utilized the following criteria to judge the validity of the methodology and the technical adequacy of the nonlinear analysis:

- 1) development of the computer model and input data should be based on conservative assumptions and engineering judgement
- 2) the computer program utilized has been verified and qualified for nonlinear analysis of piping systems
- 3) the results obtained from the computer output have been correctly classified and compared to appropriate Code criteria
- 4) the results are shown to satisfy applicable NRC regulatory requirements.

**TECHNICAL EVALUATION REPORT FOR
WATTS BAR FEEDWATER CHECK VALVE SLAM ANALYSIS**

2.0 DESCRIPTION OF ANALYSES PERFORMED BY TVA

TVA's initial response (Reference 4) described the scope of the nonlinear analyses performed, the analysis models and the loading conditions as follows:

Analyses Performed

Main Feedwater Lines

The main feedwater lines for loops 1, 2 and 4 were uniquely analyzed. The main feedwater line for loop 3 is qualified by similarity to loops 2 and 4.

Bypass and Wet Layup Lines

The lines for loops 1 and 4 were uniquely analyzed; the lines for loops 2 and 3 are qualified by similarity.

Creation of ANSYS Models and Input Data Listing

Supports

Several supports have been upgraded in the analysis to maintain structural integrity based on meeting ASME Appendix F requirements during the combined postulated loading scenario. To determine the design loads for these supports, the analysis utilizes appropriate elastic stiffnesses and infinite capacity in the model. Supports which were not modified are modeled with a zero capacity once the applied load exceeds the calculated support ultimate capacity. Whip restraints are modeled using gap elements.

Material Properties

Trilinear stress-strain curves are used for the nonlinear elements. The curves are based on values at temperature derived from ASME sponsored tests. For linear elements, the ASME Code value for modulus, E, is used.

Loading Conditions

Initial Conditions

Pressure, deadweight, temperature and thermal anchor movements at the steam generators were considered.

Seismic Loads

Three orthogonal (NS, EW and Vertical) input time histories were generated for each structure using the envelope of OBE/SSE and Sets B+C with Regulatory Guide damping. The time histories are approximately 20.5 seconds long with the strong motion beginning at 3 seconds. The overall maximum time step during the seismic event was 0.005 seconds, with a maximum time step of 0.001 seconds during the check valve slam event.

**TECHNICAL EVALUATION REPORT FOR
WATTS BAR FEEDWATER CHECK VALVE SLAM ANALYSIS**

2.0 DESCRIPTION OF ANALYSES PERFORMED BY TVA (Continued)

Loading Conditions (Continued)

Check Valve Slam Force-Time History Development

New versions of RELAP/REFORCE were utilized to generate the check valve slam (CVS) force-time histories. The RELAP models consider simultaneous blowdown of all loops due to the header break. The check valve model was revised to more accurately represent the valve disc geometry.

Sensitivity Analyses

Sensitivity analyses were performed for each main feedwater line analysis to determine whether the calculated stresses (due to operating + seismic + CVS loads) are sensitive to the start time of the CVS event with respect to the seismic time history.

3.0 TECHNICAL EVALUATION OF TVA'S CHECK VALVE SLAM ANALYSIS

BNL's review and evaluation of the technical adequacy of TVA's nonlinear check valve slam analysis of the feedwater loops is based on the information presented by TVA in references 4 and 6. Based on a review of TVA's initial submittal documenting the completion of the feedwater check valve slam analysis (Reference 4), BNL prepared and submitted a request for additional clarifying information. TVA's second submittal (Reference 6) provided substantial documentation, analysis details, drawings and ANSYS input/output data in response to BNL's RAI. TVA responded to all of BNL's requests and provided sufficient information to permit BNL to evaluate the technical adequacy of the analysis.

3.1 Source and Adequacy of Code Sponsored Stress-Strain Tests

Reference 4 stated that the material properties used in the nonlinear analysis had been revised to be consistent with the results of ASME Code sponsored tests. BNL requested TVA (Reference 5) to identify the ASME Code sponsored tests and to describe the material properties which were revised.

TVA responded (Reference 6) that the ASME had sponsored developmental testing of SA-333, Grade 6, ferritic steel piping material to characterize its stress-strain relationship in support of Section XI of the Code. The source data used to define the stress-strain curves were taken from EPRI Report NP-6045 and NUREG/CR-4082. Data from these 2 sources were used to construct a lower bound true stress-strain curve (Figure 1 of TVA's response) valid for temperatures up to 550°F. The lower bound curve was compared to CMTR data for the feedwater piping material to ensure that the lower bound material curve is representative of the installed material. The trilinear true stress-strain curve used in the analysis (Figure 2) was constructed from and closely represents the lower bound curve of Figure 1.

