
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

September 1995



AVAILABILITY NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 2120 L Street, NW., Lower Level, Washington, DC 20555-0001
2. The Superintendent of Documents, U.S. Government Printing Office, P. O. Box 37082, Washington, DC 20402-9328
3. The National Technical Information Service, Springfield, VA 22161-0002

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC bulletins, circulars, information notices, inspection and investigation notices; licensee event reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the Government Printing Office: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, international agreement reports, grantee reports, and NRC booklets and brochures. Also available are regulatory guides, NRC regulations in the *Code of Federal Regulations*, and *Nuclear Regulatory Commission Issuances*.

Documents available from the National Technical Information Service include NUREG-series reports and technical reports prepared by other Federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, and transactions. *Federal Register* notices, Federal and State legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Office of Administration, Distribution and Mail Services Section, U.S. Nuclear Regulatory Commission, Washington DC 20555-0001.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, Two White Flint North, 11545 Rockville Pike, Rockville, MD 20852-2738, for use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018-3308.

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

September 1995



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), Supplement No. 12 (October 1993), Supplement No. 13 (April 1994), Supplement No. 14 (December 1994), and Supplement No. 15 (June 1995) issued by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission with respect to the application filed by the Tennessee Valley Authority, as applicant and owner, for licenses to operate the Watts Bar Nuclear Plant, Units 1 and 2 (Docket Nos. 50-390 and 50-391). The facility is located in Rhea County, Tennessee, near the Watts Bar Dam on the Tennessee River. This supplement provides recent information regarding resolution of some of the outstanding and confirmatory items, and proposed license conditions identified in the SER.

CONTENTS

	<u>Page</u>
ABSTRACT	iii
ABBREVIATIONS	viii
 1 INTRODUCTION AND DISCUSSION	 1-1
1.1 Introduction	1-1
1.7 Summary of Outstanding Issues	1-2
1.8 Summary of Confirmatory Issues	1-4
1.9 Summary of Proposed License Conditions	1-6
1.12 Approved Technical Issues for Incorporation in the License as Exemptions	 1-9
1.13 Implementation of Corrective Action Programs and Special Programs	 1-10
1.13.1 Corrective Action Programs	1-10
1.13.2 Special Programs	1-18
 3 DESIGN CRITERIA-STRUCTURE, COMPONENTS, EQUIPMENT, AND SYSTEMS	 3-1
3.7 Seismic Design	3-1
3.7.1 Seismic Input	3-1
3.7.1.1 Ground Response Spectra	3-1
3.7.2 Seismic Analysis	3-2
3.8 Design of Seismic Category I Structures	3-3
3.8.3 Other Seismic Category I Structures	3-3
 4 REACTOR	 4-1
4.4 Thermal-Hydraulic Design	4-1
4.4.3 Thermal-Hydraulic Design Methodology	4-1
4.4.3.4 Reactor Coolant System Temperature Measurement	 4-1
4.4.5 Loose Parts Monitoring System	4-1
 5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS	 5-1
5.2 Integrity of Reactor Coolant Pressure Boundary	5-1
5.2.4 Reactor Coolant Pressure Boundary Inservice Inspection and Testing	 5-1
5.2.6 Reactor Vessel and Internals Modeling*	5-1
5.3 Reactor Vessel	5-2
5.3.2 Pressure-Temperature Limits	5-2
 6 ENGINEERED SAFETY FEATURES	 6-1
6.4 Control Room Habitability	6-1
 7 INSTRUMENTATION AND CONTROL	 7-1
7.1 Introduction	7-1
 8 ELECTRICAL POWER SYSTEMS	 8-1
8.3 Onsite Electric Power System	8-1
8.3.3 Common Electrical Features and Requirements	8-1

	<u>Page</u>
8.3.3.1 Compliance With GDCs 2 and 4	8-1
8.3.3.1.6 Cable Damage Near Splices and Terminations	8-1
8.3.3.3 Physical Independence (Compliance With GDC 17)	8-2
8.3.3.5 Compliance With GDC 18	8-4
8.3.3.5.3 Time Constraint for Stability of EDG During No-Load Startup Testing	8-4
9 AUXILIARY SYSTEMS	9-1
9.1 Fuel Storage Facility	9-1
9.1.2 Spent Fuel Storage	9-1
9.3 Process Auxiliaries	9-2
9.3.2 Process Sampling System	9-2
9.4 Heating, Ventilation, and Air Conditioning Systems	9-3
9.4.5 Engineered Safety Features Ventilation System	9-3
11 RADIOACTIVE WASTE MANAGEMENT	11-1
11.1 Summary Description	11-1
11.2 Liquid Waste Management	11-3
11.2.1 System Description and Review Discussion	11-3
11.2.2 Conclusion	11-14
11.3 Gaseous Waste Management	11-14
11.3.1 System Description and Review Discussion	11-14
11.3.2 Conclusion	11-20
11.4 Solid Waste Management	11-20
11.4.1 System Description and Review Discussion	11-20
11.4.2 Conclusion	11-25
11.5 Process and Effluent Radiological Monitoring and Sampling System	11-25
11.5.1 System Description and Review Discussion	11-25
11.5.2 Conclusion	11-29
11.6 Evaluation Findings	11-29
13 CONDUCT OF OPERATIONS	13-1
13.1 Organizational Structure of Applicant	13-1
14 INITIAL TEST PROGRAM	14-1
14.2 Preoperational Tests	14-1
14.2.3 Conclusion	14-2

APPENDICES

A	CHRONOLOGY OF RADIOLOGICAL REVIEW OF WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2, OPERATING LICENSE REVIEW
E	PRINCIPAL CONTRIBUTORS
EE	DISPOSITION OF ALL GENERIC SAFETY ISSUES APPLICABLE TO WATTS BAR

TABLES

	<u>Page</u>
6.1 Assumptions Used for Calculating Watts Bar Control Room Habitability Parameters	6-2
6.2 Control Room Personnel Doses	6-2
11.1 Calculated Docket RM 50-2, Dose Commitments to a Maximally Exposed Individual and Releases for Watts Bar Units 1 and 2	11-2
11.2 Principal Parameters Used in the Calculation of Liquid and Gaseous Effluents for Watts Bar, Units 1 and 2	11-8
11.3 Calculated Releases of Radioactive Materials in Liquid Effluents From Watts Bar, Units 1 and 2	11-10
11.4 Calculated Appendix I Dose Commitments to a Maximally Exposed Individual for Watts Bar Units 1 and 2	11-12
11.5 Staff's Calculated Releases of Radioactive Materials in Gaseous Effluents From Watts Bar Units 1 and 2	11-13
11.6 Summary of Atmospheric Dispersion Factors (χ/Q) and Relative Deposition Values for Maximum Site Boundary and Receptor Locations Near Watts Bar Units 1 and 2	11-19

ABBREVIATIONS

ALARA	as low as is reasonably achievable
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
B&O	Bulletins and Orders
BTP	branch technical position
B&W	Babcock and Wilcox
CAP	corrective action program
CCW	component cooling water
CDWE	condensate demineralization waste evaporator
CFR	<i>Code of Federal Regulations</i>
CNPP	Corporate Nuclear Performance Plan
CSST	common station service transformer
CVCS	chemical and volume control system
DAW	dry active waste
DCRDR	detailed control room design review
DF	decontamination factor
EDG	emergency diesel generator
EPIP	emergency plan implementing procedure
ERCW	essential raw cooling water
ESF	emergency safety feature
FDCT	floor drain collector tank
FMU	flow measurement uncertainty
FSAR	final safety analysis report
GDC	general design criterion
GL	generic letter
GRPS	gaseous radwaste processing system
GSI	generic safety issue
GWMS	gaseous waste management system
HCLC	high-crud low-conductivity
HEPA	high-efficiency particulate air
HIC	higher intensity container
ICS	internal concrete structure
IE	Office of Inspection and Enforcement
IEEE	Institute of Electrical and Electronics Engineers
IPEEE	individual plant examination external event
ITP	Initial Test Program
LCHC	low-crud high-conductivity
LLD	lower limit of detection
LOCA	loss-of-coolant accident
LPMS	loose parts monitoring system
LWMS	liquid waste management system

LWR	light-water reactor
MPA	multiplant action
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
NUDOCS	NRC document control system
OBE	operating basis earthquake
ODCM	Offsite Dose Calculation Manual
OL	operating license
PASS	postaccident sampling system
PCC	primary coolant concentration
PCP	Process Control Program
PGA	peak ground acceleration
PWR	pressurized-water reactor
QA	quality assurance
RAI	request for additional information
RCDT	reactor coolant drain tank
RCL	reactor coolant loop
RCP	reactor coolant pump
RCS	reactor coolant system
RG	regulatory guide
RHR	residual heat removal
RMS	radiation monitoring system
RO	reactor operator
RPV	reactor pressure vessel
RTD	resistance temperature detector
SBLOCA	small-break loss-of-coolant accident
SER	safety evaluation report
SG	steam generator
SNP	Sequoyah Nuclear Plant
SP	special program
SPDS	safety parameter display system
SR	surveillance requirement
SRO	senior reactor operator
SRP	standard review plan
SRST	spent resin storage tank
SSE	safe shutdown earthquake
SSER	supplement to safety evaluation report
SSI	soil-structure interaction
SSRS	site-specific response spectrum
STA	shift technical advisor
SWMS	solid waste management system
TAC	technical assignment control
TDCT	tritiated drain collector tank
TI	temporary instruction
TMI	Three Mile Island
TRM	Technical Requirements Manual
TSS	Technical Specifications

TVA	Tennessee Valley Authority
USI	unresolved safety issue
WBNPP	Watts Bar Nuclear Performance Plan
WISP	Workload Information and Scheduling Program
WOG	Westinghouse Owners Group

1 INTRODUCTION AND DISCUSSION

1.1 Introduction

In June 1982, the Nuclear Regulatory Commission staff (NRC staff or staff) issued a Safety Evaluation Report, NUREG-0847, regarding the application by the Tennessee Valley Authority (TVA or the applicant) for licenses to operate the Watts Bar Nuclear Plant, Units 1 and 2. The Safety Evaluation Report (SER) was followed by SER Supplement No. 1 (SSER 1, September 1982), Supplement No. 2 (SSER 2, January 1984), Supplement No. 3 (SSER 3, January 1985), Supplement No. 4 (SSER 4, March 1985), Supplement No. 5 (SSER 5, November 1990), Supplement No. 6 (SSER 6, April 1991), Supplement No. 7 (SSER 7, September 1991), Supplement No. 8 (SSER 8, January 1992), Supplement No. 9 (SSER 9, June 1992), Supplement No. 10 (SSER 10, October 1992), Supplement No. 11 (SSER 11, April 1993), Supplement No. 12 (October 1993), Supplement No. 13 (SSER 13, April 1994), Supplement No. 14 (SSER 14, December 1994), and Supplement No. 15 (SSER 15, June 1995). As of this date, the staff has completed its review of the applicant's Final Safety Analysis Report (FSAR) up to Amendment 89.

The SER and its supplements were written to agree with the format and scope outlined in the Standard Review Plan (SRP, NUREG-0800). Issues raised by the SRP review that were not closed out when 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 in the SRP, the staff issues generic requirements or recommendations in the form of technical reports, bulletins, and generic letters. Each of these documents carries its own applicability, work scope, and acceptance criteria; some are applicable to Watts Bar. The review and implementation status of applicable generic issues are addressed in Appendix EE of this supplement.

Each of the following sections and 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 continues the chronology of the safety review. Appendix E lists principal contributors to this supplement. Appendix EE is added in this supplement, as stated above. The other appendices are not changed by this supplement.

The Project Manager is Peter S. Tam. Mr. Tam may be contacted by calling (301) 415-7000, or by writing to the following address:

Mr. Peter S. Tam
Mail Stop 0-14B21
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

1.7 Summary of Outstanding Issues

In SER Section 1.7, the staff listed 17 outstanding issues (open items) that had not been resolved at the time the SER was issued. Additional outstanding issues were added in SER supplements that followed. In this section, the staff 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 licensing status meetings.

<u>Issue</u> ¹	<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)	Resolved (SSER 14)	3.9.6
(4) Qualification of equipment (a) Seismic (TAC M71919) (b) Environmental (TAC M63591)	Resolved (SSER 9) Resolved (SSER 15)	3.10 3.11
(5) Preservice inspection program (TAC M63627)	Resolved for Unit 1 (SSERs 10 and 12)	5.2.4, 6.6, App. Z
(6) Pressure-temperature limits for Unit 2 only	On hold (SER)	5.3.2, 5.3.3
(7) Model D-3 steam generator preheater tube degradation	Resolved (SSER 4)	5.4.2.2
(8) Branch Technical Position CSB 6-4	Resolved (SSER 3)	6.2.4
(9) H ₂ analysis review	Resolved (SSER 4)	6.2.5
(10) Safety valve sizing analysis (WCAP-7769)	Resolved (SSER 2)	5.2.2
(11) Compliance of proposed design change to the offsite power system to GDCs 17 and 18 (TAC M63649)	Resolved (SSER 13)	8.2
(12) Fire-protection program (TAC M63648)	Under review (SER)	9.5.1

¹The TAC (technical assignment control) numbers that appear in parentheses after some issue titles and elsewhere in this document, are internal NRC control numbers 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>
(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)	Resolved (SSER 15)	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 (TACs M79717, M80346):		
(a) Number of OBE events	Resolved (SSER 8)	3.7.3
(b) 1.2 multi-mode factor	Resolved (SSER 9)	3.7.3
(c) Code usage	Resolved (SSER 8)	3.7.3
(d) Conduit damping values	Resolved (SSER 8)	3.7.3
(e) Worst case, critical case, bounding calculations	Resolved (SSER 12)	3.7.3
(f) Mass eccentricities	Resolved (SSER 8)	3.7.2.1.2
(g) Comparison of set A versus set B response	Resolved (SSER 11)	3.7.2.12
(h) Category 1(L) piping qualification	Resolved (SSER 8)	3.9.3
(i) Pressure relief devices	Resolved (SSER 7)	3.9.3.3
(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
(23) Containment isolation using closed systems (TAC M63597)	Resolved (SSER 12)	6.2.4
(24) Main steamline break outside containment (TAC M63632)	Resolved (SSER 14)	3.6.1

<u>Issue</u>	<u>Status</u>	<u>Section</u>
(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

In SER Section 1.8, the staff listed 42 confirmatory issues for which additional information and documentation were required to confirm preliminary conclusions. Issue 43 was added in SSER 6. In this section, the staff 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.

<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
(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

<u>Issue</u>	<u>Status</u>	<u>Section</u>
(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 feed-water lines and feedwater bypass lines (TAC M63623)	Resolved (SSER 5)	6.2.4
(19) Compliance with GDC 51	Resolved (SSER 4)	6.2.7, App. H
(20) Insulation survey (sump debris)	Resolved (SSER 2)	6.3.3
(21) Safety system setpoint methodology	Resolved (SSER 4)	7.1.3.1
(22) Steam generator water level reference leg	Resolved (SSER 2)	7.2.5.9
(23) Containment sump level measurement	Resolved (SSER 2)	7.3.2
(24) IE Bulletin 80-06	Resolved (SSER 3)	7.3.5
(25) Overpressure protection during low-temperature operation	Resolved (SSER 4)	7.6.5
(26) Availability of offsite circuits	Resolved (SSER 2)	8.2.2.1
(27) Non-safety loads powered from the Class 1E ac distribution system	Resolved (SSER 2)	8.3.1.1
(28) Low and/or degraded grid voltage condition (TAC M63649)	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

<u>Issue</u>	<u>Status</u>	<u>Section</u>
(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)	Resolved (SSER 15)	18.2

1.9 Summary of Proposed License Conditions

In Section 1.9 of the SER and in SSERs that followed, the staff listed 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 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

<u>Proposed Condition</u>	<u>Status</u>	<u>Section</u>
(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 (TACs 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
(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

<u>Proposed Condition</u>	<u>Status</u>	<u>Section</u>
(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)	Resolved (SSER 14)	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
(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

<u>Proposed Condition</u>	<u>Status</u>	<u>Section</u>
(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)	Resolved (SSER 15)	18.1
(38) Physical security of fuel in containment (TACs 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)	Resolved (SSER 14)	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
(44) Physical Security Plan (TAC M63657, M83973)	Opened (SSER 15)	13.6

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) Schedule to implement the vehicle bomb rule (Section 13.6.9, SSER 15) (TAC M90696)

In addition to these, the staff granted an exemption to the applicant on December 15, 1994, which will also be incorporated in the operating license:

(4) Issuance, storage, and retrieval of badges for personnel (TAC M90729)

The staff reevaluated three technical issues previously approved for exemption from various provisions of Appendix G to 10 CFR Part 50 in SSER 14. As a result, Section 5.3.1.1 of SSER 14 reports that these exemptions are no longer needed.

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 October 1995.
------------------------	---

NRC inspections:

Inspection Reports 50-390, 391/90-09 (June 22, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/90-22 (November 21, 1990); 50-390, 391/90-24 (December 17, 1990); 50-390, 391/90-27 (December 20, 1990); 50-390, 391/90-30 (February 25, 1991); 50-390, 391/91-07 (May 31, 1991); 50-390, 391/91-09 (July 15, 1991); 50-390, 391/91-12 (July 12, 1991); 50-390, 391/91-31 (January 13, 1992); 50-390, 391/92-01 (March 17, 1992); audit report of June 12, 1992 (Appendix Y of SSER 9); 50-390, 391/92-05 (April 17, 1992); 50-390, 391/92-13 (July 16, 1992); 50-390, 391/92-18 (August 14, 1992); 50-390, 391/92-22 (September 18, 1992); 50-390, 391/92-26 (October 16, 1992); 50-390, 391/92-30 (November 13, 1992); 50-390, 391/92-35 (December 15, 1992); 50-390, 391/92-40 (January 15, 1993); 50-390, 391/93-10 (March 19, 1993); 50-390, 391/93-11 (March 25, 1993); 50-390, 391/93-35 (June 10, 1993); 50-390, 391/93-40 (July 15, 1993); 50-390, 391/93-48 (August 13, 1993); 50-390, 391/93-56 (September 20, 1993); 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); 50-390, 391/94-18 (April 18, 1994); 50-390, 391/94-32 (May 16, 1994); 50-390, 391/94-35 (June 20, 1994); 50-390, 391/94-45 (July 15, 1994); 50-390, 391/94-51 (August 11, 1994); 50-390, 391/94-53 (September 20, 1994); 50-390, 391/94-55 (September 16, 1994); 50-390, 391/94-61 (October 12, 1994); 50-390, 391/94-66 (November 16, 1994); 50-390, 391/94-75 (December 19, 1994); 50-390, 391/94-82 (January 13, 1995); 50-390, 391/94-88 (February 15, 1995); 50-390, 391/95-17 (April 13, 1995); 50-390, 391/95-45 (August 15, 1995); 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 September 1995.

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); 50-390/94-64 (December 15, 1994); 50-390, 391/94-88 (February 15, 1995); 50-390, 391/95-23 (May 2, 1995); 50-390, 391/95-27 (May 31, 1995); 50-390, 391/95-35 (June 28, 1995); 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; Inspection Report 50-390/95-36 (June 21, 1995).

Implementation status: 100%

NRC inspections: Complete: 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); 50390, 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); 50-390, 391/94-69 (November 18, 1994); 50-390/95-36 (June 21, 1995); 50-390, 391/95-47 (August 16, 1995).

(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 September 1995.

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); 50-390, 391/94-32 (May 16, 1994); 50-390/94-64 (December 15, 1994); 50-390, 391/94-82 (January 13, 1995); 50-390, 391/94-88 (February 15, 1995); 50-390, 391/95-23 (May 2, 1995); 50-390, 391/95-27 (May 31, 1995); 50-390, 391/95-35 (June 28, 1995); 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 September 1995.