**TECHNICAL EVALUATION REPORT FOR
WATTS BAR FEEDWATER CHECK VALVE SLAM ANALYSIS**

3.0 TECHNICAL EVALUATION OF CHECK VALVE SLAM ANALYSIS (Cont'd.)

3.1 Source/Adequacy of Code Stress-Strain Tests (Cont'd.)

BNL reviewed the information in TVA's response and finds that the trilinear true stress-strain curve used in the nonlinear analysis of the feedwater lines is a reasonable and acceptable representation of the ASME lower bound curve. In addition, the key numerical values of the yield stress (S_y), the ultimate strength (S_u) and the modulus of elasticity (E) from the trilinear curve were independently checked by BNL against and correspond closely with the same values in the 1992 ASME Code, Section II, Part D, Material Properties. The use of the trilinear stress-strain curve developed by TVA for this analysis is considered technically adequate and acceptable.

3.2 Generation of Seismic Input Time Histories

TVA generated individual seismic input time histories meeting Standard Review Plan (SRP) requirements for four separate structures (Reference 4). BNL requested TVA (Reference 5) to describe the methodology for generating the seismic input time histories and to identify the specific SRP requirements utilized.

TVA responded (Reference 6) that three orthogonal time histories were developed for each structure from an initial "seed" time history using an iterative Fourier analysis/synthesis procedure. The requirements in II.1.b of SRP Section 3.7.1 (Reference 7) were met by synthesizing each time history such that the 3% damped spectra envelopes the relevant target spectra utilizing the specified frequency intervals and enveloping criteria. Target spectra were the B+C set for the structures supporting the four feedwater lines. In addition, the analysis satisfied the criteria in II.6.b(2) of SRP Section 3.7.2 (Reference 8) by generating the time histories such that the components of the earthquake motions specified in the 3 mutually orthogonal directions are statistically independent.

Based on the information contained in TVA's response which describes the generation of the seismic input time histories, BNL considers the methodology utilized for their generation to be acceptable.

3.3 Similarity Claims for FW Lines, Bypass and Wet Layup Lines

TVA claimed (Reference 4) that feedwater loop 3 is qualified by similarity to loops 2 and 4. In addition, qualification claims were made for loops 2 and 3 of the bypass and wet layup lines. BNL requested TVA (Reference 5) to provide the logic to substantiate the similarity claims in view of the differences in the quantity and location of the rigid restraints (RR), snubbers (DS) and pipe whip restraints (PD) of the various loops.

**TECHNICAL EVALUATION REPORT FOR
WATTS BAR FEEDWATER CHECK VALVE SLAM ANALYSIS**

3.0 TECHNICAL EVALUATION OF CHECK VALVE SLAM ANALYSIS (Cont'd.)

3.3 Similarity Claims for FW, Bypass, and Layup Lines (Cont'd.)

TVA responded (Reference 6) with drawings and a detailed explanation providing a logical basis for the similarity claims made in the earlier report. Inside containment, the location, quantity and restraint directions of the pipe whip restraints were shown to be essentially identical for feedwater loops 3 and 4. Outside containment, review of whip restraint drawing 48W1708-01 shows that, similarly, feedwater loops 2 and 3 have essentially identical pipe whip restraints.

A review of the drawings and an evaluation of the explanations provided in the TVA response indicates that the rigid restraints and snubbers for feedwater loop 3, both inside and outside containment, are essentially similar. The loads for the support evaluations of loop 3 were obtained from an envelope of the corresponding support loads from the analyses of loops 1, 2 and 4.

The drawings and the explanations provided by TVA to demonstrate equivalency of loops 2 and 3 to loops 1 and 4 (which were uniquely analyzed) of the bypass and wet layup lines were reviewed by BNL.

BNL found that the explanations provided by TVA do substantiate the similarity claims for the various lines and that the conclusions reached by TVA are based on reasonable assumptions and engineering judgement. BNL concurs that the qualification of feedwater loop 3 and bypass/wet layup lines 2 and 3 by similarity with the analysis results for the other lines which were uniquely analyzed is technically acceptable.

3.4 Technical Adequacy of the Analysis Input/Output

TVA's initial submittal (Reference 4) lacked sufficient details for a technical adequacy review of the computer model, the ANSYS input file, the ASME Code classification of the output and comparison to appropriate Code criteria. BNL requested TVA (Reference 5) to provide the following information for a typical loop: the ANSYS input file listing, an isometric drawing of the loop and a whip restraint identifying node and element numbers, ANSYS postprocessing output and plots, and sections of the final stress report containing the discussion and summary of the results.

TVA responded (Reference 6) by providing all of the requested information for Feedwater Loop 1 since this loop was found to have the maximum stresses. BNL reviewed the entire response package for consistency with the procedures in the ANSYS User's Manual and the applicable criteria in the ASME Code. The ANSYS input data file and the stress report were reviewed in detail. The remainder of the response package was scanned and spot checked since detailed checking of the analysis is beyond BNL's review scope.

**TECHNICAL EVALUATION REPORT FOR
WATTS BAR FEEDWATER CHECK VALVE SLAM ANALYSIS**

3.0 TECHNICAL EVALUATION OF CHECK VALVE SLAM ANALYSIS (Cont'd.)

3.4 Technical Adequacy of the Analysis Input/Output (Cont'd.)

BNL found that the analysis was conducted using appropriate methods as given in the ANSYS User's Manual and that the results were correctly evaluated in accordance with the applicable analysis rules and criteria in the ASME Code. Based on its review of the response package, in particular the input/output, and the classification and evaluation of the results, BNL finds TVA's analysis of Feedwater Loop 1 technically adequate and acceptable.

3.5 Verification/Qualification of ANSYS Nonlinear Analysis Methods

TVA stated that research and review of published results for comparisons of nonlinear analysis versus testing were performed to verify the propriety and accuracy of the ANSYS model and analysis methods used in the feedwater system assessment (Reference 4). BNL requested TVA (Reference 5) to identify the papers reviewed and to discuss the results of their review.

TVA's response (Reference 6) discusses both verification, i.e., that the program provides accurate results consistent with the algorithms, and qualification, i.e., that the computed results reasonably predict actual behavior. BNL agrees with TVA that the ANSYS Verification Manual demonstrates that, for simplified test cases, the ANSYS results compare very well with known textbook solutions. The advantage of this approach is the ability of comparing ANSYS results to a closed-form solution. However, the limitations of this approach are that all of the element types and the analysis option (KAN4, Nonlinear Transient Dynamic) utilized in the feedwater piping analyses are not assessed together in one verification test case. Element types and analysis options are verified against known closed-form solutions either individually or in limited combinations as shown in the ANSYS Verification Manual.

Swanson Analysis Systems, Inc., (SASI), is the developer of the ANSYS finite element computer program. According to SASI, the ANSYS program is developed, maintained, tested and supported under the SASI Quality Assurance Program, which is set up to meet the requirements of the NRC, Rules and Regulations, Title 10, Chapter 1, CFR, Part 50, Appendix B. The error reporting system which is part of the QA program is designed to address the requirements of Title 10, Chapter 1, CFR, Part 21, "Reporting of Defects and Noncompliances," for commercial grade items such as ANSYS. More detailed information about the ANSYS QA Program is described in the SASI letter and abstract in Attachment 1. Also included in Attachment 1 is a description of "ANSYS Verification and Validation" procedures from ANSYS News, First Issue, 1993.

Based on SASI's extensive QA Program and industry experience as reported in papers in the open literature, BNL accepts SASI's claim with regard to verification of the ANSYS program.

**TECHNICAL EVALUATION REPORT FOR
WATTS BAR FEEDWATER CHECK VALVE SLAM ANALYSIS**

3.0 TECHNICAL EVALUATION OF CHECK VALVE SLAM ANALYSIS (Cont'd.)

3.5 Verification/Qualification of ANSYS Nonlinear Analysis Methods (Cont'd.)

The aspect of computer program qualification, on the other hand, has not been addressed in depth by developers of finite element computer programs particularly for nonlinear analysis of dynamically loaded piping systems.

To support its claim of ANSYS program qualification for the feedwater loop analyses, TVA presented abstracts from three published papers which document ANSYS nonlinear analyses. In each paper, several of the ANSYS elements used in the solutions were the same as those used in the feedwater nonlinear analysis. TVA claims that these papers demonstrate that the nonlinear analysis option in ANSYS can reliably predict the behaviour and thus qualify solutions of dynamically loaded nonlinear piping systems.

BNL reviewed TVA's submittal and found that two (2) of the three (3) papers cited by TVA discuss nonlinear analysis and confirmatory testing of an automobile door frame and a steam turbine casing. The results/conclusions cited in these 2 papers are only indirectly applicable to qualification of analyses of dynamically loaded piping systems since they are nonlinear analyses but do not utilize the same ANSYS element types. The third paper, "Nonlinear Analysis of Offshore Pipelaying Problems," is closely related to the feedwater piping analyses but not applicable for qualification because the paper presents a comparison of ANSYS results to results computed from industry standard formulae and not to test results.