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); 50-390, 391/94-18 (April 18, 1994); 50-390, 391/94-31 (May 11, 1994); 50-390, 391/94-45 (July 15, 1994); 50-390, 391/94-53 (September 20, 1994); 50-390, 391/94-66 (November 16, 1994); 50-390, 391/94-82 (January 13, 1995); 50-390, 391/94-88 (February 15, 1995); 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: 100%

NRC inspections: Complete: 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); 50-390, 391/95-30 (June 22, 1995); 50-390, 391/95-55 (August 28, 1995).

(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 results to be published in SSER 18.

Implementation status: 100%

NRC inspections: Complete: Inspection Reports 50-390, 391/94-45 (July 15, 1994); 50-390, 391/94-63 (November 2, 1994); 50-390, 391/94-62 (November 16, 1994); 50-390, 391/94-66 (November 16, 1994); 50-390, 391/94-78 (December 21, 1994); 50-390, 391/94-82 (January 13, 1995); 50-390, 391/95-03 (January 31, 1995); 50-390, 391/95-13 (March 1, 1995); 50-390, 391/95-16 (April 6, 1995); 50-390, 391/95-26 (May 1, 1995); 50-390, 391/95-32 (June 9, 1995); 50-390, 391/95-39 and 95-40 (July 18, 1995); 50-390, 391/95-61 (September 12, 1995).

(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: 100%

NRC inspections: Complete: 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); 50-390, 391/94-32 (May 16, 1994); 50-390, 391/94-55 (September 16, 1994); 50-390, 391/95-06 (March 16, 1995); 50-390, 391/95-23 (May 2, 1995); 50-390, 391/95-27 (May 31, 1995); 50-390, 391/95-35 (June 28, 1995); 50-390, 391/95-53 (September 8, 1995).

(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: 100%

NRC inspections: Complete: 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); 50-390, 391/95-23 (May 2, 1995); 50-390, 391/95-35 (June 28, 1995); 50-390, 391/95-46 (August 1, 1995).

(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; Appendix K of SSER 6; letter, P. S. Tam (NRC) to O. D. Kingsley (TVA), May 5, 1994.

Implementation status: 100%.

NRC inspections: Complete: Inspection Reports 50-390, 391/90-14 (August 3, 1990); 50-390, 391/90-23 (November 19, 1990); 50-390, 391/90-29 (January 29, 1991); 50390, 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); 50-390, 391/94-24 (July 1, 1994); 50-390, 391/94-32 (May 16, 1994); 50-390, 391/94-55 (September 16, 1994); 50-390, 391/95-23 (May 2, 1995); 50-390, 391/95-27 (May 31, 1995); 50-390, 391/95-35 (June 28, 1995); 50-390, 391/95-53 and 95-61 (September 8, 1995).

(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: 100% (certified by letter, W. J. Museler (TVA), to NRC, April 27, 1994); staff concurrence in Inspection Report 50-390, 391/94-40 (June 24, 1994).

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

1990); 50390, 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); 50-390, 391/94-28 (May 5, 1994); 50-390, 391/94-40 (June 24, 1994).

(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 (TVA), 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, April 5, 1994, and February 6, 1995.

Implementation status: 100%

NRC inspections: Complete: 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); 50-390/94-201 (December 14, 1994); 50-390, 391/95-34 (June 23, 1995); 50-390, 391/95-50 (August 29, 1995).

(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: 100%. 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 September 1995.

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); 50-390, 391/95-10 (March 17, 1995); 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; these inspection reports also address recurrence control: 50-390, 391/93-02 (February 2, 1993); 50-390, 391/93-84 (December 21, 1993); 50-390, 391/94-79 (January 11, 1995).

Implementation status: 100% (certified by letter, W. J. Museler (TVA) to NRC, January 9, 1993); staff concurrence in Inspection Report 50-390, 391/94-79 (January 11, 1995).

NRC inspections: Complete: 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); 50-390, 391/94-49 (July 21, 1994); 50-390, 391/94-79 (January 11, 1995).

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 (TVA) to NRC, December 30, 1993); staff concurrence in Inspection Report 50-390, 391/95-38 (July 11, 1995).

NRC inspections: Complete: Inspection Report 50-390, 391/93-56 (September 20, 1993); 50-390, 391/95-38 (July 11, 1995).

(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; Section 18.1 of SSER 5 and 15.

Implementation status: 100%

NRC inspections: Complete: Inspection Reports 50-390, 391/94-22 (April 28, 1994); audit reports in SSER 5 and 15.

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

Program review status: Complete: NUREG-1232, Vol. 4; Section 3.11 of SSER 15.

Implementation status: 100%

NRC inspections: Complete: Inspection Reports 50-390, 391/93-63 (October 18, 1993; 50-390, 391/94-28 (April 18, 1994); 50-390, 391/94-74 (January 13, 1995); 50-390, 391/95-15 (April 5, 1995); 50-390, 391/95-54 (September 8, 1995).

(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: Complete: NUREG-1232, Vol. 4; Section 3.11 of SSER 15.

Implementation status: 100%

NRC inspections: Complete: Inspection Reports 50-390, 391/95-15 (April 5, 1995); 50-390, 391/95-54 (September 8, 1995).

(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: 100%

NRC inspections: Complete: Inspection Reports 50-390, 391/93-85 (January 14, 1994); 50-390, 391/95-53 (September 8, 1995); 50-390, 391/95-61 (September 12, 1995).

(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 October 1995.

NRC inspections: Inspection Reports 50-390, 391/94-56 (October 6, 1994); 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).

1.17 Financial Assurance for Decommissioning

By letter dated June 16, 1995, the applicant submitted its statement of intent by its Board of Directors to provide funds when needed for the decommissioning of Watts Bar, Unit 1. The staff reviewed this letter and the attached statement and concludes that TVA, as a Federal Government utility, has complied with NRC decommissioning funding assurance regulations by issuing a statement of intent for providing decommissioning funds for Watts Bar, Unit 1, pursuant to 10 CFR 50.75(e)(3)(iv) in an amount consistent with the formulae specified in 10 CFR 50.75(c).

The staff tracked this review by TAC M91522.

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

The staff reviewed Amendments 79 and 89 of the FSAR. In reviewing Amendment 79, the staff raised two safety issues: (1) nonlinear analysis of the internal concrete structure – nuclear steam supply system (ICS-NSSS) and (2) the use of higher allowable stresses for the operating basis earthquake (OBE). By letters of May 3 and October 11, 1994, the staff requested additional information on the two issues. The applicant responded by letters dated February 3 and August 18, 1995, and by FSAR Amendment 89. The staff's review of these applicant submittals appears below in Sections 3.7.2 and 3.8.3.

3.7 Seismic Design

3.7.1 Seismic Input

3.7.1.1 Ground Response Spectra

In SSER 6, the staff reported the value of the peak ground acceleration (PGA) of the vertical component of the site-specific response spectrum (SSRS) as 0.18 g; in FSAR Sections 2.5 and 3.7.1, the applicant stated it was 0.15 g. The applicant stated in a letter dated July 18, 1995, that all Watts Bar design criteria, calculations, and analysis reports are consistent with the FSAR in the use of the vertical SSRS PGA of 0.15 g. In SSER 6, the staff conveyed its review of FSAR Amendments 54 through 64, and the staff's inspection of the Seismic Analysis Corrective Action Program (CAP) (Inspection Report 50-390, 391/89-21). Inspection Report 50-390, 391/89-21 accepted Watts Bar calculation WCG-1-342 (which derives synthetic time histories for use with the SSRS) and Revision 1 of the Seismic Analysis CAP, both of which state that the vertical PGA for the SSRS (or Set B) is 0.15 g.

FSAR Amendments 57 and 64 were part of the material reviewed by the staff in preparing SSER 6. FSAR Amendment 57 incorporated an "additions" spectrum developed in 1980 (called Criterion B) that was used only for the original design of the additional diesel generator building. Criterion B used Regulatory Guide 1.60 spectral shapes and both vertical and horizontal PGAs of 0.18 g. When the Seismic Analysis CAP was developed, the applicant canceled Criterion B. Then FSAR Amendment 64 removed the Criterion B seismic basis and replaced it with the Set A, Set B, and Set C bases that were described in the Seismic Analysis CAP. Apparently, the vertical PGA of 0.18 g for the SSRS in SSER 6 was erroneously carried over from the review of FSAR Amendment 57.

In order to calculate the SSRS, it is necessary to characterize the size of earthquake, its distance from the epicenter, and the site conditions being modeled, and to collect a set of earthquake recordings that approximate these characteristics. The applicant collected 13 sets of strong-motion records that meet the criteria for the Watts Bar SSRS. The staff accepted this data set in the SER and has no new information that would cause it to change that position.

The appropriate level of conservatism for an SSRS is specified in Standard Review Plan Section 2.5.2.6 as being the 84th percentile of the appropriate data base. The staff reviewed again the vertical component data set that was used to develop the Watts Bar SSRS, and found that the PGA of the 84th percentile of the data is 0.15 g. This confirms the applicant's position that the correct PGA for the vertical component of the Watts Bar SSRS is 0.15 g.

The staff tracked its review by TAC M92973.

3.7.2 Seismic Analysis

In FSAR Section 3.7, the applicant described the seismic model of the ICS-NSSS for the purpose of assessing the plant structure seismic response. The applicant stated in the FSAR that the NSSS supports in the coupled model exhibit nonlinear behavior because of the gaps and tension-only tie rods at the NSSS and ICS interfaces. In Question 4 of its May 3, 1994, letter, the staff asked the applicant to submit the detailed methodology used in developing the dynamic models of the NSSS supports utilized in the coupled seismic model. In its response of August 18, 1994, the applicant stated that the integration of the NSSS and the ICS model was previously reviewed by the staff and found to be acceptable. In addition, the applicant presented the details which documented the modeling of the NSSS and the ICS. However, the staff contended, in its October 11, 1994, letter, that the applicant had not provided detailed information on converting a nonlinear structure to a linear model and on the validity of the conversion and, therefore, the staff concluded that the applicant had not provided the information requested by the staff. The staff also reviewed the inspection reports cited in the applicant's August 18, 1994, letter, and concluded that the inspector had not reviewed the approximation of the nonlinear NSSS model to an equivalent linear model and the associated boundary conditions. In its October 11, 1994, letter, the staff concluded that the applicant's August 18, 1994, response was insufficient and asked the applicant to submit further details on the combined ICS and NSSS models.

By letter dated February 3, 1995, the applicant sent additional information about the structural model, such as the geometry and the resulting forces and moments at the ICS-NSSS interface. In Question 4 of its May 3, 1994, letter, the staff had requested a detailed description of the linearized NSSS support stiffnesses that were used for the ICS-NSSS analyses and a discussion of how these representations of the supports adequately model the nonlinear system evaluated. The applicant stated that Westinghouse had investigated different NSSS analysis cases and had identified the supports or tie rods that will be activated under a specific loading condition. The applicant also stated that Westinghouse had developed an active support list table which the applicant had used in performing the seismic analysis. The applicant concluded that since only a specific set of NSSS supports with their specified orientation are activated for each different loading condition, a linear support stiffness can be developed.

In order to discuss and resolve this issue, the staff conducted an onsite review of the structural seismic modeling and analysis of Watts Bar Unit 1 coupled ICS-NSSS model on June 20, 1995 (report issued by letter, P. S. Tam (NRC) to O. D. Kingsley (TVA), July 19, 1995). The staff reviewed the NSSS model development calculation and the modeling of nonlinear NSSS supports in linear analyses to verify that the criteria documented in the FSAR are

properly implemented in the analysis. In addition, the staff conducted a walkdown of Watts Bar Unit 1 to ensure that the representative NSSS supports observed during the walkdown conform to the configuration modeled in the seismic analysis.

During the June 20, 1995, review, the applicant and Westinghouse presented the detailed construction configuration and modeling methodology for the NSSS supports. The NSSS components included in the coupled model for the ICS consist of the reactor pressure vessel (RPV), four loops of the primary reactor coolant loop (RCL) piping (hot legs, cold legs, and cross-over legs), the steam generator (SG), and the reactor coolant pump (RCP) associated with each loop. The NSSS models for four loops consist of masses and mass moments of inertia lumped at the nodal points of RPV, RCL piping, SG, and RCP, and interconnected with elastic elements. The stiffness properties of the elastic elements are represented by various 12 x 12 generalized stiffness matrices. For the purpose of linear response analyses, four linearized NSSS analyses, each with a unique set of linearized NSSS support stiffness, are used to bound the nonlinear support behavior under various dynamic loading conditions. For each NSSS analysis case, a specific set of NSSS supports with their specified orientation are activated for a particular loading condition, and a set of linear support stiffness is provided to represent the active supports.

The staff concludes that it is common industry practice to construct pipe supports with gaps and that the supports are active only if the gaps close during the earthquake motion. The staff recognizes that linear elastic modeling of the pipe supports is an acceptable methodology as long as the overall analysis adequately models various support conditions. On the basis of presentations of the various cases considered to model the RVP supports, SG upper and lower supports, crossover leg restraints on the SG and RCP sides, RCP lower supports, and RCP tie rods, the staff concludes that the ICS-NSSS coupled model appropriately represents the actual configuration and that the enveloping of the response conservatively represents the seismic response of the ICS-NSSS structural system.

The staff tracked its review by TACs M90549 and M91523.

3.8 Design of Seismic Category I Structures

3.8.3 Other Seismic Category I Structures

The evaluation that follows is based on the staff's review of FSAR Sections 3.8.3, 3.8.4, and 3.8.6.

In its letter of May 3, 1994, the staff raised the issue concerning allowable stresses for Category I structural steel. In particular, the staff noted that the applicant did not specify one allowable stress for the safe-shutdown earthquake (SSE) and another for the OBE. The applicant selected a single allowable material yield stress (S_y) of 0.9 for both the SSE and the OBE. In its letter of October 11, 1994, the staff told the applicant that this is not acceptable. The applicant agreed with the staff in its February 3, 1995, submittal and committed to revise the criteria to reflect the staff's position. The alternate allowable stress criteria submitted are 0.6 S_y for OBE and 0.9 S_y for SSE. These values are acceptable because they are consistent with the staff's guidance in the Standard Review Plan.

The applicant proposed an exception in the revision noted above, in particular, allowable stress criteria in FSAR Tables 3.8.6-1 and 3.8.6-2. The loads, their combinations, and corresponding allowable stress criteria in the tables apply to seismic Category I(L) polar cranes and auxiliary building bridge cranes. The applicant combined the SSE and OBE as extreme environment load and assigned a single allowable that corresponds to the SSE (i.e., $0.9 S_y$ for bending). The applicant stated that an exception should be made to the changes agreed on in the two-tier allowable stress criteria (above) and that the single-tier allowable stress criterion for the crane design should be left alone. The applicant cited several references that contain examples and recommendations for using one allowable stress criterion for both SSE and OBE for the crane design.

In Standard Review Plan Section 9.1.5, "Overhead Heavy Load Handling System," the staff referred to NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants." In NUREG-0554, the staff stated that the crane should be designed and constructed in accordance with Regulatory Position 2 of Regulatory Guide 1.29, "Seismic Design Classification." Regulatory Position 2 notes that components whose continued function is not required, but whose failure could reduce the functioning of any plant feature, should be designed and constructed so that the SSE would not cause such failure. The staff thus concludes that the applicant's proposal for the crane design is acceptable.

The staff tracked its review by TACs M90549 and M91523.

4 REACTOR

4.4 Thermal-Hydraulic Design

4.4.3 Thermal-Hydraulic Design Methodology

4.4.3.4 Reactor Coolant System Temperature Measurement*

In FSAR Amendment 88, the applicant stated that the flow measurement uncertainty (FMU) value for the reactor coolant system (RCS) at Watts Bar was reduced from 1.8 percent to 1.5 percent. This FMU value appeared low compared to other plants. In a letter of March 8, 1995, the staff asked the applicant to justify the FMU value and to include the uncertainty from the elbow tap readings and the 0.1-percent Venturi fouling penalty.

By letter dated July 20, 1995, the applicant responded. The applicant's letter contained four enclosures: Enclosure 1 responded to the three questions in the staff's letter of March 8, 1995; Enclosure 2 presented proposed changes to the Unit 1 Technical Specifications based on the updated FMU analysis described in Enclosure 1; Enclosure 3 is a proprietary topical report WCAP-14419 ("Westinghouse Instrument Uncertainty Methodology for Reactor Coolant System Flow Measurement, Tennessee Valley Authority, Watts Bar," by W. H. Moomau and C. R. Tuley, June 1995); Enclosure 4 is the non-proprietary version of WCAP-14420.

The applicant based the FMU values on WCAP-14419, which contained the analysis to arrive at the FMU values. This analysis was acceptable.

WCAP-14419 gave the following FMU values: 1.6 percent using the process computer and 1.9 percent using the control board indication. To allow for a 0.1-percent feedwater Venturi fouling penalty, these FMU values are increased to 1.7 percent and 2.0 percent, respectively. The corresponding RCS total flow rate values are $\geq 397,000$ gpm (process computer) or $\geq 398,000$ gpm (control board computer). These values are used in the Unit 1 Technical Specifications and are acceptable because the analysis on which they are based is acceptable.

In SSER 8, the staff stated that it will track this effort by TAC 81063; however, the staff tracked its review by TAC M91682.

4.4.5 Loose Parts Monitoring System

By letter dated July 18, 1995, the applicant submitted changes to the loose parts monitoring system (LPMS) and associated commitments documented in the following two references: (1) letter from L. M. Mills (TVA) to E. Adensam (NRC), dated February 25, 1982, and (2) letter from L. M. Mills to E. Adensam, dated November 10, 1982. The staff previously approved the design of the LPMS in the SER and SSER 3. The changes communicated by the July 18, 1995, letter include the removal of the solenoid-operated impact subsystem, and alternate

*Section 4.4.3.4 was introduced into the SER in SSER 8.

methods used to demonstrate periodic channel functional testing and channel calibration during refueling outages due to sensor inaccessibility.

The solenoid-operated impact subsystem was designed to demonstrate channel operability as part of periodic channel functional testing. During the conduct of preoperational testing, the subsystem was found to be ineffective in performing its intended design. The subsystem experienced excessive electromagnetic interference with the sensor output signal, and was unable to detect a solenoid-induced impact.

In response to these problems, the solenoid-operated impact subsystem was removed and a portable system was added to upgrade the performance of the LPMS. The portable system is used as one of the actions needed to demonstrate channel functional test requirements as described in Regulatory Guide (RG) 1.133, Rev. 1, Section C.3.a.2.d.

The portable system collects signal information from each LPMS channel and performs spectral and statistical analyses. The system offers improved techniques in obtaining information that can be used to (1) determine if a channel exhibits proper functional characteristics, (2) distinguish between the sounds made by the impacts of loose parts and normal plant noise, (3) trend the behavior of the channels for comparison purposes, and (4) determine the relative size and mass of a loose part impact.

In addition, RG 1.133, Rev. 1, Section C.3.a.3 specifies the use of a controlled mechanical input at cold shutdown or refueling as part of channel calibration activities. After initial channel calibration and baseline spectral data are obtained, the computer-based analytical system may be used to demonstrate channel calibration as an option to the use of a control mechanical input. This option applies to sensors located in areas where plant personnel radiation exposure is considered to be excessive.

The staff reviewed the applicant's revised commitments in its letter of July 18, 1995, and found that they are in compliance with RG 1.133 and are thus acceptable.

The staff tracked its review by TAC M91523.

5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.4 Reactor Coolant Pressure Boundary Inservice Inspection and Testing

In SSER 10 and 12 the staff authorized a number of alternatives to portions of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section II. Subsequently, by letter of August 9, 1995, the applicant requested additional alternatives from the code. The staff authorized those alternatives by letter dated September 18, 1995; that letter is incorporated into this section by reference.