BNL is not aware of any source in the open literature containing comparisons of results of nonlinear ANSYS analyses of dynamically loaded piping systems with full scale model testing. Although confirmatory testing of piping systems is not available for qualifying the nonlinear analyses performed by TVA for the feedwater loops, this does not imply that the results of the TVA analyses are incorrect.

BNL's review and evaluation of SASI's QA program, the computer model, the input/output files, and the evaluation methodology used by TVA provide an adequate basis for resolving the concerns with regard to the lack of qualification testing of the ANSYS program for nonlinear piping system analyses. BNL concludes that the use of the ANSYS program for the analysis of the feedwater loops for the check valve slam loads is an appropriate analytical tool and provides a reasonable estimate of the response of the piping system.

**TECHNICAL EVALUATION REPORT FOR
WATTS BAR FEEDWATER CHECK VALVE SLAM ANALYSIS**

3.0 TECHNICAL EVALUATION OF CHECK VALVE SLAM ANALYSIS (Cont'd.)

3.5 Verification/Qualification of ANSYS Nonlinear Analysis Methods (Cont'd.)

BNL was additionally concerned about the use of ANSYS element STIF52, 3-D Interface, which is used as the gap element for the pipe whip restraints by TVA in the feedwater loop analyses in view of BNL's recent experience with this ANSYS element in a similar nonlinear analysis of a reactor coolant bypass line with seismic stops. The analysis results of the bypass line were found to be unsatisfactory because of convergence problems with the STIF52 gap elements which kept oscillating between the open and closed condition. A comparison of the input files of the two analyses showed that the bypass line analysis had utilized the KEYOPT(3) option of the STIF52 input (small stiffness assigned to an open or sliding gap) which had been cited in ANSYS Class 3 Error Report 88-59. On the other hand, the STIF52 element in TVA's analysis did not utilize the KEYOPT(3) option but used KEYOPT(6)=2 instead (convergence based on changing element status within a 10% tolerance on tangential force). A search revealed that there are no outstanding ANSYS Error Reports on KEYOPT(6).

BNL was further concerned about the duration of the closed (i.e., active) status of the STIF52 gap element in the analysis, and requested TVA to provide the time, duration and identity of the pipe whip restraints which contact the pipe during the check valve slam transient. TVA responded with the requested information (Reference 9). A detailed review of TVA's response indicated that 4 pipe whip restraints on Loop 1 contact the pipe during the transient. However, a review of the times and durations of contact indicated that there is no contact at the time of maximum stress in the pipe, i.e., none of the pipe whip restraints are active at the time when the maximum stress occurs.

The results of the reviews conducted by BNL resolve the concerns with respect to the use of the STIF52 element in TVA's nonlinear analyses.

3.6 Source of Physical/Mechanical Properties vs. Temperature

BNL requested that TVA provide the source of the mechanical and physical properties of the piping materials that were used in the analysis (References 5). TVA replied (Reference 6) that the source of the properties used to define the nonlinear stress-strain piping material curve for temperatures up to 550°F is discussed in Section 3.1. Values for the coefficient of thermal expansion (alpha) were taken from the ASME B&PV Code, Section III, Appendices, Table I-5.0. BNL is satisfied that the correct source and values of temperature dependent mechanical and physical properties were used in the nonlinear analysis.

**TECHNICAL EVALUATION REPORT FOR
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3.0 TECHNICAL EVALUATION OF CHECK VALVE SLAM ANALYSIS (Cont'd.)

3.7 Use of CMTR Values

BNL questioned whether CMTR values of the piping material were used in the analysis (Reference 5). TVA stated that CMTR values were not used in the analysis; material properties used are discussed in Sections 3.1 and 3.6. TVA's response is acceptable.

3.8 Simultaneous Blowdown of Piping Loops

TVA claimed that the methodology of considering simultaneous blowdown of all piping loops due to a header break (rather than considering one loop at a time) is both conservative and an improvement to the RELAP analysis (Reference 4). BNL requested TVA to provide the technical basis for this claim (Reference 5).

TVA responded (Reference 6) that improvements made to the RELAP/REFORCE programs included expanding the model to all 4 loops, establishing break locations compatible with FSAR commitments, improving the geometry of the check valve model, and changing the piping system to a zero slope to ensure consistency with the REFORCE post processor. BNL agrees that these enhancements are more realistic because they are representative of the actual physical configuration of the system. The methodology used by TVA is technically correct and acceptable.

3.9 Differential Movement of Supports in Adjacent Structures

TVA's letter (reference 2) stated that in the analysis, differential movements of support points in adjacent structures are characterized by constant relative displacements of support points superimposed over the inertial displacements. BNL requested (Reference 5) additional clarifying information with respect to implementation of the methodology described by TVA.

TVA responded (Reference 6) that for the final nonlinear analysis, separate seismic displacement time histories are generated for each support structure. Loops 1 and 4 are supported by 3 structures (the RCL, ICS and ACB); loops 2 and 3 are supported by the RCL, ICS and the NSV (north steam valve room). Within each structure, the time histories corresponding to each structure were imposed on the support points. The differential displacements between the time histories for adjacent structures were compared to, and shown to conservatively envelope, the differential displacements from the building analyses.

BNL agrees that the methodology utilized by TVA to account for the differential movements of support points in adjacent structures is appropriate and acceptable.

**TECHNICAL EVALUATION REPORT FOR
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3.0 TECHNICAL EVALUATION OF CHECK VALVE SLAM ANALYSIS (Cont'd.)

3.10 Postprocessing and Classification of Stress Results

In Reference 4, TVA described the ANSYS analysis results and the criteria utilized to evaluate the structural adequacy of the feedwater loops. BNL found that the information in Reference 4 lacked sufficient detail for a review and technical evaluation and requested TVA (Reference 5) to provide additional clarifying information.

TVA responded (Reference 6) to all of BNL's requests for additional information as follows:

- a) The applicable ASME Code criteria used for evaluating the analysis results is listed in the WBN FSAR as the 1980 Code up to and including the Winter 1982 Addenda. Table F-1322.2-1 and F-1324.6 of the Winter 1982 Addenda were utilized. Calculated stresses were compared to the allowables from F-1341.2 of the Summer 1982 Addenda as follows:

F-1341.2(a): General Primary Membrane Stress Intensity
 $P_m \leq 0.7(S_u) = 45.92 \text{ ksi}$
F-1341.2(b): Maximum Primary Stress Intensity at Any Location
 $P_L \leq 0.9(S_u) = 59.04 \text{ ksi}$
- b) The term "stress" used in TVA's report is generic and means "stress intensity."
- c) ANSYS time history post processor, POST 26, was used to scan the output results to determine the maximum stress intensity.
- d) Maximum calculated stresses were found to be located in the elbow(s) immediately adjacent to the steam generator. For Loop 1, the maximum stresses were found to be in elements 4 and 5.
- e) For the Loop 1 elbow, the maximum calculated stress intensity was 52.0 ksi which is below the allowable primary stress intensity at any location of 59.04 ksi.
- f) There is only one point in time during the transient that the value of P_m is exceeded at more than one geometric location. During a 4-millisecond period, the stress intensity is 48.4 ksi at the end of the elbow and 47.3 ksi at the midpoint of the elbow. These values exceed the general primary membrane stress intensity allowable value (P_m) of 45.92 ksi by 5.4% and 3.0% but are below the maximum primary stress intensity allowable at a local location value (P_L) of 59.04 ksi. Since the stresses are highly localized in time and location, BNL considers them to be more representative of a local rather than a general stress condition and are acceptable when compared to the P_L local membrane allowable of 59.04 ksi.

**TECHNICAL EVALUATION REPORT FOR
WATTS BAR FEEDWATER CHECK VALVE SLAM ANALYSIS**

3.0 TECHNICAL EVALUATION OF CHECK VALVE SLAM ANALYSIS (Cont'd.)

**3.10 Postprocessing and Classification of Stress Results
(Cont'd.)**

The Watts Bar Nuclear Plant FSAR, Amendment 64, Section 3.7.3.8, contains analytical procedures for piping other than NSSS. Section 3.7.3.8, General, states that the Code of record for Category I piping systems is the ASME B&PV Code, Section III, Division 1, 1971 Edition up to and including the Summer of 1973 Addenda. As permitted by NA-1140 for utilizing more recent editions of the Code, the FSAR states that the 1980 Edition up to and including the Winter 1982 Edition may be used for the stress qualification of Class 1 piping. TVA responded that they utilized the criteria in Appendix F, Rules for Evaluation of Service Loadings with Level D Service Limits, Summer 1982 Addenda to evaluate the analysis results. The feedwater loop is categorized as TVA Safety Class B piping which, according to Table 3.2-2 of the WBNP FSAR, is ASME B&PV Code, Section III, Class 2.