The staff tracked this effort by TAC M93313.

5.2.6 Reactor Vessel and Internals Modeling*

By FSAR Amendment 89, the applicant revised FSAR Section 5.2.1.10.6.4 to indicate that the reactor pressure vessel (RPV) and a mathematical model of the internal plant components is a three-dimensional nonlinear finite element model which represents the dynamic characteristics of the reactor vessel and its internal components in the six geometric degrees of freedom. The model was developed using the WECAN computer code. The FSAR contains a description of the WECAN computer code which is used to determine the response of the reactor vessel and its internal components. The FSAR also discusses the analytical methodology of the code.

The applicant's earlier analytical model (described in FSAR Amendment 78) consisted of two nonlinear elastic models connected at a common mode: one model represented the dynamic vertical characteristics of the vessel and its internal components, and the other model represented the transitional and rotational characteristics of the structure. These two models were combined in the DARI-WOSTAS code to represent motion of the reactor vessel and its internal components in the plane of the vessel centerline and the broken pipe centerline.

The submodels based on the WECAN computer code and representing the reactor core barrel, neutron panels, lower support plate, tie plates, secondary core support components, upper support plate, guide tubes, core plates, and fuel are discussed in FSAR Amendment 89.

The staff approved the WECAN computer code in other similar applications involving three-dimensional nonlinear finite element analyses and finds its use acceptable in the modeling and analysis of the RPV and internal components for Watts Bar.

The staff tracked its review by TAC M91523.

*In the SER and previous SSERs, there was no section that appeared appropriate to convey the staff's evaluation of FSAR Section 5.2.1.10.6.4. Hence Section 5.2.6 is added in this SSER (SSER 16).

5.3 Reactor Vessel

5.3.2 Pressure-Temperature Limits

By letter dated September 22, 1995, the staff found acceptable the pressure temperature limits methodology and the pressure temperature limits report for Unit 1. The staff's evaluation in that letter supersedes the evaluation in the SER, and will be referenced in the administrative controls section of the Watts Bar Unit 1 Technical Specifications.

The staff tracked its review by TAC M89048.

6 ENGINEERED SAFETY FEATURES

6.4 Control Room Habitability

In the SER and SSER 1, the staff found the control room design acceptable regarding its capability to adequately protect its occupants from radiation under normal and accident conditions. The staff stated that the control design met the requirements of General Design Criterion (GDC) 19 of 10 CFR Part 50. Subsequently, the applicant changed some design parameters and updated the FSAR. The staff reviewed the FSAR as updated to Amendments 90 pertaining to this subject.

On the basis of its evaluation, the staff finds that the calculated radiological consequences of a design-basis accident are within the acceptance criteria contained in Standard Review Plan Section 6.4. Specifically, with respect to GDC 19, the applicant will protect the control room operators against radiation by the use of shielding and by the installation of a filtration system to remove airborne contaminants. After an accident, isolation of the normal makeup air occurs automatically in response to the accident signal (safety injection) or the high gaseous radioactivity signal. This places the control room ventilation system in its emergency operating mode. In this mode the CRVS maintains a positive pressure with one train at a makeup flow rate ≤ 711 cubic feet per minute (cfm) and a recirculation flow rate between 2960 and 3618 cfm. The assumptions used for calculating the control room habitability are listed in Table 6.1. The results are presented in Table 6.2.

The staff finds that the control room habitability systems are adequate to provide safe, habitable conditions within the control room under both normal and accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body for the duration of the accident. The staff concludes that the control room design satisfies the requirements of GDC 19, as well as the guideline set forth in NUREG-0737, Item III.D.3.4 (control room habitability) and is, therefore, acceptable.

The staff tracked this effort by TAC M92973.

**Table 6.1 Assumptions Used for Calculating Watts Bar
Control Room Habitability Parameters**

Item	Value
Power Level (MWt)	3592
Primary Containment Leak Rates (%)	
0-24 hours	0.25
24 hours-30 days	0.125
Bypass Leakage Fraction	0
Atmospheric Dilution Factor (λ/Q) values (sec/m ³)	
0-2 hr	3.11×10^{-3}
2-8 hr	1.64×10^{-3}
8-24 hr	8.34×10^{-4}
24-96 hr	4.36×10^{-4}
96-720 hr	1.06×10^{-4}
Filter Recirculation Flow (cfm)	3,275
Unfiltered Inflow (cfm)	51
Filter Inflow (cfm) (makeup)	711
Intake Filter Efficiency (%)	89
Control Room Volume (ft ³)	257,198
Control Room Occupancy Factor	
0-1 day	1
1-4 days	0.6
4-30 days	0.4

Table 6.2 Control Room Personnel Doses, Sievert (rem)

Item	Thyroid Dose	Gamma Whole Body	Beta Skin Dose
Staff Evaluation	0.013 (1.3)	0.002 (0.24)	0.063 (6.3)
Applicant Evaluation	0.09 (8.94)	0.026 (2.6)	0.22 (21.9)
GDC 19 Acceptance Criteria	0.30* (30)	0.05 (5)	0.30 (30)

*Equivalent dose for any other organ, for the duration of the accident.

7 INSTRUMENTATION AND CONTROL

7.1 Introduction

By letter dated May 11, 1995, the applicant submitted Amendment 89 to the FSAR. Amendment 89 incorporates Note 11 to Table 7.1-1 which states that the design of the Eagle-21 process protection system cabinets will not fully comply with Position C.6(a) of Regulatory Guide (RG) 1.118, "Periodic Testing of Electric Power and Protection Systems," Revision 2. In order to eliminate unintended reconfiguration of electric power and protection system channels or equipment after a test that uses temporary test setups, Position C.6(a) of RG 1.118 generally prohibits the use of temporary test setups. The position permits the use of temporary jumper wires with test equipment where safety system equipment to be tested is provided with facilities specifically designed for connection of the test equipment and where these facilities are considered part of the safety system.

By letter dated August 21, 1995, the applicant provided additional justification.

The staff review of the proposed Note 11 identified the following issues as needing further explanation from the applicant:

- (1) the specific Eagle-21 system-related tests that do not meet Position C.6(a) of RG 1.118
- (2) actions proposed to assure that the use of jumper wires or lifting leads for performing these tests does not compromise the design-basis safety function of the system or component being tested

The applicant explained that the periodic surveillance tests in the Eagle-21 cabinets that require the use of temporary jumpers include resistance temperature detector (RTD) response time testing and RTD cross-calibration. The applicant also stated that the exception to RG 1.118 Position C.6.(a) is that the test equipment and jumper wires are not considered safety equipment. The applicant said that this exception was previously approved by the staff and documented in Section 8.3.3.5.1 of SSER 13. This approval was limited to testing of electrical power systems and was based on the applicant's commitments made in a letter dated September 13, 1991 as follows: (1) to identify each safety system component where temporary jumpers are utilized in surveillance procedures implemented periodically and (2) to perform an analysis (e.g., a 10 CFR 50.59-type of analysis) for each test that requires jumper(s) to demonstrate that the jumper(s) will not compromise the design basis of the system or component being tested. The applicant stated that although the commitments are for the electrical power systems, the commitments also apply to periodic surveillance testing in the Eagle-21 process protection system.

On this basis, the staff concludes that the addition of Note 11 to Table 7.1-1 proposed in Amendment 89 is consistent with previously accepted deviations from RG 1.118 in that the proposed use of jumpers on a limited basis for

testing of Eagle-21 equipment will not adversely affect the tested components.
The proposed deviation is, therefore, acceptable.

The staff tracked it review by TAC M91523.

8 ELECTRICAL POWER SYSTEMS

8.3 Onsite Electric Power System

8.3.3 Common Electrical Features and Requirements

8.3.3.1 Compliance With GDCs 2 and 4

8.3.3.1.6 Cable Damage Near Splices and Terminations

In an April 14, 1995, letter, the applicant submitted a deficiency report (Construction Deficiency Report 390/95-02) in accordance with 10 CFR 50.55(e) encompassing damage (nicks, cuts, abrasions, etc.) attributed to poor workmanship during splicing and terminating of electrical cables. The safety implication of this damage was that should the circuit associated with a damaged cable fail because of moisture intrusion, power or a control function could be lost to the safety-related end devices and could interfere with the plant's safe shutdown capability. To correct this deficiency, the applicant is inspecting Class 1E splices and terminations (covered by 10 CFR 50.49) at accessible locations for damage. The applicant will repair damaged cables using approved methods.

In response to concerns raised by Region II personnel, the applicant met with the staff on April 27, 1995 (meeting summary dated May 9, 1995), to discuss cable damage, corrective actions, and repair techniques. During that meeting, additional concerns were raised pertaining to possible damage to cables that would not be inspected (e.g., cables in mild environments or cables not qualified to the standards of 10 CFR 50.49). As a result, a conference call was held with the applicant on May 11, 1995, to discuss specific concerns. It was subsequently determined that the staff needed to assess the acceptability of not inspecting Class 1E cable splices and terminations located in mild environments (non-10 CFR 50.49).

The applicant responded to this specific concern in a May 23, 1995, letter and offered the following justification:

- Only approximately 8 percent of the damage discovered during inspections to date was of a sufficient depth to need repair.
- Mild environments do not contain the high radiation doses or the moisture that could lead to common mode cable failures affecting redundant equipment.
- The ability of cables, including those with some damage, to perform their safety function under normal environmental conditions is substantiated by post-installation testing and recent laboratory dielectric testing of damaged cables. Those laboratory tests demonstrated that cables whose insulation was cut half-way through were still able to withstand their full factory test voltage.
- Control and instrumentation cables operate at ambient temperatures and thus lack the thermal stimulation necessary to propagate partial-wall

defects. In the mild environment, partial-wall cracks or cuts in power cables would propagate slowly and any subsequent failure (if not detected and repaired during normal maintenance activities) would occur randomly over the life of the plant.

The staff agrees with the applicant's justification and, therefore, finds that limiting inspections for damaged Class 1E cables to 10 CFR 50.49 installations is acceptable. The staff tracked its review of electrical power systems by TAC M92297.

8.3.3.3 Physical Independence (Compliance With GDC 17)

(5) Separation Between Open Cable Trays and Conduits

In SSER 13, the staff stated that there were several differences between RG 1.75 and the Watts Bar General Design Criterion WB-DC-30-4, "Separation/Isolation," pertaining to the electrical separation for divisional open cable trays and conduits. The staff noted that WB-DC-30-4 allowed separation distances even smaller than those supported by the latest industry guidance in ANSI/IEEE Standard 384-1992, "Standard Criteria for Independence of Class 1E Equipment and Circuits." Because of these differences, the staff stated that the applicant's justification (supported by analysis or test) for deviation from staff and industry guidance would be reviewed. This concern was transmitted to the applicant in a March 28, 1994, letter.

In SSER 14, the staff stated that a deviation from ANSI/IEEE Standard 384-1992 involving free air cable-to-conduit separation was reported in NRC Inspection Report 50-390/94-18, and that the staff questioned the adequacy and acceptability of the applicant's case-by-case analyses for deviations from WB-DC-30-4. These concerns were conveyed to the applicant during a May 12, 1994, conference call, along with requests for a detailed description of the plant's electrical separation criteria and for justification of each deviation from RG 1.75 and industry guidance.

In SSER 14, the staff also stated that the applicant responded with a general discussion of Watts Bar's deviation from RG 1.75 in a July 29, 1994, letter. That letter discussed electrical separation tests used as justification for deviations at other nuclear plants and provided a matrix comparing each separation requirement in WB-DC-30-4 to ANSI/IEEE Standard 384-1974 and the referenced plant tests. After reviewing the applicant's letter, the staff requested more information in an August 22, 1994, letter.

In the August 22, 1994, letter, the staff requested information comparing the material and flame-retardant characteristics of the cables and tray cover type at Watts Bar to the referenced test specimens. The staff also asked if General Electric TEFZEL cables were installed at Watts Bar and requested a discussion of the plant's ground fault overcurrent protection and the specific details associated with dielectric breakdown of cable insulation during overcurrent conditions. The applicant responded in a January 11, 1995, letter and during a review at the site on April 6, 1995, which also addressed agenda items contained in the staff's letter of March 16, 1995, to the applicant.

The staff reviewed the applicant's January 11, 1995, response, met with the applicant on April 6, 1995, and concluded that the referenced tests used to support electrical separation deviations at other nuclear power plants did

provide some engineering support for the electrical separation between open-top cable trays and conduits at Watts Bar. In an April 12, 1995, conference call with the applicant, the staff requested technical justification in the areas that still needed to be addressed. Those areas were the following:

- (1) judgment of the risk associated with the lack of adequate electrical separation between open top cable trays and conduits
- (2) periodic testing of breakers for which credit is taken to ensure adequate electrical separation
- (3) details to be included in the applicant's case-by-case analyses for deviations from WB-DC-30-4

The applicant addressed these three topics in a June 5, 1995, letter as follows:

- (1) The applicant judged the frequency of a postulated worst-case scenario (motor failure (locked rotor) concurrent with failure of its protective device) to be very low. In addition, the applicant judged that the most likely scenario would be a situation in which the fault condition would lead to the motor windings opening (stopping the flow of locked rotor current) or shorting (placing the resulting fault current into the instantaneous range of the backup protective device).
- (2) The applicant stated that to meet the intent of RG 1.75, credit is taken for cable protection and that the cable protective devices are of high quality commensurate with their importance to safety. Plant procedures require periodic testing of single (not in series with a fuse or other breaker) non-Class 1E breakers that protect Class 1E buses from non-Class 1E loads, primary containment penetrations, and non-Class 1E cables which could be associated circuits to enhance breaker trip function reliability. This testing includes primary current injection performed every 18 months on at least 10 percent of each type of breaker. Also the 6.9-kV reactor coolant pump penetration overcurrent protective relays are calibrated at 18-month intervals.
- (3) The applicant performs case-by-case evaluations to justify, if possible, deviations from WB-DC-30-4. These evaluations determine whether or not a worst-case fault in a cable in one tray can propagate and damage cables in an adjacent raceway. The effect of any postulated damage is evaluated to determine if the loss of all cables in both raceways could lead to a failure of a safety-related function including a worst-case single active failure in addition to the initiating fault. The applicant considered the following information in the evaluation:
 - Redundant protective devices, sized to protect the cable from auto-ignition, are adequate to prevent fault propagation even if one of the protective devices fails.
 - Worst-case fault currents in low- or medium-voltage signal circuits are of sufficiently low energy to pose no threat to cables in adjacent raceways.

- Loss of power to all circuits fed from the protective device resulting from clearing of the fault is evaluated to determine if a safety-related function is lost. Clearing the fault through the backup protective device is also considered.
- Functional redundancy, including verification of the availability of the same component in the opposite train and supporting equipment, is evaluated for fault propagation scenarios. Synergistic effects of the cable failure are also considered.

As stated above, the applicant's June 5, 1995, letter contained a discussion of breaker testing limited to single non-Class 1E breakers protecting either Class 1E buses from non-Class 1E loads, penetrations, or possible non-Class 1E associated circuits. In an effort to provide additional justification for the plant's divisional open cable tray-to-conduit and free air cable-to-conduit separation, the applicant stated, during a June 28, 1995, conference call, that the plant's breaker testing program would be expanded to encompass all (not just non-Class 1E breakers as stated earlier) single breakers in circuits determined not to be separated (open cable tray and free air cable-to-conduit) per the guidance of ANSI/IEEE Standard 384-1992. The breaker testing program was discussed in a June 29, 1995, meeting with the staff (meeting summary dated July 6, 1995) and specific details of breaker testing were discussed during conference calls on July 13 and 20, 1995. The applicant's letter, dated July 24, 1995, documented the test program commitment and the testing details discussed previously with the staff.

On the basis of the engineering support provided by the referenced tests at other nuclear power plants, the high quality of protective devices and periodic testing of selected single breakers (where IEEE Standard 384-1992 is not met), and the low probability for the propagation of damage between cables in redundant raceways/conduits leading to the complete loss of a safety-related function, the staff finds that the separation between open cable trays (including cables in free air) and conduits as specified in WB-DC-30-4 is adequate. Also, the staff finds that the type of information considered (as discussed above in item 3) to justify the case-by-case deviations from WB-DC-30-4 is acceptable.

The staff tracked its review by TAC M89109.

8.3.3.5 Compliance With GDC 18

8.3.3.5.3 Time Constraint for Stability of EDG During No-Load Startup Testing*

As part of efforts related to generic improvements to technical specifications (TSs), the staff has agreed to delete the upper voltage and frequency limits for the initial 10-second period following the fast start during the no-load emergency diesel generator (EDG) testing to be required in Surveillance Requirement (SR) Sections 3.8.1.7, 3.8.1.12, 3.8.1.15, and 3.8.1.21 of the plant's TS. To compensate, a statement is being added to those surveillances requiring that the voltage and frequency of the EDG's output remain within specified limits during steady-state operation following the initial period of transient operation. Experience has shown that a typical EDG will have a

period of voltage and frequency oscillations in its output before reaching steady-state operation which could exceed the upper limits if undampened by application of a load. The period of oscillations may extend beyond the current 10-second acceptance criterion of the SR.

Approximately 10 seconds after an emergency start of an EDG, some safety loads are applied. Because of this, the overvoltage and overfrequency excursions will be dampened and will pose no threat to the loads or the EDG if returned to within limits (steady-state operation) in a reasonable time. Accordingly, the applicant has committed (as reflected in the Bases for the TSs) to monitor and trend the actual time to reach steady-state operation obtained during these surveillances as a means of ensuring that there is no voltage regulator or governor degradation which could cause the EDG to become inoperable. On the basis of this commitment, the staff finds these TS changes acceptable.

The staff tracked its review by TAC M76742.

9 AUXILIARY SYSTEMS

9.1 Fuel Storage Facility

9.1.2 Spent Fuel Storage

In Section 9.1.2 of the SER and SSERs 5 and 15, the staff concluded that the spent fuel storage facility was acceptable. In the SER, the staff stated that the spent fuel pool (shared by both units) would provide high density storage for up to 1312 fuel assemblies. The staff also said that the racks are designed to preclude the inadvertent placement of a fuel assembly in other than a design storage location. In FSAR Amendment 89, the applicant deleted the statement that the spent fuel racks are designed to preclude the inadvertent placement of a fuel assembly in other than a design location, and said that the total number of "usable" spent fuel storage locations has been reduced to 484 positions.

The maximum storage capacity had to be reduced because fabrication deficiencies in spent fuel racks caused a reduction to 1022 usable locations and because of the potential degradation of the neutron absorber "Boraflex," which further reduced the total number of usable positions to 484. The usable 484 positions are a result of the proposed checkerboard storage arrangement that will be maintained by administrative controls until future Boraflex surveillance data are available. By letter dated April 21, 1995, the applicant described these changes in detail, and also described the proposed FSAR changes that were subsequently placed in FSAR Amendment 89. In Section 9.1.2 of SSER 15, the staff concluded that the storage restrictions and checkerboard arrangement were acceptable.

As a result of these restrictions, the applicant deleted as a design basis the statement that the spent fuel racks are designed to preclude the inadvertent placement of a fuel assembly in other than a design location. Although the design still precludes the placement of a fuel assembly in a position that was not originally designed to receive a fuel assembly (physical dimensions of storage racks have not changed), the design no longer physically precludes placement of a fuel assembly in an unanalyzed position (unanalyzed without taking credit for Boraflex). Therefore, the statement was deleted for clarity and to prevent future misinterpretation of the design basis.

The staff reviewed the structural aspects of the fabrication deficiencies. The applicant's structural reanalysis was driven by fabrication deficiencies related to welding that were discovered after the racks were installed. Reduction of storage capacity from 1022 to 484 assemblies increases the safety margin over that calculated at the time of original approval in the SER. In its April 21, 1995, letter, the applicant indicated that it was investigating replacing Watts Bar's existing spent fuel storage racks with racks removed from Sequoyah Nuclear Plant. The Sequoyah racks do not contain Boraflex, and provide fuel storage capacity approximately equal to the original design capacity of Watts Bar's existing racks. Thus, the present fuel racks with reduced capacity will be needed only until the Sequoyah racks would become available. It is the staff's understanding that the Sequoyah racks would be available before the first scheduled refueling.

On the basis of its review, the staff concludes that the proposed changes are acceptable. This conclusion is based on the staff's criticality evaluation in Section 9.1.2 of SSER 15, and on the fact that the changes more accurately describe existing design and conditions. The conclusion remains valid as long as no single rack load exceeds 80 percent of the original capacity (for example, the rack originally designed for 88 fuel assemblies is limited to 70 fuel assemblies).

The staff tracked its review by TACs M91523 and M92159.

9.3 Process Auxiliaries

9.3.2 Process Sampling System

In SSER 3, the staff stated that the postaccident sampling system conformed to all 11 criteria of Item II.B.3 of NUREG-0737 and was, therefore, acceptable. The staff also stated that before restart following the first refueling outage, the applicant will be required to submit a final procedure for estimating the degree of core damage. As stated in the original SER (1982), this would be assured by a license condition (proposed License Condition 19).

In SSER 5, the staff revised this requirement, reasoning that since there was a 5-year delay in licensing, the applicant should submit the procedure at an earlier date. In response, the applicant submitted the procedure by letter dated June 10, 1994. This resolved the staff's concerns and proposed License Condition 19 was deleted in SSER 14.

The methodology submitted in the applicant's June 10, 1994, letter uses radionuclide concentration data to estimate the degree of core damage, as specified in Criterion 2 of NUREG-0737, Item II.B.3, "Postaccident Sampling Capability," and uses core temperature, reactor vessel level, containment radiation, and containment hydrogen concentration data to verify the estimate of core damage. The applicant also revised the emergency plan implementing procedure governing the use of this methodology, as well as other plant data, for assessing the degree of reactor core damage. The revised implementing procedure, CECC-EPIP-19, Revision 5, was submitted by letter dated May 25, 1995.

The staff has reviewed the submittals. The applicant provided a plant-specific procedure for using radionuclide data and other plant data to estimate core damage in terms of classes of core damage. This approach is consistent with the intent of NUREG-0737, Item II.B.3. The applicant's methodology is based on the Westinghouse Owners Group generic methodology, which the staff previously reviewed and accepted. The results of the example provided by the applicant to illustrate the methodology appear to be consistent with NUREG/BR-0150, Volume 1, Revision 3, RTM-93, "Response Technical Manual." The staff concludes that the applicant's methodology and associated implementing procedure are acceptable.

The staff tracked its review by TAC M77543.

9.4 Heating, Ventilation, and Air Conditioning Systems

9.4.5 Engineered Safety Features Ventilation System

In Section 9.4.5 of the SER and SSERs 9, 10, and 11, the staff concluded that the diesel generator building ventilation system was acceptable. In SSER 9, the staff concluded that the ventilation system for the additional diesel generator was acceptable based on the similarity in design to the ventilation systems evaluated in the SER for the original diesel generators. In FSAR Amendment 89, the applicant revised the description of the diesel generator ventilation system as a result of changes in the system design.

The initial design and, therefore, the system description in the SER for each diesel generator, contained a battery hood exhaust fan, two diesel generator room exhaust fans, and an electrical board room exhaust fan. The final design consists of two automatically initiated diesel generator room exhaust fans, one generator and electrical panel cooling fan which starts when either of the diesel exhaust fans starts, and a manually controlled electrical board room fan. Manually controlled exhaust fans also exist (also in the initial design) for the lube oil storage room, fuel oil transfer room, and the toilet room. The battery hood exhaust fan was eliminated in the final design. The battery areas are now ventilated by the diesel generator exhaust fans.

The ventilation system for the additional diesel generator building is similar to the ventilation systems for the other diesel building systems except for some different components. The diesel unit is served by an independent ventilation system. The ventilation subsystem for the additional diesel engine room consists of two automatically initiated room exhaust fans. All other areas of the additional diesel generator building are ventilated by manually controlled ventilation and exhaust fans. The exhaust fan in the muffler room provides ventilation as required during warm weather, or will start along with the diesel generator when in the auto mode.

Having reviewed the design changes, the staff concludes that the judgments reached in the SER and SSERs 9, 10, and 11, regarding the diesel generator building's ventilation systems, have not changed and the systems are still acceptable. The ventilation safety functions and design bases have remained essentially the same, except for the deletion of the battery hood exhaust fans. The applicant has performed calculations which show that the operation of the diesel engine room exhaust fans on a monthly basis as required by the technical specifications is adequate to maintain the hydrogen levels well below the explosive limits and will prevent the buildup of hydrogen gas above 2 percent by volume. During diesel operation, the battery areas are also adequately ventilated by the diesel room exhaust system to prevent hydrogen gas buildup. Thus, the design change to eliminate the battery hood exhaust fan is acceptable.

The staff tracked its review by TAC M91523.

11 RADIOACTIVE WASTE MANAGEMENT

The evaluation in this chapter supersedes that in Sections 11.1 through 11.6 of the SER and SSER 4. These sections were rewritten to take into consideration FSAR amendments up to and including Amendment 90, and the applicant's responses dated August 5, August 19, and November 4, 1994, and February 17, March 7, April 12, and July 21, 1995, to staff requests for additional information (RAIs). The rewrite also reflects the information submitted by the applicant on July 9, 1993, pertaining to the proposed Process Control Program (PCP) for processing wet solid wastes, and June 16, 1995, pertaining to noble gas radiation monitors (TMI Item II.F.1). The evaluation herein primarily concerns the operation of Unit 1 (the applicant has not fixed a firm date to commence operation of Unit 2); however, the evaluation as it relates to gaseous and liquid effluents (i.e., quantities of radioactive materials released to the environment via gaseous and liquid effluents and the offsite radiological consequences of such releases) applies to both units.

The staff tracked its review by TACs M84429, M87197, M90253, and M91523.

11.1 Summary Description

The radioactive waste management systems are designed to control the handling and treatment of liquid, gaseous, and solid wastes. The liquid waste management system (LWMS) processes wastes from equipment and floor drains, sample wastes, decontamination and laboratory wastes, regenerant chemical wastes, and laundry/hot shower wastes. The gaseous waste management system (GWMS) provides a holdup capacity to allow the decay of short-lived noble gases stripped from the primary coolant and treatment of ventilation exhausts through high-efficiency particulate air (HEPA) adsorbers and charcoal adsorbers, as necessary, to reduce releases of radioactive materials to as low as is reasonably achievable (ALARA) levels in accordance with 10 CFR Part 20 and 10 CFR 50.34a. The solid waste management system (SWMS) provides for processing of the solid wastes generated during Unit 1 operation, and packaging and storage of such processed wastes before they are shipped to a licensed disposal facility.

As described below, the liquid and gaseous radwaste management systems for Watts Bar are capable of maintaining releases of radioactive materials in liquid and gaseous effluents to ALARA levels in accordance with 10 CFR 50.34a and the numerical guides on offsite radiation doses specified in 10 CFR Part 50 (Appendix I Sections II.A, II.B, and II.C). Further, the liquid and gaseous radwaste management systems comply with the guidance on offsite radiation doses and releases of radioactive materials to unrestricted areas via liquid and gaseous effluents specified in Docket RM-50-2, an annex to Appendix I. Demonstration of compliance with the guidance spelled out in the annex is an option to demonstration of compliance with Appendix I (Section II.D). Section II.D deals with the performance of cost-benefit analyses to determine the adequacy of liquid and gaseous radwaste management systems in controlling population doses that come from liquid and gaseous effluents, and delineates the acceptance criterion for such analyses. The *Code of Federal Regulations* (CFR) gives applicants the option of demonstrating

that the liquid and gaseous radwaste management systems of their reactors comply with either the annex guidelines or Appendix I (Section II.D) guidelines, provided they had docketed the applications for construction permits of their reactors between January 2, 1971, and June 4, 1976. Since this option is available to Watts Bar (the applicant docketed its application for a construction permit for Watts Bar on May 14, 1971), the applicant has elected to demonstrate compliance of the liquid and gaseous radwaste management systems with the annex guidelines. Calculated maximum exposure doses from liquid and gaseous effluents are compared with Docket RM-50-2 design objectives for doses in Table 11.1. The table also compares calculated applicable releases of radioactive materials from liquid and gaseous effluents to any unrestricted area to Docket RM-50-2 design objectives for releases.

Table 11.1 Calculated Docket RM-50-2,^a Dose Commitments to a Maximally Exposed Individual and Releases for Watts Bar Units 1 and 2

Parameter	Annual Doses and Releases	
	RM-50-2 Design Objectives ^b	Calculated Doses and Releases ^c
<u>Liquid effluents</u>		
Dose to total body or any organ from all pathways (mrem)	5	2.0
Activity-release estimate, excluding tritium (Ci)	5	3.3
<u>Noble-Gas Effluents (at Site Boundary)</u>		
Gamma dose in air (mrad)	10	1.0
Beta dose in air (mrad)	20	4.4
Dose to total body of an individual (mrem)	5	0.8
Dose to skin of an individual (mrem)	15	4.0
<u>Radioiodines and particulates^d</u>		
Activity-release estimate, I-131 (Ci)	1	0.17
Dose to any organ from all air pathways (mrem)	15	14.0 (thyroid)

^aAn optional method of demonstrating compliance with the cost-benefit section (Section II.D) of Appendix I to 10 CFR 50.

^bDesign objectives are on a per site basis for doses and on a per reactor basis for releases.

^cLocations resulting in maximum doses are represented. Calculated doses and releases are on a per site and per reactor basis, respectively.

^dCarbon-14 and tritium have been added to this category.

This evaluation is based on the staff's review of the applicant's design, design criteria, and design bases for the liquid, solid, and gaseous radwaste management systems and process and effluent monitoring and sampling systems described in Watts Bar FSAR Sections 11.1 through 11.5 up to and including acceptance criteria in Sections 11.1 through 11.5 of NUREG-0800, the Standard Review Plan (SRP), as the basis for its evaluation. Those SRP sections include compliance with 10 CFR Part 50 (Appendix I) guidelines for release of radioactive materials to the environment via liquid and gaseous effluents and the offsite radiological consequences due to the effluents and applicable GDCs, as acceptance criteria. Additionally, in lieu of 10 CFR 20.106, which the subject SRP sections include as an acceptance criterion, the staff used the 10 CFR 20.1302 compliance criterion as one of the current evaluation requirements. 10 CFR 20.1302 defines the criteria for radionuclide concentration limits in liquid and gaseous effluents in unrestricted areas. Guidelines for implementing the requirements of the acceptance criteria are in the ANSI standards, regulatory guides, and other documents identified in the subject SRP sections. Conformance with the acceptance criteria constitutes a basis for concluding that the radioactive waste management systems and the process and effluent monitoring and sampling systems conform to the requirements of 10 CFR Part 20 and 10 CFR Part 50.

11.2 Liquid Waste Management

11.2.1 System Description and Review Discussion

The LWMS for Watts Bar is shared between Units 1 and 2. The LWMS consists of process equipment and instrumentation necessary to collect, process, monitor, and recirculate or discharge the processed liquid radwastes. As its basis for evaluating the LWMS, the staff used the acceptance criteria specified in SRP Section 11.2 to assess system compliance with 10 CFR 50.34a and 10 CFR 20.1302 (in lieu of 10 CFR 20.106); 10 CFR Part 50 (Appendix A, GDCs 60 and 61), and 10 CFR Part 50 (Appendix I). For information on these requirements, see Section 11.1 above.

The LWMS consists of tritiated and non-tritiated waste subsystems, and a laundry/hot shower drain subsystem. The LWMS processes the radwastes generated in these subsystems, as appropriate, and subsequently releases these wastes to the environment. The LWMS also has the capability to process condensate polishing demineralizer regenerant wastes, as needed. Additionally, for tritium control, the LWMS releases the shim bleed portion (for boron control) of the letdown flow from the primary system to the environment. In this context, it should be noted that the chemical and volume control system (CVCS) processes the letdown flow from the primary system to control boron concentration in the primary reactor coolant and ensures reactor water purity. A deep bed-regenerable condensate polishing demineralizer system treats turbine condensate. In its evaluation model, the staff assumed that the steam generator blowdown will be cooled, processed by the regenerable condensate demineralizers, and returned to the condenser hotwell, as stated in FSAR Section 10.4.8.2. The staff has assumed that the regenerant wastes will be discharged to the environment without any treatment. FSAR Section 11.2.4 states that laundry, hot shower, and decontamination wastes will be normally released to the environment without treatment; however, if these wastes are found to be radioactive in excess of preestablished limits, they will be dispatched to the non-tritiated subsystem for treatment before release. The

staff has assumed that these wastes will be released to the environment without treatment.

The principal process equipment in the LWMS for treating tritiated wastes, shim bleed wastes, and non-tritiated wastes is a set of five non-regenerable mobile demineralizers. The mobile demineralizer system treats the liquid radwastes by filtration and demineralization. The first bed is loaded with ion-specific filtration media/carbon, followed by another ion-specific medium, a cation-specific bed, and then two mixed beds, all in series. The mobile demineralizer system described above is shared by both the tritiated and non-tritiated waste subsystems of both units. FSAR Table 11.2-7b indicates that, if needed, regenerant wastes from both units can also be processed by this mobile demineralizer system. However, as stated above, in its evaluation, the staff has not assumed that the regenerant wastes will be processed by the mobile demineralizer system before release. The treated non-tritiated and tritiated liquid radwastes are collected either in a single cask decontamination collection tank of capacity 56.8 m³ (15,000 gal) shared by both units or in the single CVCS monitor tank of capacity 68.1 m³ (18,000 gal) shared by both units, sampled, and discharged (if acceptable), or recirculated for further processing before discharge. Discharged material from the cask decontamination collection tank passes through one of two tank filters which removes large particulate matter. The filters are shared by both units.

The shim bleed, primary coolant pump seal leakage, primary coolant equipment drains inside the containment, primary coolant leakage from miscellaneous sources inside the containment, primary coolant system equipment drains outside the containment, primary coolant sampling system drains, and spent fuel pit line drains send waste to the tritiated waste subsystem. The auxiliary building floor drains and secondary coolant sampling drains send waste to the non-tritiated waste subsystem. The regenerant wastes are categorized as high-crud low-conductivity (HCLC) wastes and low-crud high-conductivity (LCHC) wastes. The processing of these liquid waste streams is briefly described below.

The shim bleed is collected in one of the two CVCS holdup tanks which are shared by both units. Each tank has a capacity of 477 m³ (126,000 gal). The waste is subsequently processed by the mobile demineralizer system described above, collected in the cask decontamination tank or CVCS monitor tank, and then handled as described above. The applicant estimates that the normal generation of this waste is 9.2 m³ per day per unit (2432 gallons per day per unit) and assumes that all of the processed waste will be discharged to the environment.

The primary coolant pump seal leakage (labeled as "equipment drain wastes") from each unit is collected in the unit's reactor coolant drain tank (capacity 1.3 m³ (350 gal)). These wastes are then usually dispatched to the single tritiated drain collector tank (TDCT) (capacity 93.5 m³ (24,700 gal)) shared by both units to be processed along with other tritiated wastes, excluding shim bleed wastes which are labeled as "clean wastes." The TDCT contents (i.e., equipment drain wastes and clean wastes) are processed by a TDCT and floor drain collector tank (FDCT) discharge filter shared by both units and the mobile demineralizer system, collected in the cask decontamination tank or CVCS monitor tank, and are dispositioned as identified above. The applicant estimates that normal input to the TDCT is 5.72 m³ per day (1510 gpd) per unit, made up of 0.08 m³ per day (20 gpd) per unit for equipment drain wastes

and 5.64 m³ per day (1490 gpd) per unit for clean wastes, and assumes that all of the processed wastes will be discharged to the environment.

The auxiliary building floor drains and secondary coolant sampling drains are collected in the single FDCT (capacity 87 m³ (23,000 gal)) shared by both units, processed by the TDCT and FDCT discharge filter and the mobile demineralizer system, collected in the cask decontamination tank or CVCS monitor tank, and are then dispositioned as discussed above. The applicant estimates that the normal generation of this waste (labeled as "dirty waste") is 6.1 m³ per day (1600 gpd) per unit and that all of the processed waste will be discharged to the environment.

The HCLC wastes are collected in one of the two HCLC tanks shared by both units, each of capacity 72 m³ (19,000 gal). The tank contents are processed by circulating the contents through a high-crud filter shared by both units. The circulation adequately mixes the tank contents and removes particulate matter. The tank contents are subsequently sampled and, if acceptable, are discharged to the environment, routed to the condenser hotwell, or processed further by a vendor before disposal. The LCHC wastes are collected in a neutralization tank of capacity 75.7 m³ (20,000 gal) shared by both units, neutralized by adding chemicals, circulated for mixing, and then pumped to a non-reclaimable waste tank of capacity 37.8 m³ (10,000 gal). The contents of the non-reclaimable waste tank are sampled and, if acceptable, are discharged to the environment, pumped to a vendor for further processing and disposal, or pumped to the FDCT for further processing before discharge. The applicant estimates that the normal generation of the HCLC and LCHC regenerant wastes is 12.9 m³ per day (3400 gpd) per unit with 70 percent as HCLC waste and 30 percent as LCHC waste, and assumes that all of the regenerant waste will be discharged to the environment without being processed by the mobile demineralizer system.

The laundry/hot shower wastes are collected in one of the two laundry/hot shower tanks shared by both units, each of capacity 2.3 m³ (600 gal), and the tank contents are circulated for mixing, sampled, and if acceptable, discharged to the environment or routed to the FDCT for processing by the mobile demineralizer system prior to discharge. The wastes are discharged to the environment or routed to the FDCT via a laundry tank basket strainer shared by both units. A chemical drain tank of capacity 2.3 m³ (600 gal) shared by both units normally collects radiochemical laboratory wastes and decontamination wastes. Additionally, three waste condensate tanks, each of capacity of 5.7 m³ (1500 gal), shared by both units are available to collect laundry/hot shower wastes and radiochemical laboratory and decontamination wastes. Cross ties exist among laundry/hot shower tanks, the chemical drain tank, and waste condensate tanks; also, cross ties exist between the associated tank transfer pumps. The applicant estimates that the normal generation of the wastes collected in these tanks (these wastes are collectively labeled as "detergent and decontamination wastes") is 2 m³ per day (540 gpd) per unit and assumes that all the waste will be released without being processed by the mobile demineralizer system.

The staff notes that the applicant assumed that the normal generation rate of the shim bleed waste is within the range specified for that waste generation rate in NUREG-0017 (Rev. 1). The staff also notes that the applicant's assumed normal generation rates, for other liquid wastes discussed above, are in accordance with the specified normal generation rates given in NUREG-0017,

for such wastes. The values in that document conform to industry standards. As stated above, the applicant assumes that the mobile demineralizer system will normally process the shim bleed wastes, other tritiated wastes, and floor drain waste, all totaling 42 m³ per day (11,084 gpd) for both units. FSAR Table 11.2-1 states that the mobile demineralizer system will normally process liquid wastes at a rate of 218 m³ per day (57,600 gpd). Therefore, it is evident that the processing capability for the mobile demineralizer system has sufficient margin to process any surge in the generation of these liquid wastes. Furthermore, in the unlikely event that the demineralizers have to process regenerant wastes and detergent and decontamination wastes, the mobile demineralizer system has adequate capability to handle even surges in the generation of such wastes. This is because the mobile demineralizer system can process liquid waste streams at the maximum design flow rate of 763.1 m³ per day (201,600 gpd). For these reasons, the applicant's assumed normal waste generation rates and proposed processing for liquid waste streams are acceptable. The staff also finds acceptable the applicant's assumed total discharge of all liquid wastes discussed above (i.e., shim bleed, other tritiated wastes, floor drains, regenerant wastes, and detergent and decontamination wastes), since it is conservative. In its evaluation, the staff has also assumed total discharge of these liquid wastes.