At a meeting held at WBNP on September 24-25, 1991, TVA presented an overview of its evaluation of the feedwater line for the check valve slam event and stated that calculated stresses will be compared to maximum allowable values as specified in ASME Appendix F. In a letter to the TVA, the NRC stated that the use of ASME Code Appendix F limits is considered appropriate for the evaluation of the check valve slam event (Reference 3). BNL reviewed the various Code editions and found that the requirements in Appendix F, F-1341.2, have not changed, i.e., they are the same in the 1980 and 1993 editions.

BNL finds that the criteria used by TVA to evaluate the feedwater piping is appropriate and conservative because Class 1 criteria was used to evaluate a Class 2 system. BNL also finds that TVA used standard and acceptable methods to obtain the magnitude and time of the maximum stresses from the ANSYS postprocessing files and to classify the stresses.

4.0 CONCLUSIONS

Based on our review and evaluation of the information provided by TVA as discussed in this TER, BNL concludes that TVA has adequately demonstrated by conducting a nonlinear ANSYS analysis that the subject feedwater loops meet the selected ASME B&PV Code criteria when subjected to the dynamic loads resulting from a check valve slam.

**TECHNICAL EVALUATION REPORT FOR
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REFERENCES:

1. "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2, Docket Nos. 50-390 and 50-391," NUREG-0847, Supplement No. 6, April 1991.
2. Letter from J. Garrity (TVA) to NRC, "Responses to Additional NRC Questions on ...Feedwater Check Valve Slam Analysis," August 22, 1991, T04-910822-817.
3. Letter from P. Tam (NRC) to D. Nauman (TVA), "Site Review of Outstanding Issue 20(a), Feedwater Check Valve Slam (TAC Nos. 79718 and 80345)," October 11, 1991.
4. Letter from W. Museler (TVA) to NRC, "Completion of Feedwater Check Valve Slam Analysis," August 4, 1992, T04-920804-966.
5. Letter from P. Tam (NRC) to M. Medford (TVA), "Request for Additional Information on Outstanding Issue 20(a)," November 20, 1992.
6. Letter from W. Museler (TVA) to NRC, "Response to Request for Additional Information (RAI) - Outstanding Issue 20(a) - Feedwater Check Valve Slam - TAC M79718 and M80345," January 11, 1993.
7. USNRC Office of Nuclear Reactor Regulation, Standard Review Plan, Section 3.7.1, Seismic Design Parameters, NUREG-0800, Revision 1, July 1981.
8. USNRC Office of Nuclear Reactor Regulation, Standard Review Plan, Section 3.7.2, Seismic System Analysis, NUREG-0800, Revision 1, July 1981.
9. Letter from W. Museler (TVA) to NRC, "Response to NRC Request for Additional Information - Feedwater Check Valve Slam Whip Restraint Contact Times - TAC M79718 and M80345," April 15, 1993.

**TECHNICAL EVALUATION REPORT FOR
WATTS BAR FEEDWATER CHECK VALVE SLAM ANALYSIS**

ATTACHMENT 1

QUALITY ASSURANCE PROGRAM AT SWANSON SYSTEMS (SASI)
FOR THE ANSYS FINITE ELEMENT COMPUTER PROGRAM

Swanson Analysis Systems, Inc.

Johnson Road, P.O. Box 65, Houston, PA 15342-0065



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March 15, 1991

Mr. Joseph Braverman
Brookhaven National Laboratory
Bldg. 475C
Upton, NY 11973

Dear Mr. Braverman:

This letter is in response to your request for information regarding the nature and scope of our QA program's verification activities. While it is very hard to do justice to a QA program such as our own in the text of a short letter, I hope the following information meets your need.

The ANSYS program is developed, maintained, tested and supported under the SASI Quality Assurance program, which is set up to meet the requirements of The United States Nuclear Regulatory Commission, Rules and Regulations, Title 10, Chapter 1, Code of Federal Regulations, Part 50, Appendix B. The error-reporting system which is part of the QA program is designed to address the requirements of Title 10, Chapter 1, Code of Federal Regulations, Part 21, titles "Reporting of Defects and Noncompliances" for commercial grade items such as ANSYS®.

Verification of ANSYS results is accomplished through formal procedures which require comparison with theoretical calculations or alternate numerical methods, not just cursory checks to see if results look "reasonable". Calculations are formally reviewed and testing results are maintained in a controlled fashion. Although the QA testing library is a tool used by the Development Department in producing and testing new features, the QA Department controls changes to this library.