As discussed above, liquid radwastes processed before discharge are released to the environment (i.e., Tennessee River) via the cooling tower blowdown line, from the CVCS monitor tank, cask decontamination collector tank, laundry/hot shower tank, chemical drain tank, waste condensate tank, high crud tank, or non-reclaimable waste tank only after sampling of the subject tank contents shows that such a release is permissible. The LWMS intermittently discharges liquid effluents in batches to the environment. Regenerant wastes are discharged intermittently in batches to the environment. All LWMS discharges are made through a single system discharge line to the cooling tower blowdown line; the regenerant wastes are discharged through another discharge line to the cooling tower blowdown line. A minimum dilution flow of 75.7 m³ per minute (20,000 gpm) dilutes the liquid radwaste stream discharges to the cooling tower blowdown line.

All liquid radwastes released to the environment are monitored by two radiation monitors before dilution and discharge; one monitor is on the common discharge line for the LWMS discharges and the other is on the discharge line for the regenerant wastes. These monitors are located downstream of the CVCS monitor tank, cask decontamination collector tank, laundry/hot shower tank, chemical drain tank, and waste condensate tank for the liquid wastes excluding regenerant wastes, and downstream of the high-crud tank and non-reclaimable waste tank for the regenerant wastes. Each of these radiation monitors will terminate liquid waste releases to unrestricted areas before the discharge concentration via the associated discharge line exceeds a predetermined setpoint for the applicable monitor, to comply with the limits in 10 CFR Part 20 (Appendix B, Table 2, Column 2) for liquid effluent concentrations of radionuclides in unrestricted areas. To ensure such compliance, the applicant has given the methodology for establishing the operational setpoints for these radiation monitors in the Watts Bar "Offsite Dose Calculation Manual" (ODCM, previously accepted by the staff in letter dated July 26, 1994). As discussed above, the radiation monitors provide for controlled and monitored release of liquid radwastes to unrestricted areas, as required by GDCs 60 and 64.

FSAR Table 11.2-3 lists the LWMS equipment, such as the number of tanks, filters, and pumps, and their design parameters, such as capacities and flow rates. FSAR Table 11.2-1 lists the primary coolant activity fractions for the different liquid waste streams, the liquid waste generation rates for the different kinds of liquid wastes, the kinds of demineralizers in the mobile demineralizer system and the associated effective decontamination factors (DFs), and the release fractions of the various liquid waste streams. The staff used the information in these tables in conjunction with the NUREG-0017 (Rev. 1) methodology, to calculate expected liquid effluents from either Watts Bar unit during normal plant operations including anticipated operational occurrences. The staff calculated the expected liquid effluents running the GALE code with Watts Bar-specific inputs. The principal parameters used in the GALE run for obtaining liquid and gaseous effluents from any Watts Bar unit are given in Table 11.2. From the GALE run output, the staff has determined that the total quantity of all radioactive material in liquid effluents released annually to unrestricted areas from either Watts Bar unit during its normal operation including anticipated operational occurrences, will not exceed 122.1 GBq (3.3 Ci), excluding tritium and dissolved gases, and 47.4 TBq (1280 Ci) for tritium. Thus, the staff finds that the calculated total quantity of all radioactive material in liquid effluents released annually to unrestricted areas from either Watts Bar unit during its normal operation including anticipated operational occurrences, will not exceed the Docket RM-50-2 limit for total quantity of radioactive material (i.e., 185 GBq (5 Ci)). The staff finds that the applicant's calculated values (FSAR Table 11.2.7) are about the same as the staff's calculated values (Table 11.3).

Using the liquid effluent source terms given in Table 11.2, the mathematical models and guidance contained in RG 1.109 for calculating liquid pathway doses to an offsite individual, and site-specific parameters for calculating maximally exposed offsite individual doses due to liquid effluents, the staff calculated these doses due to liquid effluents from Watts Bar Units 1 and 2 during the normal plant operation including anticipated operational occurrences. Table 11.4 compares the calculated doses with Appendix I design objectives. Table 11.1 compares the calculated doses and releases with Docket RM-50-2 design objectives. In its dose calculation, the staff did not consider ingestion of food irrigated with waters that are contaminated with radioactivity; also, the staff has not considered ingestion of invertebrates. This is because the Tennessee River is not used for irrigation, and invertebrates are not harvested from the river for consumption. Also, the staff did not consider doses from swimming in or boating on the Tennessee River, since these doses were found at the Sequoyah Nuclear Plant (SNP) to be several orders of magnitude lower than the dose received from shoreline recreation as shown in the applicant's submittal dated August 5, 1994. The current dose estimates differ from the earlier estimates (NUREG-0498, "Final Environmental Statement for Watts Bar Units 1 and 2," December 1978) primarily because of differences in the liquid effluent source terms caused by changes in the liquid waste treatment system design. The total body and organ dose estimates come principally from fish consumption. As can be seen from Table 11.1, the annual dose to total body or any organ of an individual in any unrestricted area from all applicable pathways of liquid effluents resulting from normal operation including anticipated operational occurrences at Units 1 and 2 does not exceed 0.05 mSv (5 mrem). Thus, the design of the LWMS complies with the radiation dose objectives of Docket RM-50-2 for liquid effluents from LWRs at a site. Additionally, Table 11.4 shows that the LWMS complies with the dose objectives of 10 CFR Part 50 (Appendix I, Section II.A)

for liquid effluents from any LWR. The applicant's February 17, 1995, submittal and FSAR Table 11.2-11 also demonstrate compliance with these dose objectives.

Table 11.2 Principal Parameters Used in the Calculation of Liquid and Gaseous Effluents for Watts Bar, Units 1 and 2

Parameter	Value
Thermal Power, MWt	3582
Total Steam Flow Rate, kg/h	6.8E6 (1.5E7 lb/h)
Primary Coolant Mass, kg	2.45E5 (5.4E5 lb)
Letdown Purification Rate, L/min	283.9 (75 gpm)
Letdown Cation Demineralizer Rate, L/min	28.4 (7.5 gpm)
Number of U-Tube SGs	4
Mass of Water in each SG, kg	3.95E4 (8.7E4 lb)
SG Blowdown Rate, kg/h	1.36E4 (3.0E4 lb/h)
Condensate Demineralizer Regeneration Time, days	6.0
Condensate Demineralizer Flow Fraction	0.55
Ratio of Steam to Water Concentration in the SG:	
Halogens	0.01
Particulates	0.005
<u>Liquid Waste Inputs</u>	
<i>Shim Bleed Waste</i>	
Waste Collection Rate per Unit, m ³ /day	9.2 (2432 gpd)
DF for	
Halogens	1E3
Cs and Rb	1E3
Co-58	1E2
Others ^a	1E3
Collection Time, days	20.7
Process Time, days	0.875
Fraction Discharged	1.0
<i>Equipment Drain Waste</i>	
Waste Collection Rate per Unit, m ³ /day	0.076 (20 gpd)
PCA Fraction	0.1
DF for	
Halogens	1E3
Cs and Rb	1E3
Co-58	1E2
Others	1E3
Collection Time, days	3.3
Process Time, days	0.086
Fraction Discharged	1.0
<i>Clean Waste</i>	
Waste Collection Rate per Unit, m ³ /day	5.64 (1490 gpd)
PCA Fraction	0.072

Table 11.2 (Cont.)

Parameter	Value
DF for	
Halogens	1E3
Cs and Rb	1E3
Co-58	1E2
Others	1E3
Collection Time, days	3.3
Process Time, days	0.086
Fraction Discharged	1.0
<i>Dirty Waste</i>	
Waste Collection Rate per Unit, m ³ /day	6.1 (1600 gpd)
PCA Fraction	0.013
DF for	
Halogens	1E3
Cs and Rb	1E3
Co-58	1E2
Others	1E3
Collection Time, days	2.875
Process Time, days	0.08
Fraction Discharged	1.0
<i>Regenerant Waste</i>	
Waste Collection Rate per Unit, m ³ /day	12.9 (3400 gpd)
DF for Halogens, Cs and Rb, Co-58, Others	1
In laundry/hot shower wastes	1.0
Collection Time, days	3.0
Process Time, days	0.051
Fraction Discharged	1.0
<u>Gaseous Waste Inputs^b</u>	
Holdup time for Xenon and Krypton Stripped from the Primary System, days	88.5
Fill Time of Waste Gas Storage Tanks, days	35.4
Waste Gas Processing System, Particulate Filter Efficiency, %	99
<i>Containment Inputs for Gaseous Waste</i>	
Free Volume, m ³	3.1E4 (1.1E6 ft ³)
Number of High Volume Purges per Year	24
Containment High Volume Purge Efficiency, %	
Iodine	60
Particulate Filter	99

^aExcludes tritium and dissolved noble gases.

^bThere is no continuous stripping of full letdown flow.

Table 11.3 Calculated Releases of Radioactive Materials in Liquid Effluents
From Watts Bar, Units 1 and 2

Nuclide	Ci/yr/unit	MBq/yr/unit
<i>Corrosion and Activation Products</i>		
Na-24	1.5E2	4.1E-3
P-32	6.7	1.8E-4
Cr-51	1.3E3	3.4E-2
Mn-54	7.8E2	2.1E-2
Fe-55	7.4E2	2.0E-2
Fe-59	1.9E2	5.1E-3
Co-58	2.4E3	6.5E-2
Co-60	7.4E2	2.0E-2
Ni-63	6.3E1	1.7E-3
Zn-65	2.0E2	5.4E-3
W-187	2.8E1	7.6E-4
Np-239	1.6E2	4.2E-3
<i>Fission Products</i>		
Sr-89	5.6E1	1.5E-3
Sr-90	5.2	1.4E-4
Y-90	3.7	1.0E-4
Sr-91	1.1	3.0E-5
Y-91m	0.74	2.0E-5
Y-91	6.7	1.8E-4
Y-93	5.9	1.6E-4
Zr-95	1.85E2	5.0E-3
Nb-95m	1.85	5.0E-5
Nb-95	1.8E2	4.9E-3
Mo-99	5.9E2	1.6E-2
Tc-99m	5.6E2	1.5E-2
Ru-103	2.7E3	7.3E-2
Rh-103m	2.7E3	7.3E-2
Ru-106	3.6E4	9.7E-1
Rh-106	3.6E4	9.6E-1
Ag-110m	5.6E2	1.5E-2
Ag-110	6.7E1	1.8E-3
Sb-124	1.6E1	4.3E-4
Te-129m	6.7E1	1.8E-3
Te-129	4.4E1	1.2E-3
Te-131m	3.0E1	8.1E-4
Te-131	5.6	1.5E-4
I-131	1.1E4	2.9E-1
Te-132	1.9E2	5.1E-3
I-132	2.3E2	6.2E-3
I-133	1.2E3	3.2E-2
I-134	8.1	2.2E-4
Cs-134	3.7E3	9.9E-2
I-135	2.1E2	5.6E3

Table 11.3 (Cont.)

Nuclide	Ci/yr/unit	MBq/yr/unit
<i>Fission Products (cont.)</i>		
Cs-136	2.9E2	7.8E-3
Cs-137	4.8E3	1.3E-1
Ba-137m	4.1E3	1.1E-1
Ba-140	3.7E3	9.9E-2
La-140	4.4E3	1.2E-1
Ce-141	5.9E1	1.6E-3
Ce-143	7.0E1	1.9E-3
Pr-143	6.7E1	1.8E-3
Ce-144	1.7E3	4.5E-2
Pr-144	1.5E3	4.1E-2
All others	0.0	0.0
Total	1.2E5	3.3

Notes: Tritium release is 4.7E7 MBq/yr/unit (1280 Ci/yr/reactor).

0.0 indicates that the value is less than 0.37 MBq/yr/unit (1.0E-5 Ci/yr/unit).

The staff independently calculated the annual average liquid effluent concentrations of radionuclides using these assumptions: primary coolant concentrations (PCCs) for iodine isotopes consistent with the Unit 1 Technical Specifications (TSs) limit for PCC of iodine isotopes, a PCC corresponding to 1-percent failed fuel or expected values (as given by GALE code run), whichever is greater, for fission products other than iodine isotopes and dissolved gases, expected values for corrosion products (as given by GALE code run); tritium PCC given in NUREG-0017 (Rev. 1, Table 2-2); annual liquid effluent data as given by GALE code run (see Table 11.3), and a minimum liquid effluent dilution flow of 75.7 m³ per minute (20,000 gpm). The staff determined that the sum of the ratios of the annual average liquid effluent concentrations of radionuclides in any unrestricted area, due to normal operation (including anticipated operational occurrences) of either Watts Bar unit, to the liquid effluent concentration limits for the respective radionuclides specified in 10 CFR Part 20 (Appendix B, Table 2, Column 2), is less than 1.0. Therefore, the LWMS for Watts Bar Units 1 and 2 complies with 10 CFR 20.1302, since the subject section requires that the annual average concentrations of radioactive materials in liquid effluents in an unrestricted area due to operation of any LWR do not exceed the limits specified for the corresponding radionuclides in the subject table column. FSAR Table 11.2-7b demonstrates compliance with 10 CFR 20.1302.

The staff has reviewed the quality assurance provisions for LWMS components, the quality group classifications used for system components, the seismic design applied to structures housing the LWMS components, and other design features incorporated in the system to meet the guidelines of RG 1.143,

Table 11.4 Calculated Appendix I Dose Commitments to a Maximally Exposed Individual for Watts Bar Units 1 and 2

Parameter	Annual Dose per Reactor Unit	
	Appendix I design objectives ^a	Calculated doses ^b
<u>Liquid Effluents</u>		
Dose to Total Body From All Pathways (mrem)	3	0.8
Dose to Any Organ From All Pathways (mrem)	10	1.0
<u>Noble-gas effluents (at site boundary)</u>		
Gamma Dose in Air (mrad)	10	0.5
Beta Dose in Air (mrad)	20	2.2
Dose to Total Body of an Individual (mrem)	5	0.4
Dose to Skin of an Individual (mrem)	15	2.0
<u>Radioiodines and particulates^c</u>		
Dose to Any Organ From All Air Pathways (mrem)	15	7.0 (thyroid)

^aDesign Objectives from Sections II.A, II.B, and II.C, of Appendix I, 10 CFR Part 50 consider doses to maximally exposed individual.

^bLocations resulting in maximum doses are represented here.

^cCarbon-14 and tritium have been added to this category.

Table 11.5 Staff's Calculated Releases of Radioactive Materials in Gaseous Effluents from Watts Bar Units 1 and 2

Nuclide	Waste Gas Processing Sys.		Reactor Building		Aux. Building ^a		Turbine Building ^b		Air Ejector		Total	
	Release/yr/unit MBq	Ci	Release/yr/unit MBq	Ci	Release/yr/unit MBq	Ci	Release/yr/unit MBq	Ci	Release/yr/unit MBq	Ci	Release/yr/unit MBq	Ci
Kr-83m	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Kr-85m	0.0	0.0	2.6E5	7.0	1.1E5	3.0	0.0	0.0	7.4E4	2.0	4.4E5	1.2E1
Kr-85	4.8E7	1.3E3	3.4E7	9.2E2	3.3E5	9.0	0.0	0.0	1.5E5	4.0	8.1E7	2.2E3
Kr-87	0.0	0.0	7.4E4	2.0	1.1E5	3.0	0.0	0.0	3.7E4	1.0	2.2E5	6.0
Kr-88	0.0	0.0	3.0E5	8.0	2.2E5	6.0	0.0	0.0	1.1E5	3.0	6.3E5	1.7E1
Kr-89	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Xe-131m	2.6E5	7.0	4.1E7	1.1E3	5.6E5	1.5E1	0.0	0.0	2.6E5	7.0	4.1E7	1.1E3
Xe-133m	0.0	0.0	1.4E6	3.9E1	3.7E4	1.0	0.0	0.0	0.0	0.0	1.5E6	4.0E1
Xe-133	0.0	0.0	1.0E8	2.8E3	2.0E6	5.5E1	0.0	0.0	9.6E5	2.6E1	1.1E8	2.9E3
Xe-135m	0.0	0.0	0.0	0.0	1.1E5	3.0	0.0	0.0	3.7E4	1.0	1.5E5	4.0
Xe-135	0.0	0.0	3.1E6	8.3E1	6.7E5	1.8E1	0.0	0.0	3.3E5	9.0	4.1E6	1.1E2
Xe-137	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Xe-138	0.0	0.0	0.0	0.0	1.1E5	3.0	0.0	0.0	3.7E4	1.0	1.5E5	4.0
I-131	0.0	0.0	4.1E2	1.1E-2	5.4E3	1.46E-1	4.4E2	1.2E-2	0.0	0.0	6.3E3	1.7E-1
I-133	0.0	0.0	7.8E2	2.1E-2	1.7E4	4.68E-1	1.2E3	3.2E-2	4.1	1.1E-4	1.9E4	5.2E-1
Cr-51	5.2E-3	1.4E-7	3.4	9.2E-5	1.9E1	5.0E-4	0.0	0.0	0.0	0.0	2.2E1	5.9E-4
Mn-54	7.8E-4	2.1E-8	2.0	5.3E-5	1.4E1	3.8E-4	0.0	0.0	0.0	0.0	1.6E1	4.3E-4
Co-57	0.0	0.0	3.0E-1	8.2E-6	0.0	0.0	0.0	0.0	0.0	0.0	3.0E-1	8.2E-6
Co-58	3.2E-3	8.7E-8	9.3	2.5E-4	8.5E2	2.3E-2	0.0	0.0	0.0	0.0	8.5E2	2.3E-2
Co-60	5.2E-3	1.4E-7	9.6E-1	2.6E-5	3.2E2	8.7E-3	0.0	0.0	0.0	0.0	3.2E2	8.7E-3
Fe-59	6.7E-4	1.8E-8	1.0	2.7E-5	1.9	5.0E-5	0.0	0.0	0.0	0.0	2.8	7.7E-5
Sr-89	1.6E-2	4.4E-7	4.8	1.3E-4	1.1E2	2.9E-3	0.0	0.0	0.0	0.0	1.1E2	3.0E-3
Sr-90	6.3E-3	1.7E-7	1.9	5.2E-5	4.1E1	1.1E-3	0.0	0.0	0.0	0.0	4.1E1	1.1E-3
Zr-95	1.8E-3	4.8E-8	0.0	0.0	3.7E1	1.0E-3	0.0	0.0	0.0	0.0	3.7E1	1.0E-3
Nb-95	1.4E-3	3.7E-8	6.7E-1	1.8E-5	8.9E1	2.4E-3	0.0	0.0	0.0	0.0	8.9E1	2.4E-3
Ru-103	1.2E-3	3.2E-8	5.9E-1	1.6E-5	2.3	6.1E-5	0.0	0.0	0.0	0.0	2.85	7.7E-5
Ru-106	1.0E-3	2.7E-8	0.0	0.0	2.8	7.5E-5	0.0	0.0	0.0	0.0	2.8	7.5E-5
Sb-125	0.0	0.0	0.0	0.0	2.3	6.1E-5	0.0	0.0	0.0	0.0	2.3	6.1E-5
Cs-134	1.2E-2	3.3E-7	9.3E-1	2.5E-5	8.3E1	2.24E-3	0.0	0.0	0.0	0.0	8.5E1	2.3E-3
Cs-136	2.0E-3	5.3E-8	1.2	3.2E-5	1.8	4.8E-5	0.0	0.0	0.0	0.0	3.0	8.0E-5
Cs-137	2.8E-2	7.7E-7	2.0	5.5E-5	1.3E2	3.4E-3	0.0	0.0	0.0	0.0	1.3E2	3.5E-3
Ba-140	8.5E-3	2.3E-7	0.0	0.0	1.5E1	4.0E-4	0.0	0.0	0.0	0.0	1.5E1	4.0E-4
Ce-141	8.1E-4	2.2E-8	4.8E-1	1.3E-5	9.6E-1	2.6E-5	0.0	0.0	0.0	0.0	1.4	3.9E-5
H-3	0.0	0.0	9.3E5	2.5E1	4.3E6	1.15E2	0.0	0.0	0.0	0.0	5.2E6	1.4E2
C-14	4.4E4	1.2	5.9E4	1.6	1.7E5	4.5	0.0	0.0	0.0	0.0	2.7E5	7.3
Ar-41	0.0	0.0	1.3E6	3.4E1	0.0	0.0	0.0	0.0	0.0	0.0	1.3E6	3.4E1

^aIncludes fuel handling area releases^bIncludes SG blowdown vent offgas releases

NOTE: 0.0 appearing in the table indicates release in less than 3.7E4 MBq (1.0 Ci)/yr for noble gases and 3.7 MBq (1E-4 Ci)/yr for iodines.

"Design Guidance for Radwaste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants." Specifically, the staff has reviewed the applicant's February 17, 1995, submittal, which discussed in detail how the Watts Bar LWMS conforms to the guidelines of RG 1.143. On the basis of its review, the staff finds that the LWMS is housed in seismic Category I structures. The system has design features to control the release of radioactive materials as a result of overflows from the system tanks. Tank levels are monitored either locally or in the control room, and high-level alarms are activated should preset levels be exceeded. Overflow provisions such as sumps, trenches, and overflow lines permit collection and subsequent processing of the overflow, as applicable. Design features such as elevated duct work, curbs, and routing of floor drains in buildings housing the LWMS prevent radioactive leakage from entering unmonitored and non-radioactive systems. On this basis, the staff has determined that the Watts Bar LWMS conforms to the applicable guidelines of RG 1.143, and thus complies with GDCs 60 and 61, and 10 CFR 50.34a, insofar as these relate to control of radioactive materials released to the environment via liquid effluents, assuring adequate safety of equipment that may contain liquid radwastes under normal and postulated accident conditions, and adequacy of design information for the LWMS, respectively.

11.2.2 Conclusion

On the basis of its review, as discussed in Section 11.2.1 above, the staff concludes that the applicant has submitted sufficient design information for the Watts Bar Units 1 and 2 LWMS in accordance with 10 CFR 50.34a requirements. The staff also concludes that the Watts Bar LWMS contains the equipment and design features necessary to (1) control the release of radioactive materials in liquid effluents to unrestricted areas in accordance with GDC 60 and 10 CFR Part 50 (Appendix I, Annex), i.e., Docket RM-50-2, guidance on quantity of radioactive material released to unrestricted areas via liquid effluents and (2) assure adequate safety under normal and postulated accident conditions in accordance with GDC 61. The staff further concludes that the Watts Bar LWMS complies with the requirements of 10 CFR 20.1302 with respect to liquid effluent concentrations of radionuclides in any unrestricted area during periods of fission product leakage into the primary reactor coolant at design levels from the fuel (i.e., 1%). Also, the staff concludes that the Watts Bar LWMS complies with 10 CFR Part 50 (Appendix I, Section II.A and Annex to Appendix I), i.e., Docket RM-50-2, guides on radiation doses due to liquid effluents to an offsite individual in an unrestricted area. For these reasons, the staff concludes that the LWMS for Watts Bar Units 1 and 2 meets the acceptance criteria of SRP Section 11.2 and is, therefore, acceptable.

11.3 Gaseous Waste Management

11.3.1 System Description and Review Discussion

The gaseous waste management system (GWMS) comprises the gaseous radwaste processing system (GRPS) and the portions of the plant ventilation system and main condenser evacuation system that relate to gaseous effluent from these systems to the environment. For its evaluation of the GWMS, the staff used the acceptance criteria given in SRP Section 11.3 and SRP BTP ETS B11-5 to assess system compliance with 10 CFR 50.34a; 10 CFR 20.1302 (in lieu of 10 CFR 20.106), 10 CFR Part 50 (Appendix A - GDC 3, GDC 60, and GDC 61), and 10 CFR

Part 50 (Appendix I, Sections 11.B and II.C and Annex), i.e., Docket RM-50-2. See Section 11.1 for information on these requirements.

The GWMS controls, collects, processes, stores, and disposes of, as applicable, the gaseous radwastes generated during normal operation including anticipated operational occurrences of Watts Bar Units 1 and 2. The system consists of equipment and instrumentation necessary to reduce releases of radioactive gases and particulates to the environment via gaseous effluents from the plant. The principal sources of gaseous radwastes in the plant are the effluents from the GRPS, the main condenser evacuation system, and ventilation exhausts from containment purging and auxiliary building including the fuel handling area.

The GRPS for Watts Bar is shared between Units 1 and 2. The system comprises two waste gas compressors, nine waste gas decay tanks (also known as waste gas storage tanks), each with a design pressure of 1136 kPa (150 psig) and a volume of 17 m³ (600 ft³), and associated piping, valves, and instrumentation. (The gas stripper is part of the CVCS and so is not mentioned here.) The major inputs to the GRPS are the hydrogenated fission gases, which are the reactor coolant system (RCS) gases stripped from the CVCS volume control tank letdown flow during RCS dilution and boration (i.e., gases stripped from the shim bleed), gases stripped during RCS degassing preceding reactor shutdown (two RCS volumes are assumed to be degassed annually), and gases stripped from the reactor coolant drain tank. The stripped waste gases from the various sources are discharged to a vent header. The gases discharged to the vent header are compressed into the pressurized storage tanks for decay before release to the environment by redundant 1.13 m³ per minute (40 ft³ per minute) capacity compressors, as needed. Gas from the waste gas storage tank is released to the environment via the applicable unit's shield building vent after passing through HEPA filters and charcoal adsorbers installed in the exhaust pathway of the applicable unit's reactor building purge ventilating system. The HEPA filter and charcoal adsorber reduce the release of radioiodine and particulates to the environment via gaseous effluents from the GRPS. On the basis of information and a waste gas flowrate of 4.9 m³ per minute (173 ft³ per minute) to the pressurized storage tanks given in Table 11.3-4, the staff calculated a decay time of 88.5 days for xenon and krypton radionuclides in the storage tanks, before release of these radionuclides to the environment. On the basis of this calculation, the staff considers the system capacity and design adequate for meeting the demands of the plant during normal operation including anticipated operational occurrences.

The GRPS includes an in-line radiation monitor upstream of the shield building vent, which continuously monitors the gaseous release from the waste gas storage tanks. To comply with 10 CFR Part 20 (Appendix B, Table 2, Column 1) limits for gaseous effluent concentrations of radionuclides in unrestricted areas, the monitor initiates termination of the release when the radiation level in the release reaches a predetermined setpoint for the radiation monitor. The methodology for establishing the setpoint for the radiation monitor is given in the Watts Bar ODCM. As discussed above, the design of the system ensures control and monitored release of radioactive materials to the environment, as required by GDCs 60 and 64.

Since the potential exists for explosive mixtures of hydrogen and oxygen to build up in the GRPS, SRP Section 11.3 guidelines indicate that the system should either be designed to withstand the effects of a hydrogen explosion or

should be designed to preclude the formation or buildup of explosive mixtures. The Watts Bar GRPS is designed to preclude the formation of explosive mixtures. For this purpose, the system has two automatic oxygen analyzers, one downstream of the waste gas compressors to continuously monitor the oxygen concentration level in the discharge of the operating compressor and another to sequentially monitor oxygen concentration levels in the gas space of the CVCS volume control tank, pressurizer relief tank, holdup tanks, reactor coolant drain tanks, spent resin storage tank, and waste gas storage tanks. These analyzers annunciate high alarm and high-high alarm at 2-percent and 4-percent oxygen concentration (by volume), respectively, either locally or in the main control room, as applicable. The oxygen monitoring system for the GRPS does not conform to SRP Section 11.3 guidelines, because it is not designed to automatically initiate such corrective actions as isolation of oxygen sources from the GRPS and injection of nitrogen diluent into the system to eliminate the potential for explosion, upon high-high oxygen concentration level alarm setting (4% oxygen concentration by volume) of either monitor. However, Watts Bar FSAR Section 11.3.2 briefly describes the operator actions designed to prevent the formation of an explosive mixture in the GRPS, both on high and high-high oxygen concentration alarms of either analyzer. The applicant's February 17, 1995, submittal elaborates on these operator actions at high and high-high oxygen concentration level alarms. These actions consist of isolating the affected waste gas storage tank, determining the source of high oxygen concentration level with the help of the sequential analyzer, and reducing oxygen concentration in the source tank to less than 2 percent by volume within 48 hours by nitrogen purge or tank release or both, minimizing the processes that could cause an increase in the vent head pressure and suspending waste gas compressor operation, as applicable. Furthermore, as per the February 17 submittal, the applicant will develop a program to satisfy the administrative controls for TS 5.7.2.15, "Explosive Gas and Storage Tank Radioactivity Monitoring Program." This program will provide for monitoring and control of potential explosive mixtures contained in the waste gas storage system. The program will limit the concentration of oxygen and will provide surveillance to ensure that the limits are not exceeded. Additionally, as stated in the applicant's July 21, 1995, submittal, this program will establish corrective actions (such as grab sampling once every 4 hours and analysis of the sample within the following 4 hours) should either analyzer be inoperable, and will establish reporting requirements, should either monitor continue to be inoperable for more than 30 days. On the basis of this information, particularly, the proposed operator actions not only at high-high oxygen concentration level (i.e., 4% by volume) but also at high oxygen concentration level (i.e., 2% by volume), as monitored by either analyzer, the staff has determined that the oxygen monitoring system for the GRPS meets the intent and purpose of SRP Section 11.3 guidelines for such monitoring systems and that, consequently, the GRPS complies with GDC 3. These findings supersede the staff's safety evaluation in Section 11.3 of SSER 8.

The staff has reviewed the quality assurance provisions for GRPS components, the quality group classifications used for GRPS components, the seismic design applied to waste gas storage tanks and the associated piping and valves, the seismic design applied to the structure that houses the GRPS, and other design features incorporated in the GRPS to meet the guidelines of RG 1.143. Specifically, the staff has reviewed the applicant's February 17, 1995, submittal, which discusses in detail how the Watts Bar GRPS meets the guidelines of RG 1.143. On the basis of its review, the staff finds that the

GRPS is housed in a seismic Category I structure (i.e., the auxiliary building) and that the waste gas storage tanks and associated piping and valves are seismic Category I, and thus meets RG 1.143 guidelines for the seismic design of the GRPS. The staff also finds that the GRPS meets the other guidelines of RG 1.143 (e.g., has provisions which, in conjunction with operator actions as explained above, will prevent formation of explosive mixtures in the GRPS). Thus, the staff finds that the GWMS for Watts Bar complies with GDCs 60 and 61 and 10 CFR 50.34a, insofar as they relate to the control of radioactive materials released to the environment via gaseous effluents, assuring adequate safety under normal and postulated accident conditions for the equipment that may store gaseous radwastes, and adequacy of design information for the GWMS, respectively.

The applicant has analyzed the radiological consequences at the exclusion area boundary (EAB) due to GRPS failure. The applicant has based its analysis on the release of accumulated radioactive material in one waste gas storage tank over a 2-hour time period. The applicant has also assumed that the tank fails immediately after it is filled, and has determined that, even for the case of activity accumulation in the tank based on 1-percent failed fuel for fission products, the total body dose to an individual at the EAB will be less than 5 mSv (500 mrem). It may be noted that BTP ETSB 11-5 limits the total body dose to an individual in an unrestricted area due to GRPS failure to 5 mSv (500 mrem). The staff has independently analyzed the GRPS failure, using the subject BTP assumptions and site-specific 0-2 hour atmospheric dispersion factor. On the basis of its analysis, the staff has determined that the 0-2 hour total body dose to an individual at the EAB due to GRPS failure will not exceed 5 mSv (500 mrem). Therefore, the staff finds that the GRPS at the Watts Bar Plant complies with BTP ETSB 11-5 guidelines.

During normal plant operation including anticipated operational occurrences, besides the gaseous effluent from the GRPS, the auxiliary building ventilation exhaust including the exhaust from the fuel handling area, the containment purge exhaust, and the condenser air removal system exhaust also contribute to gaseous effluents from the plant to the environment. The containment purge exhaust is filtered by HEPA filters and charcoal adsorbers installed in the exhaust pathway of the reactor building purge ventilating system. In its February 17, 1995, submittal, the applicant states that this filter system is designed, constructed, and tested in accordance with the guidelines of RG 1.52, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants." FSAR Table 6.5-1 discusses how this filter system conforms to the RG 1.52 guidelines. The staff has reviewed this table and finds that the filter system meets the intent and purpose of the RG 1.52 guidelines.

The GRPS discharge and the containment purge exhaust are released to the environment through the applicable unit's shield building vent. The auxiliary building exhaust, which includes the exhaust from the fuel handling area, is released to the environment through the exhaust vent of the auxiliary building vent. The condenser air removal system exhaust is released through the exhaust vent of the condenser air removal system; the vent is located on the roof of the unit's turbine building. All these exhausts are monitored before being released to the environment. Additionally, the service building exhaust is filtered by HEPA filters and discharges to the environment through two exhaust vents on the service building roof. The exhaust from potentially

contaminated areas of the service building, such as the radiochemical laboratory, the titration room, the protective clothing decontamination facility, and the ventilation room is monitored before being released to the environment; however, the staff does not expect the service building exhaust to be a significant contributor to radioactive gaseous effluents from the plant to the environment. The turbine building exhaust is released through multiple vents of the turbine building; the exhaust is not monitored since it is not expected to have any detectable radioactivity. For each of the gaseous effluent release points mentioned above, FSAR Section 11.3.8 describes such release point characteristics such as vent location, its size and shape, and effluent flow rate through it, as applicable. The staff reviewed the information and finds it acceptable.

Using the plant-specific parametric values given in FSAR Tables 11.1-6 and 11.3-1 and NUREG-0017 (Rev. 1) methodology, the staff calculated the expected gaseous effluents from either unit during normal plant operations including anticipated operational occurrences. Specifically, the staff calculated the expected gaseous effluents running the GALE code with Watts Bar-specific values. The principal data used in the GALE run for obtaining liquid and gaseous effluents from either Watts Bar unit are given in Table 11.2. From the GALE run output, the staff has determined that the expected annual gaseous effluent from any Watts Bar unit during normal operation including anticipated operational occurrences will not exceed 236.8 TBq (6400 Ci) for noble gases, 6.3 GBq (0.17 Ci) for iodine-131, 25.5 GBq (0.69 Ci) for iodines, 1.7 GBq (0.047 Ci) for particulates, 5.2 TBq (140 Ci) for tritium, 1.3 TBq (34 Ci) for argon-41, and 270 GBq (7.3 Ci) for carbon-14. Thus, the staff finds that the calculated iodine-131 annual release to unrestricted areas from either unit during its normal plant operation including anticipated operational occurrences will not exceed the Docket RM-50-2 annual limit for iodine-131 release to any unrestricted area via gaseous effluents from any LWR, i.e., 37 GBq (1 Ci). The staff's calculated values for individual radionuclides in gaseous effluents are given in Table 11.5; the applicant's corresponding values are given in FSAR Table 11.3-9. These tables show that the applicant's calculated value for noble gases, i.e., 200.6 TBq (5421 Ci), is less than the staff's calculated value for noble gases, i.e., 236.8 TBq (6400 Ci); however, the applicant's calculated values for all other isotopes are about the same as the staff's calculated values for these isotopes.

Using the gaseous effluent source terms given in Table 11.5, the mathematical models and guidance contained in RG 1.109 for calculating gaseous effluent pathway doses to an offsite individual, and site-specific parameters for calculating maximally exposed offsite individual doses from gaseous effluents, the staff calculated these doses from gaseous effluents during normal plant operation (including anticipated operational occurrences). Table 11.4 compares calculated doses with Appendix I design objectives. Table 11.1 compares the calculated doses and iodine-131 release with Docket RM-50-2 design objectives. The current NRC dose estimates differ from the earlier estimates (NUREG-0498, "Final Environmental Statement for Watts Bar Units 1 and 2," December 1978) because of changes in the assumptions and analytical models. For example, in the cow-milk pathway dose calculation, the staff assumed that the cow obtains all of its food from pasture for only 10 months of the year. The meteorological parameters used in these calculations are summarized in Table 11.6.

Table 11.6 Summary of Atmospheric Dispersion Factors (χ/Q) and Relative Deposition Values for Maximum Site Boundary and Receptor Locations Near Watts Bar Units 1 and 2

Location ^(a)	Source ^(b)	Relative χ/Q (sec/m ³)	Deposition (m ⁻²)
Nearest Effluent Site Boundary (1250m SE)	-- ^b	6.95x10 ⁻⁶	7.03x 10 ⁻⁹
Nearest Residence (1400m SE)	-- ^b	5.87 x 10 ⁻⁶	5.80x 10 ⁻⁹
Nearest Garden (1400m SE)	-- ^b	5.87 x 10 ⁻⁶	5.80 x 10 ⁻⁹
Nearest Milk Cow (2438m SSW)	-- ^b	1.13 x 10 ⁻⁶	2.55x 10 ⁻⁹
Nearest Milk Goat (None Identified)			

^a"Nearest" refers to that type of location where the highest radiation dose is expected to occur from all appropriate pathways. Beta and gamma air total body doses, and skin doses from noble gases are determined at the site boundary in the sector where the maximum potential value is likely to occur.

^bSources: All releases are assumed to be continuous. Releases from the shield building, turbine building (TB), and auxiliary building (AB) vents are treated as ground level.

As can be seen from the table, the annual air doses in any unrestricted area due to gamma radiation and beta radiation resulting from noble gas radionuclides in gaseous effluents from Watts Bar, Units 1 and 2 will not exceed 0.1 mGy (10 mrad) and 0.2 mGy (20 mrad), respectively; the annual total body dose and the skin dose to an individual in any unrestricted area due to noble gas radionuclides in gaseous effluents will not exceed 0.05 mSv (5 mrem) and 0.15 mSv (15 mrem), respectively. The table also shows that the cumulative annual dose from all applicable pathways to any organ of an individual in any unrestricted area due to radioiodines, particulates, carbon-14, and tritium in gaseous effluents will not exceed 0.15 mSv (15 mrem). A comparison of these doses with the radiation dose objectives of Docket RM-50-2 and Appendix I, Sections II.B and II.C shows that the Watts Bar GWMS design complies with Docket RM-50-2 and Appendix I, Sections II.B and II.C radiation dose objectives. The applicant's February 17, 1995, submittal and FSAR Table 11.3-13 also demonstrate compliance with these dose objectives.