The verification test set consisted of over 3500 individual test cases at Rev. 4.4A. Over 1000 of these tests are used in acceptance testing for all versions of the program (each separate operating system and compiler) before we release any software for production use. A small subset (approx. 200 problems) are formally published by SASI in the form of the ANSYS Verification Manual. This manual is available for purchase through local ASD's or directly through SASI (contact Charlotte Vargo).

Design controls take the form of review and sign off at key points in the design process. Extensive testing of all documented new features is required. Design reviews are conducted before and after programming activity is performed. Traceable documentation of the design and testing process is used to control new additions to the program. Complete source code listings, testing results and documentation are stored on microfiche allowing for re-creation of any archived ANSYS version.

Mr. Joseph Braverman
Brookhaven National Laboratory
Page Two
March 15, 1991

The formal verification process described has been in place since 1983, when the ANSYS program was at Revision 4.1C. All ANSYS versions since that time, on all supported platforms, such as the VAX 11/780 or VAX 3600 running under VMS, have been subject to formal verification as outlined above.

As we discussed, I am sending you more detailed information on our QA program and testing services under separate cover.

Sincerely yours,

SWANSON ANALYSIS SYSTEMS, INC.



Mark C. Imgrund, Manager
Quality Assurance Department

bp



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Abstract

Quality Assurance at Swanson Analysis Systems, Inc.

Quality and customer service are SASI's two most important corporate values. Our high commitment to quality is reflected throughout our corporate environment. The SASI quality assurance program is a part of that commitment to high quality products and services for our customers.

Since 1983, the responsibility for all quality assurance activities at SASI has been assigned to a separate quality assurance department. The QA Department is autonomous and organizationally independent from other departments which are responsible for the design, documentation, distribution, and support of ANSYS. It reports directly to the President of SASI.

Since the formation of the QA department, the primary functions have been:

1. Verifying existing program capabilities through independent testing.
2. Defining and controlling activities affecting ANSYS quality, such as:
 - Quality Assurance Manual
 - Technical Procedures Manual
 - Acceptance test requirements for product release
 - Design controls for future revisions
 - Traceability of the design and testing process
3. Performing studies concerning improvements to the quality of ANSYS.
4. Verifying and validating new program features.
5. Training new employees in quality-related responsibilities.
6. Conducting and participating in internal and external audits.
7. Performing benchmarks of program performance.
8. Verifying and reviewing ANSYS error fixes.
9. Notifying users of Class3 errors.

Verification of ANSYS results is accomplished by comparison with theoretical calculations or alternate numerical methods, not just cursory checks to see if results look "reasonable". Calculations are formally reviewed and testing results are maintained in a controlled fashion. Although the QA testing library is a tool used by the Development department in producing and testing new features, the QA department controls changes to this library. Design controls take the form of review and sign off at key points in the design process. Extensive testing of all documented new features is required.

Design reviews are conducted before and after programming activity is performed. Traceable documentation of the design and testing process is used to control new additions to the program. Complete source code listings, testing results and documentation are stored on microfiche allowing for re-creation of any archived ANSYS version.

The entire set of quality assurance activities are defined in the ANSYS QA Manual and ANSYS Technical Procedures Manual. The QA Manual is a controlled document issued to SASI employees and licensees who opt for a formal quality assurance agreement with SASI. The Technical Procedures Manual is an internal SASI document detailing specific employee requirements and responsibilities regarding quality-related activities. This document is not available for external distribution. All employees have been trained in their QA-related activities and are re-trained as necessary when their responsibilities or the technical procedures change.

BIBLIOGRAPHIC DATA SHEET

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10. SUPPLEMENTARY NOTES

Docket Nos. 50-390 and 50-391

11. ABSTRACT (200 words or less)

Supplement No. 13 to the Safety Evaluation Report for the application filed by the Tennessee Valley Authority for license to operate Watts Bar Nuclear Plant, Units 1 and 2, Docket Nos. 50-390 and 50-391, located in Rhea County, Tennessee, has been prepared by the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission. The purpose of this supplement is to update the Safety Evaluation with (1) additional information submitted by the applicant since Supplement No. 12 was issued, and (2) matters that the staff had under review when Supplement No. 12 was issued.