Using the TS limit for primary coolant concentrations (PCCs) of iodine isotopes, fission product (other than iodines) concentration corresponding to

1-percent failed fuel, corrosion products as given by the GALE code run, and annual gaseous effluent data for individual radionuclides as given by the GALE code run (see Table 11.5), the staff calculated the gaseous effluent values for fission products. The staff used the gaseous effluent values for fission products calculated as described above, the gaseous effluent values for others as given by the GALE code run, and the long-term atmospheric dispersion factor (χ/Q) of 1.1×10^{-5} sec m^3 (see Table 2.6 of the Final Environmental Statement for Watts Bar, NUREG-0498, Supplement No. 1, April 1995) and has determined the following: the sum of the ratios of the annual average gaseous effluent concentrations of radionuclides in any unrestricted area, due to normal operation including anticipated operational occurrences of either Watts Bar unit, to the gaseous effluent concentration limits for the respective radionuclides specified in 10 CFR Part 20 (Appendix B, Table 2, Column 1), is less than 1.0. Therefore, the GWMS complies with 10 CFR 20.1302. Watts Bar FSAR Table 11.3-9a demonstrates compliance with 10 CFR 20.1302.

11.3.2 Conclusion

The staff concludes that the applicant has submitted sufficient design information for the Watts Bar Units 1 and 2 GWMS in accordance with 10 CFR 50.34a requirements. The staff also concludes that the Watts Bar GWMS contains the equipment and design features necessary to (1) control the release of radioactive materials via gaseous effluents to unrestricted areas in accordance with GDC 60 and 10 CFR Part 50 (Annex to Appendix I), i.e., Docket RM-50-2, (2) assure adequate safety under normal and postulated accident conditions in accordance with GDC 61, and (3) preclude the formation of an explosive mixture of hydrogen and oxygen in the GRPS portion of the GWMS in accordance with GDC 3. The staff further concludes that the Watts Bar GWMS complies with the requirements of 10 CFR 20.1302 with respect to gaseous effluent concentrations of radionuclides in any unrestricted area during periods of fission product leakage at 1 percent from the fuel into the primary coolant. Also, the staff concludes that the Watts Bar GWMS complies with 10 CFR Part 50 (Appendix I, Sections II.B and II.C) and Docket RM-50-2 guides on gamma and beta radiation air doses and offsite individual doses due to gaseous effluents in any unrestricted area. The staff concludes that the GWMS for Watts Bar Units 1 and 2 meets the acceptance criteria of SRP Section 11.3 and is acceptable.

11.4 Solid Waste Management

11.4.1 System Description and Review Discussion

The staff's evaluation of the SWMS for Watts Bar is limited to Unit 1 operation. The SWMS consists of equipment and instrumentation to collect, segregate, store, process, and monitor solid wastes. The staff used the SRP Section 11.4 acceptance criteria to assess the system compliance with (1) 10 CFR 50.34a as it relates to providing adequate system design information, (2) 10 CFR 20.1302 (in lieu of 10 CFR 20.106) as it relates to ensuring that concentrations of radionuclides in gaseous and liquid effluents to any unrestricted area arising from SWMS operation are within the limits specified for corresponding radionuclides in 10 CFR Part 20 (Appendix B, Table 2, Columns 1 and 2, respectively), (3) 10 CFR Part 50 (Appendix A, GDCs 60s and GDC 64), as they relate to controlling and monitoring the release of radioactive materials to the environment, (4) GDC 63 as it relates to

monitoring radiation levels and leakage; and (5) 10 CFR Part 71 as it relates to the packaging of radioactive materials. Additionally, the staff evaluated the system's compliance with 10 CFR Part 61 (this was issued after the subject SRP section was issued) as it relates to classifying, processing, and disposing of solid wastes. The SWMS processes both wet solid wastes (e.g., spent resins, wet radioactive filters) and dry active wastes (e.g., paper, clothing, rags, and HVAC filters), as appropriate, for shipment to a licensed low-level waste processing or disposal facility or both.

The wet solid wastes consist of primary spent resins, such as the spent resins from the CVCS and fuel pool demineralizers; secondary spent resins from the condensate polisher demineralizers; spent resins from mobile demineralizers used to process shim bleed, equipment drains, and floor drains, and as needed, laundry/hot shower wastes, chemical wastes and secondary system regenerant wastes; and spent filter elements resulting from processing liquid waste streams. The wet solid wastes will normally be processed by dewatering them.

The primary spent resins are collected in a single spent resin storage tank (SRST) with a volume 8.5 m^3 (300 ft^3). Using pressurized nitrogen, and water from the primary makeup water system, the resins in the tank are transferred as a slurry to a higher intensity container (HIC) to comply with the requirements of 10 CFR Part 61 and the licensed offsite disposal facility. The container is then dewatered by means of a portable vacuum pump supplied by a vendor. Additional slurry is added to the container, and the filling and dewatering process is repeated until a level indicator in the tank shows that the desired amount of resin has been transferred. The waste in the container is subsequently dewatered to meet the freestanding liquid limitations of 10 CFR 61.56 and the licensed disposal facility. The water removed from the container during the dewatering process is routed to the tritiated drain collector tank (TDCT). The transfer piping is backflushed with water to prevent resin buildup in the piping. Before the resins are transferred from the SRST, they are sampled to determine waste classification as per 10 CFR 61.55 specifications. This, in turn, determines what kind of containers will hold the spent resins. After the dewatering process is completed, the container is capped and loaded into a licensed shipping cask and stored until shipment, either in the auxiliary building railroad bay or in the yard east of the condensate demineralization waste evaporator (CDWE) building. If primary spent resins are stored in the yard, they are stored in a shielded container. The shipping casks are licensed per 10 CFR Part 71 requirements.

Spent resins from the mobile demineralizers are first stored in the vendor-supplied holding tanks, which can hold 7.6 m^3 (267 ft^3) and are located in a waste packaging area. When a sufficient quantity is collected for shipment to an offsite disposal facility, the spent resins are sampled for waste classification and then sluiced to a liner or HIC, as applicable, using service water. They are then dewatered to meet the freestanding liquid limitations of both 10 CFR 61.56 and the licensed disposal facility. The water from the dewatering process is routed to the TDCT. After dewatering is completed, the container is capped and loaded into a licensed shipping cask. The casks are stored until shipment, as described above.

Contaminated spent resins from the condensate polishing demineralizer system (CPDS) are collected in a CPDS storage tank, which has a capacity 12.5 m^3 (441 ft^3) and is located in the turbine building. The resins are sampled for

waste classification. Subsequently, using pressurized air and water from the condensate system, the resins are transferred to a disposal liner located on a trailer in the railroad bay of the turbine building. The liner is filled and dewatered as described above for primary spent resin processing. After the dewatering is completed to meet applicable limitations, the liner is capped and stored in the yard east of the CDWE building or the railroad bay of the auxiliary building. The yard is used for storage only when the resins are in the final disposal/shipping containers.

Spent filter elements are surveyed for dose rates upon removal from the system and the dose rates are used to determine the isotopic content and waste classification. Depending upon the waste classification, the spent filter elements are normally placed in a liner or a HIC and dewatered. The water from the dewatering process is routed to the TDCT. The container is disposed of as described above.

Before the containers are capped, the processed wet solid wastes in the containers will be sampled for freestanding liquid in accordance with a Process Control Program (PCP). These wastes will be processed into a form to comply with the requirements of 10 CFR Part 61 and the licensed disposal facility. By letter dated May 24, 1994, the applicant submitted Revision 1 of its PCP for processing wet solid wastes. The PCP states that the wet solid wastes will be processed utilizing vendor-supplied equipment operated in accordance with the vendor's PCP and procedures for processing wet solid wastes. The applicant's PCP contains a general description of the methods for controlling the processing and packaging of wet solid wastes, specific parameters for each method, and the administrative controls and quality assurance required to ensure compliance with applicable regulations and requirements. The PCP applies only to processing conducted on site and, therefore, is not applied to wastes transported to an offsite vendor for processing. The PCP addresses the various issues related to processing and packaging, such as the different waste streams that have to be processed, classification of wastes, sampling and analysis to determine the classification, processing methods, system qualification tests, batch processing, testing/inspections, acceptance criteria for processed wet wastes that are ready for shipment, and corrective actions for processed wastes that do not meet the acceptance criteria. Additionally, it addresses administrative controls, quality assurance, quality controls, training and licensee-initiated changes to the PCP. The PCP requires that all Class B and Class C wastes, as defined by 10 CFR 61.55, be either placed in HICs or solidified. Since the applicant will normally dewater the wet solid waste and not solidify it, the staff considers that the Class B and Class C wastes will normally be placed in HICs. The PCP refers to an NRC-approved vendor topical report, CNSI-DW-11118-01-P, "CNSI Dewatering Control Process Containers Topical Report." The applicant is currently committed to vendor services that will be based on this topical report. On the basis of its review of the applicant's PCP and the discussion above, the staff finds the PCP for Watts Bar acceptable.

Dry active wastes (DAWs) are separated into two types: (1) incinerable wastes such as paper, clothing, rags, plastic, mop handles, and lumber, and (2) non-incinerable wastes such as tools, valves, and motors. Incinerable wastes are placed in sealed containers. Non-incinerable wastes are placed in drums, boxes, or sealed containers. All the packaged DAW containers are stored outside the auxiliary building in a storage yard adjacent to and east of the

truck bay until shipment is available. Then they are shipped off site to a licensed processing/disposal facility for further processing and disposal. At the offsite facility, processing of these wastes may involve volume reduction for compactible wastes.

On the basis of the two most recently published annual reports of radioactive materials released from operating nuclear power plants (NUREG/CR-2907, "Radioactive Materials Released From Nuclear Power Plants," Volumes 11 and 12 for the years 1990 and 1991, respectively, the staff estimates that the processed wet wastes (i.e., spent resins and spent wet filter elements) to be shipped annually from Watts Bar Unit 1 will be 37 m³ (1300 ft³) containing approximately 10.8 TBq (293 Ci). The applicant estimates that this represents 32.3 m³ (1142 ft³) of waste, containing approximately 33.3 TBq (900 Ci). On the basis of these two reports, the staff estimates that the dry wastes that will have to be shipped annually from Watts Bar Unit 1 will contain approximately 500 GBq (13.5 Ci). The applicant estimates that the uncompacted dry wastes that will have to be shipped annually from Watts Bar Unit 1 will be approximately 850 m³ (30,000 ft³), containing 422 GBq (11.4 Ci). This estimate was based on the applicant's experience with operating the Sequoyah plant.

The applicant estimates that roughly a third of the processed wet wastes will be primary spent resins collected in the SRST. This estimate agrees with industry experience. Therefore, the expected normal waste generation annually for the primary resins will be 12 m³ (433 ft³). On the basis of the stated capacity of the SRS, the staff finds that the SRST has sufficient capacity to store primary spent resins generated at the normal generation rate for 60 days. Thus, the SWMS meets the staff's branch position BTP ETSB 11-3 (Section B.III.1).

The waste packaging area is a seismic Category I structure (it is located in the seismic Category I auxiliary control building). The area is separated into three sections that are shielded. One section is for the mobile demineralizer system, the second is for the high-level storage area and the spent resin storage containers for the mobile demineralizer system, and the third section is for low-level dry active wastes. FSAR Section 11.5.5 states that the outside storage yard where the packaged DAWs are stored until shipment has sufficient storage space to accommodate one full shipment of DAW and contaminated equipment. The applicant's July 21, 1995, submittal points out that one full shipment of stored DAW and contaminated equipment will correspond to 56.4 m³ (2000 ft³) by volume. On the basis of the discussion above regarding expected normal annual generation of wet wastes from the operation of Watts Bar Unit 1, and capacities of mobile demineralizer spent resin holding tanks and the CPDS storage tank, the storage area for wet wastes other than primary spent resins is sufficient to accommodate at least 30 days of the subject wet waste generation at the normal generation rate and, therefore, meets Section B.III.2 of BTP ETSB 11-3. Regarding storage space for DAW and contaminated equipment, the storage area for the dry active wastes and packaged contaminated equipment is sufficient to accommodate one full offsite waste shipment. Therefore, the storage area meets the criteria of Section B.III.3 of BTP ETSB 11-3.

The staff has reviewed the quality assurance provisions for SWMS components, the quality group classifications used for the SWMS components, the seismic design applied to the SRST and the structure that houses the SWMS, and other

design features incorporated in the SWMS to meet the guidelines of RG 1.143. Specifically, the staff reviewed Enclosure 1, Attachment A of TVA's February 17, 1995, submittal, which discusses in detail how the Watts Bar SWMS conforms to the guidelines of RG 1.143. The major components of the SWMS, which include the SRST, are located in the seismic Category I auxiliary building; therefore, the SWMS meets RG 1.143 guidelines for seismic design of SWMS. Thus, the staff finds that the foundations and adjacent walls of the structure that houses the SWMS are designed in accordance with RG 1.143 guidelines for seismic design of SWMS to a height sufficient to contain maximum liquid inventory expected to be in the structure. The staff notes that the SRST and piping up to and including isolation valves are designed, fabricated, and tested to ASME Code Section III, Class 3 and TVA Class D, seismic Category I criteria. The staff further notes that the SWMS meets the other guidelines of RG 1.143. Thus, the SWMS for Watts Bar meets the guidelines of RG 1.143, and complies with GDC 60 and 10 CFR 50.34a, as they relate to the control of radioactive materials released to the environment during system operation and adequacy of design information for the SWMS, respectively.

The liquid and gaseous effluents from SWMS operation are released to unrestricted areas through the LWMS and the auxiliary building ventilation system, respectively. The liquids from wet waste processing are routed to the LWMS to be processed by the LWMS before release to the environment. Specifically, the water removed from the container (liner or HIC) during dewatering goes to the TDCT to be processed by the LWMS before release to the environment. The applicant states that the resins are degassed in their respective holding tank (i.e., SRST, vendor-supplied storage container for spent resins from the mobile demineralizer system). These offgases are routed to the waste gas storage tanks and are released to the environment via HEPA filters after storage in the storage tank for sufficient time to allow decay of short-lived radionuclides. Thus, the SWMS for Watts Bar complies with GDC 60 with respect to the control of radioactive material released to the environment during system operation. The liquid and gaseous effluents resulting from the SWMS operation are included in the liquid and gaseous effluents discussed in Sections 11.2 and 11.3 of this chapter. As discussed in those sections, the liquid and gaseous effluents in unrestricted areas during the normal operation of Watts Bar comply with 10 CFR 20.1302. Therefore, the liquid and gaseous effluents in unrestricted areas arising from SWMS operation comply with 10 CFR 20.1302 with respect to effluent concentration limits of radionuclides in any unrestricted area. Also, the liquid and gaseous effluents from SWMS operation are monitored by the LWMS and auxiliary building radiation monitors before being released to the environment. Thus, the SWMS complies with GDC 64 with respect to monitoring of radioactive materials before their release to the environment. As stated above, the SRST is located in the seismic Category I auxiliary building which will retain the maximum liquid and spent resin inventory of the SRST. The SRST has level indicators. Also, it has an in-line strainer to prevent the backflow of resin fines into the primary makeup water line to the SRST. The waste packaging area mentioned above has a continuous air monitor that alarms in the event the airborne contaminants reach unacceptable levels. Additionally, an area radiation monitor is located next to the mobile demineralizers and will alarm if the gamma radiation in the area reaches unacceptable levels. On this basis, the staff finds that the SWMS for Watts Bar complies with GDC 63 with respect to monitoring radiation levels in solid waste storage area. FSAR Sections 11.5.1 and 11.5.6 state that wet wastes and dry wastes will be packaged and transported in accordance with Federal (i.e.,

10 CFR Part 61 and 10 CFR Part 71), State, and TVA regulations. The staff finds this acceptable.

11.4.2 Conclusion

On the basis of its review, as discussed in Section 11.4.1 above, the staff concludes that the SWMS for Watts Bar Unit 1 complies with the requirements of 10 CFR 20.1302; GDCs 60, 63, and 64; and 10 CFR 50.34a. The staff also concludes that with vendor-provided services, the applicant will be able to comply with the requirements of 10 CFR Part 61 and 10 CFR Part 71. For these reasons, the staff concludes that the SWMS for Watts Bar Unit 1 conforms to the acceptance criteria of SRP Section 11.4 and is, therefore, acceptable.

11.5 Process and Effluent Radiological Monitoring and Sampling System

11.5.1 System Description and Review Discussion

The staff's evaluation of the process and effluent radiological monitoring and sampling system is limited to Unit 1 operation. The system is used to measure, record, and control releases of radioactive materials in plant process streams and effluent streams. The system consists of sampling and monitoring equipment designed to indicate routine operational releases, equipment or component failure, system malfunction or misoperation, and potential radiological hazards to plant personnel or to the general public. The staff used the acceptance criteria given in SRP Section 11.5 to assess system compliance with the following regulations: (1) 10 CFR 20.1302 (in lieu of 10 CFR 20.106) as it relates to ensuring concentrations of radionuclides in gaseous and liquid effluents to unrestricted areas within the limits specified in 10 CFR Part 20 (Appendix B, Table 2, Columns 1 and 2) (this compliance is required during normal plant operations including anticipated operational occurrences); (2) 10 CFR Part 50 (Appendix A, GDCs 60 and 64) as they relate to controlling and monitoring the release of radioactive materials to the environment via gaseous and liquid effluents; and (3) GDC 63 as it relates to monitoring fuel and waste storage areas and monitoring leakages.

The radiation monitoring system (RMS) is designed to provide the following capabilities: (1) early indication of equipment, component, or system malfunction or potential radiological hazards within the station consistent with 10 CFR Part 20; (2) continuous monitoring of radioactive liquid and gaseous effluent discharge paths and control of radioactive releases to the environment consistent with the requirements of 10 CFR Part 20; GDCs 60, 63 and 64; and the guidelines of RG 1.21, "Measuring and Reporting Radioactivity in Solid Radwastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents From Light-Water-Cooled Nuclear Power Plants"; (3) monitoring of airborne activity in selected locations and effluent paths for postulated accidents in accordance with the guidelines of NUREG-0737, "Clarification of TMI Action Plan Requirements," and RG 1.97, "Instrumentation for LWR Nuclear Power Plants To Assess Plant and Environment Conditions During and Following an Accident," and in compliance with the requirements of GDC 64; and (4) data collection and data storage to support compliance reporting for the applicable NRC requirements and guidelines such as GDC 64 and RG 1.21. The RMS monitors, which monitor the different liquid and gaseous streams, also indicate and record radioactivity in the applicable streams either locally or on a main control room (MCR) panel. The monitors also annunciate alarms in the MCR on high radiation or instrument malfunction.

Radiation monitors are provided for the following gaseous process and effluent streams: (1) GPRS exhaust, (2) condenser vacuum pump (also known as condenser air removal system) exhaust during normal operation, (3) condenser vacuum pump exhaust during accident situations, (4) shield building vent exhaust during normal operation, (5) shield building vent exhaust during accident situations, (6) auxiliary building ventilation exhaust, (7) service building ventilation exhaust from radiochemical laboratory, titration rooms, protection clothing decontamination facility and ventilation room, and (8) containment purge exhaust. Radiation monitors also monitor the air spaces in the lower and upper compartments of the containment and in the air spaces above the fuel pool area of the auxiliary building; these radiation monitors and the containment purge exhaust monitors are designed to seismic Category I requirements.

Radiation monitors are provided for the following liquid process and effluent streams: (1) LWMS discharge, (2) essential raw cooling water (ERCW) system effluent, (3) component cooling water (CCW) system stream downstream of the system heat exchanger, (4) steam generator blowdown effluent, (5) condensate demineralizer regenerant effluent, (6) plant liquid discharge, and (7) turbine building sump effluent. The plant liquid discharge (item 6) comprises cooling tower blowdown discharge downstream of the yard holding pond, steam generator blowdown, and LWMS and condensate demineralizer regenerant waste discharges. The ERCW system effluent monitors are designed to seismic Category I requirements.

The monitored stream, detector type, sensitivity, range, upper limit for measurement, location, seismic category, and principal radionuclides that are monitored are given for all the liquid and gaseous process and effluent radiation monitors in FSAR Tables 11.4-1 and 11.4-2. Additionally, FSAR Table 11.4-3 lists the indicators, recorders, and alarm provisions for the gaseous process and effluent radiation monitors.