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Safety Evaluation Report (SER)
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Dockets 50-390/50-391

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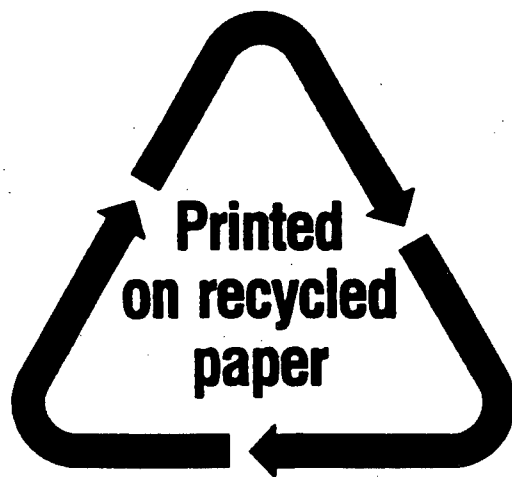
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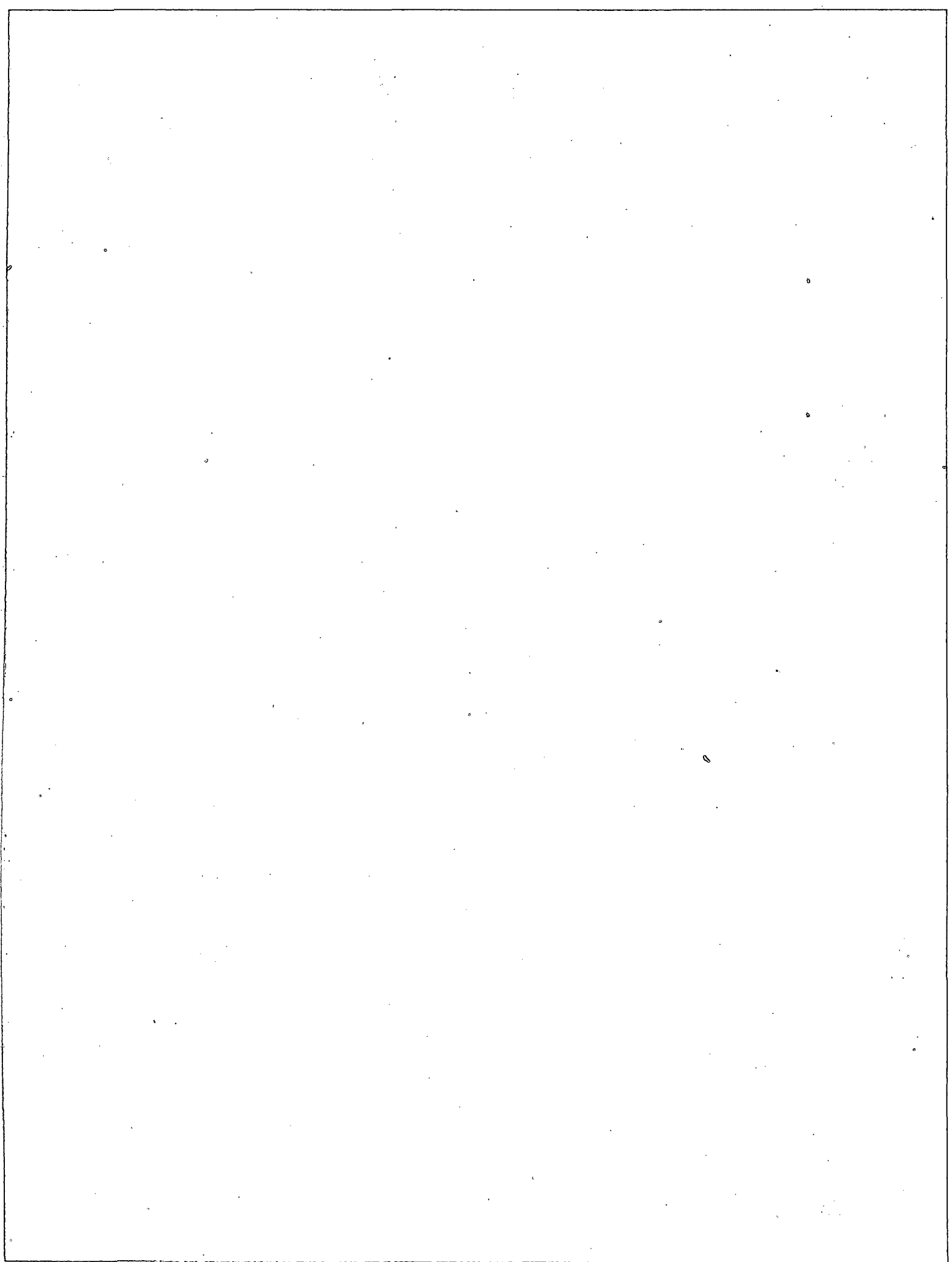
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