On the basis of its review of Watts Bar FSAR Section 11.4 as updated to Amendment 89, and the applicant's submittal of February 17, 1995, the staff finds that all applicable gaseous and liquid process and effluent streams identified in SRP Section 11.5 (Tables 1 and 2) are monitored for Watts Bar. Thus, the staff finds that the process and effluent radiological monitoring and sampling system for Watts Bar complies with GDC 64 insofar as it relates to monitoring all radioactive plant effluent streams until their release to the environment.

In addition to these radiation monitors (which include the fuel pool area monitors), the RMS contains other radiation monitors, such as main steamline monitors, reactor coolant drain tank (RCDT) monitors, reactor building floor and equipment drain monitors, RHR postaccident monitors, and MCR outside air intake monitors. Except for the main steamline monitors, all of the radiation monitors listed above are designed to seismic Category I requirements. Detector type, sensitivity, range, seismic classification, location, upper limit for measurement, principal radionuclides monitored indicators, recorders, and alarm provisions for all these monitors are given in FSAR Tables 11.4-2 and 11.4-3. The steamline monitors, one for each steam generator and located adjacent to the main steam lines inside the main steam valve vaults, monitor the radioactivity levels in the steam lines for primary to secondary leakage. The RCDT monitors monitor gross activity in the RCDT pump discharge. Upon detection of high radiation, the monitors initiate

isolation of the pump discharge and annunciation of audible and visible local and MCR alarms. The RHR accident monitors continuously monitor the RHR lines and, upon detection of high radiation, initiate annunciation of local and MCR alarms. The alarms facilitate performance of corrective operator actions in a timely manner. The reactor building floor and equipment drain monitors monitor gross activity in the floor and equipment drain sump pump discharges and, upon detection of high radiation, initiate isolation of the pump discharges and annunciation of local and MCR alarms. The MCR outside air intake radiation monitors, upon detection of high radiation in the outside air intake, initiate automatic switchover of the MCR ventilation system from the normal to the emergency mode and, thus, facilitate bringing filtered outside air into the MCR to comply with GDC 19 dose limits to the MCR operator. The fuel handling area monitors, on detecting high radiation in the area, initiate the switchover of the auxiliary building ventilation system from the normal to emergency mode, thus facilitating filtration of the exhaust from the areas before it is released to the environment (via the shield building vent) to comply with 10 CFR Part 100 dose limits at offsite locations. Besides the monitors discussed, the staff notes that the RMS has an air monitor in the radiation packaging area and another area monitor next to the demineralizers (see Section 11.4 above). These monitors monitor the radioactivity levels in the waste storage areas. On the basis of this discussion, the staff finds that the process and effluent radiological monitoring and sampling system complies with GDC 63.

The RMS initiates such control actions as reducing or terminating radioactive releases to the environment on detection of high radiation by applicable monitors. Specifically, the GRPS exhaust monitor, LWMS discharge monitor, steam generator blowdown effluent monitor, condensate demineralizer regenerant effluent monitor, and plant liquid discharge monitor initiate control actions to terminate the applicable discharge on detection of high radiation by the respective monitor. This automatic control feature is incorporated into the design of the monitors listed above, to comply with 10 CFR 20.1302 effluent concentration limits for radionuclides in unrestricted areas. Also, as stated above, the fuel handling area monitors, on detection of high radiation in the area, initiate switchover of the auxiliary building ventilation system from the normal to the emergency mode, thus facilitating filtration of the exhaust from the area before it is released to the environment. Therefore, the fuel handling area monitors control radioactive releases to the environment to comply with 10 CFR Part 100 dose limits at offsite locations. The CCW monitor, in the event of high radiation, initiates automatic closure of the CCW to prevent the release of gaseous activity. Additionally, some radiation monitors (e.g., RHR accident monitors, turbine building sump monitor) facilitate performance of corrective operator actions in a timely manner. The staff finds that the process and effluent radiological monitoring and sampling system complies with GDC 60 insofar as it relates to controlling radioactive releases to the environment during normal operation and accidents.

To comply with the numerical objectives for offsite doses given in 10 CFR Part 50 (Appendix I), the applicant has specified limits for annual and 3-month offsite doses in the ODCM for Watts Bar. Additionally, the applicant has specified the offsite dose limits in the ODCM, when treatment systems have to be used. The limits show that treatment systems have to be used when the monthly dose is likely to exceed about one-fourth of the annual dose limits (specified in the ODCM) prorated for 1 month. The ODCM also gives the methodology for establishing the operational setpoints for radiation monitors

that are required to limit instantaneous discharge concentration, to comply with 10 CFR 20.1302. The ODCM also gives the frequencies for channel check, source check, channel calibration, and channel operational test for all these monitors. The staff has reviewed the ODCM with respect to this information and finds that the monitoring system for Watts Bar conforms to the guidelines in NUREG-1301, "Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Pressurized Water Reactors," for monitoring gaseous and liquid effluent streams. The staff also finds that the effluent monitoring and sampling system for Watts Bar complies with Appendix I design objectives and 10 CFR 20.1302 numerical guides for offsite doses and effluent concentration limits, respectively.

FSAR Section 11.4; the applicant's responses dated August 19, 1994, and February 17, 1995; and the ODCM show that the sampling system at Watts Bar has the capability to sample process and effluent streams during normal plant operation. Specifically, ODCM Table 2.2-1 identifies the liquid waste discharges and for each release point gives the frequency of the required sampling, the required frequency for analysis, type of activity analysis, and the lower limit of detection (LLD). For each identified radioactive liquid effluent, the effluent is analyzed for principal gamma emitters, dissolved and entrained noble gases, tritium, gross alpha, strontium-89 and -90, and iron-55. Besides these liquid streams, the CCW system, spent fuel pool treatment system, yard holding pond, CST, RWST, and primary water storage tank are analyzed for principal gamma emitters, and some are additionally analyzed for tritium (see Enclosure 1 to the applicant's submittal dated February 17, 1995). ODCM Table 2.2-2 identifies the gaseous effluent streams and process streams, and for each stream gives the sampling frequency, frequency of analysis, type of activity analysis, and LLD. The streams are analyzed for principal gamma emitters, tritium, iodine and particulate, gross alpha, and strontium-89 and -90 (the service building exhaust is analyzed for principal gamma emitters only). The staff finds that the sampling program at Watts Bar conforms to the intent and purpose of NUREG-1302 guidance for sampling.

The ODCM discusses administrative controls and states that a Radioactive Effluent Release Report will be submitted annually, and that it will contain a summary of the quantities of radioactive liquid and gaseous effluents and solid wastes released from Watts Bar during the applicable year. The ODCM further states that this report will be prepared in accordance with RG 1.21 guidance for such reports. The staff finds that the process and effluent radiological monitoring and sampling system conforms to the guidance in RG 1.21 with respect to measuring, evaluating, and reporting radioactivity releases via solid wastes and liquid and gaseous effluents. Additionally, the applicant has explained how the radiation monitoring program conforms with the intent of RG 4.15, "Quality Assurance for Radiological Monitoring Programs (Normal Operation)-Effluent Streams and Environment," with respect to quality assurance provisions for the system. The staff finds that the radiation monitoring system for Watts Bar Unit 1 meets the intent and purpose of RG 4.15, with respect to quality assurance provisions for the system.

In SSERs 5 and 6, the staff concluded that Watts Bar meets the requirements of NUREG-0737 (TMI Action Plan Item II.F.1, Attachments 1 and 2) with respect to the following: (1) capability to continuously monitor applicable gaseous effluent streams for noble gases during normal plant operations and during and following accident conditions, (2) capability to continuously sample applicable gaseous effluent streams for radioiodine and particulates during

and following an accident, and (3) capability to conduct onsite analysis of the samples to determine postaccident releases of radioiodines and particulates. Further, in SSER 14, the staff found the applicant's earlier proposed range (i.e., $3.2\text{E}-7$ to $3.5\text{E}+3$ $\mu\text{Ci/cc}$) for the condenser vacuum pump exhaust vent noble gas monitor acceptable, even though it deviates from the RG 1.97 (Rev. 2) specified range (i.e., $1\text{E}-6$ to $1\text{E}+5$), for that monitor. Also, in SSER 14, the staff approved the applicant's earlier proposed range (i.e., $1.0\text{E}-6$ to $1.0\text{E}-2$ $\mu\text{Ci/cc}$) for the auxiliary building exhaust vent noble gas monitor (see SSER 14, Section 3.3.29). However, the current version of FSAR Table 11.4-2 (Amendment 89) gives the following ranges in $\mu\text{Ci/cc}$ Xenon-133 equivalent values for noble gas effluent monitors: shield building vent, $5.0\text{E}-8$ to $4.61\text{E}+4$; condenser vacuum pump exhaust vent, $3.5\text{E}-7$ to $1.4\text{E}+5$; and auxiliary building vent, $1.0\text{E}-6$ to $4.4\text{E}-1$. During accident situations, gaseous effluents are discharged only through the shield building vent and condenser vacuum pump exhaust vent and the currently specified ranges for the noble gas effluent monitors for these vents are covered by overlapping low-, medium- (only for the condenser vacuum pump exhaust vent), and high-range monitors. The staff finds that as currently specified, the ranges for noble gas effluent monitors for the shield building vent and condenser vacuum pump exhaust vent satisfy the specified ranges for such monitors in RG 1.97 (Rev. 2) and TMI Action Plan Item II.F.1 (Attachment 1). Further, the staff finds that the currently specified ranges for the condenser vacuum pump exhaust vent and the auxiliary building vent are broader than the ranges previously approved by the staff for these monitors. The staff finds this satisfactory.

11.5.2 Conclusion

On the basis its review, the staff concludes that the process and effluent radiological monitoring and sampling system for Watts Bar Unit 1 complies with 10 CFR 20.1302 and GDCs 60, 63, and 64. The staff also concludes that the system design conforms to the guidelines of NUREG-0737 (TMI Action Plan Item II.F.1, Attachments 1 and 2), RGs 1.21 and 4.15, and applicable guidelines of RG 1.97 (Rev. 2). Thus, the system meets the acceptance criteria of SRP Section 11.5 and is, therefore, acceptable.

11.6 Evaluation Findings

The material in this section of the SER and previous SSERs has been fully superseded by the material in Sections 11.1 through 11.5 of this SSER, except for the material in Section 11.6.1, "Offsite Radiological Monitoring Program," of SSER 8 which is unaffected by this supplement.

13 CONDUCT OF OPERATIONS

13.1 Organizational Structure of Applicant

In the SER, the staff found the applicant's organizational structure acceptable. Since then, the applicant has revised FSAR Section 13.1.1 to state that organizational information is as presented in TVA Topical Report TVA-NPOD89. TVA submitted the organizational structure for its nuclear plants in the form of a topical report and revisions: Revision 0 (June 1, 1989), Revision 1 (March 9, 1990), Revision 2 (April 18, 1991), Revision 3 (April 17, 1992), Revision 4 (December 27, 1993), Revision 5 (December 16, 1994), and Revision 6 (June 29, 1995). The staff has approved these submittals by letters dated December 17, 1990 (for Revisions 0 and 1), June 26, 1991 (for Revision 2), October 26, 1992 (for Revision 3), March 17, 1995 (for Revisions 4 and 5), and August 9, 1995 (for Revision 6). The applicant's submittals and the staff's evaluations are incorporated by reference. The staff's approval of the topical report and its revisions supersedes what the staff approved in the SER.

The staff tracked its review by TACs M73755, M80313, M83261, M91154, and M92885.

13.5 Plant Procedures

13.5.3 NUREG-0737 Items

Reporting Safety Valve and Relief Valve Failures and Challenges (II.K.3.3)

In the SER, the staff reported that the applicant, in a letter dated September 14, 1981, committed (1) to promptly report failures in primary system safety valves and relief valves and (2) to report all challenges to these valves in its annual report. The staff stated that these schedules are consistent with the requirements of NUREG-0737 and are acceptable.

In a letter dated June 7, 1995, the applicant revised the original commitment as follows: (1) to promptly report failures to the primary power-operated relief valves (PORVs) and safety valves and (2) to report all challenges to these valves in the time requirements specified by the Watts Bar Technical Specifications, Section 5.9.4. The staff reviewed this revised commitment, and determined that it is still consistent with the requirements of NUREG-0737. The revised commitment is acceptable.

The staff tracked its review by TAC M91523.

14 INITIAL TEST PROGRAM

In SSERs 12 and 14, the staff found the applicant's Initial Test Program (ITP) up to FSAR Amendment 88 acceptable. Subsequently, in FSAR Amendment 89, the applicant proposed several changes in order to address or resolve three open issues that were identified in SSER 14. Amendment 89 also changed FSAR Chapter 14 as a result of plant completion and preoperational testing activities.

This review also encompasses ITP changes made pursuant to commitments the applicant made in the enclosures to its letter to the NRC dated January 5, 1995. The staff tracked its review by TACs M90253, M90254, and M91523.

14.2 Preoperational Tests

The following evaluation reflects the numbering system in SSER 14.

Item 1

- (b) In FSAR Amendment 88, Section 14.2.7, subparagraph 4.A.(1)(a,1), the applicant had taken exception to performing chemical control system (boration) operability tests in accordance with the guidance in RG 1.68 (Appendix A, subparagraphs 1.b.2 and 1.n.12). The applicant stated that boron will not actually be introduced into plant systems during preoperational testing and proposed to simulate boron system operations using demineralized water. System operations using boron would be part of surveillance testing in preparation for the power ascension phase. In addition, the applicant deleted from FSAR Table 14.2-1, "Preoperational Test Summaries," Sheets 18 and 19, "Chemical and Volume Control System Test Summary," verification of reactor coolant boron concentration adjustment as a test objective and acceptance criteria.

In SSER 14, the staff found the applicant's justification for not verifying proper boron concentration adjustment in the reactor coolant system during preoperational testing unacceptable. The staff asked the applicant to reinstate its commitment to performing boration in accordance with the guidance in RG 1.68 (Appendix A, subparagraphs 1.b.2 and 1.n.12), or to provide the necessary technical justification or analysis to demonstrate that simulating boron system operations using demineralized water confirms the ability of the system to batch, store, and transfer boric acid in accordance with the design-basis requirements described in FSAR Section 9.3.4.

In its January 5, 1995, letter, the applicant committed to reinstate Chapter 14 commitments to perform boration in accordance with RG 1.68, (Appendix A, subparagraphs 1.b.2 and 1.n.12). The staff reviewed Amendment 89, and confirmed that the exception to RG 1.68 has been deleted from FSAR Section 14.2.7, subparagraph 4.A.(1)(a,1). Additionally, FSAR Table 14.2-1, Sheets 18 and 19, have been revised to reinstate performance of boration through adjustment of boron concentration in accordance with the design requirements in FSAR Section 9.3.4. These changes are in agreement with the guidance of

RG 1.68 (Appendix A, subparagraphs 1.b.2 and 1.n.12). This item is closed.

Item 11

In Amendment 88 to FSAR Chapter 14, Table 14.2-1, "Preoperational Test Summaries," Sheet 48 of 90, "AC Power Distribution System Test Summary," the applicant deleted the requirement to verify, under "Test Method," the capability of each common station service transformer (CSST) to carry the load required to supply engineered safety feature (ESF) loads of one unit under loss-of-coolant accident (LOCA) conditions in addition to power required for shutdown of the non-accident unit in accordance with the guidance in subparagraph 1.g.(1) of Appendix A to RG 1.68. This requirement is related to the design bases of Watts Bar Units 1 and 2.

In SSER 14, the staff had concluded that it would be sufficient for the applicant to demonstrate the capability of each CSST to carry the load required to supply ESF loads of one unit (Unit 1) under LOCA conditions to comply with the provisions of RG 1.68. The basis for the staff's conclusion was that, although TVA had not formally withdrawn its license application for Unit 2, the applicant was concentrating all its efforts toward obtaining the operating license (OL) for Unit 1 only. Before issuance of an OL for Unit 2, however, the applicant would have to demonstrate the capability of each CSST to carry the load required to supply ESF loads of one unit under LOCA conditions in addition to power required for shutting down the non-accident unit.

In its January 5, 1995, letter, the applicant committed to reinstate the commitment to demonstrate the capability of each CSST to carry the load required to supply ESF loads on one unit (Unit 1) under LOCA conditions in accordance with RG 1.68, Appendix A, subparagraph g.(1). The staff reviewed Amendment 89, and confirmed that Table 14.2-1, Sheets 48 and 49 have been revised to incorporate demonstration of the capability of each CSST to carry the load required to supply ESF loads on Unit 1 under LOCA conditions. These changes are in agreement with RG 1.68, Appendix A, subparagraph g.(1), as it pertains to the design basis of Unit 1. This item is closed.

Item 13

In Amendment 88 to FSAR Chapter 14, Table 14.2-1, "Preoperational Test Summaries," Sheet 11 of 90, "Post Accident Sampling System Test Summary," the applicant deleted the requirement to confirm, under "Acceptance Criteria," the capability to safely transport all samples for onsite analysis, or to a transfer point for offsite analysis, and have them analyzed within the time span described in FSAR Section 9.3.2.6, and as discussed in Item II.B.3, "Postaccident Sampling Capability" of NUREG-0737, "Clarification of TMI Action Plan Requirements."

In Amendment 87, the applicant stated in Section 9.3.2.6 that the postaccident sampling subsystem (PASS) was "designed to meet the intent of and provide for sample acquisition, analysis, and disposal, as described in Section II.B.3 of NUREG-0737, and keep personnel exposures within GDC 19 limits." In SSER 14, the staff found the applicant's proposal of not having to demonstrate this capability during preoperational testing unacceptable. The staff asked the applicant to reinstate the text deleted from Table 14.2-1, Sheet 11 of 90, and

perform the requisite testing, or provide clarification on how Section II.B.3 of NUREG-0737 would be satisfied.

In its January 5, 1995, letter, the applicant committed to perform practical test scenario(s) during Hot Functional Testing 2 using plant personnel and procedures which would demonstrate that PASS samples can be collected, transported, and analyzed within the required time frames and dose limits as committed to by the applicant's final response to Item II.B.3 of NUREG-0737, dated July 13, 1984 and September 20, 1993. TVA's commitments in the subject letter adequately satisfy Item II.B.3 of NUREG-0737 requirements. This item is closed.

14.2.3 Conclusion

This review summarizes the staff's evaluation, with respect to the ITP, as delineated in Chapter 14 of the FSAR, updated by Amendment 89, and including commitments the applicant made in its letter of January 5, 1995. The Initial Test Program description is generally comprehensive and encompasses the major phases of the testing program guidance presented in the Standard Review Plan (NUREG-0800) and Standard Format (Regulatory Guide 1.70).

APPENDIX A

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

The following is a list of documents; most of them are referenced in this SSER. In no way is this an exhaustive list of all correspondence exchanged between the staff and the applicant during this period. The reader may obtain an exhaustive list through the NRC document control system (NUDOCS), the Public Document Room, or the Local Public Document Room.

NRC Letters and Summaries

March 3, 1995	Letter, F. J. Hebdon to O. D. Kingsley (TVA), transmitting revised draft operating license NPF-20.
March 14, 1995	Letter, P. S. Tam to O. D. Kingsley (TVA), requesting additional information on Eagle-21.
March 16, 1995	Letter, P. S. Tam to O. D. Kingsley (TVA), informing of upcoming site review of electrical separation issue.
March 17, 1995	Letter, J. P. Jaudon to O. D. Kingsley (TVA), summarizing meeting to discuss the report on reasonable assurance assessment.
March 20, 1995	Letter, P. S. Tam to O. D. Kingsley (TVA), informing of upcoming audit of Thermo-Lag seismic test documents.
March 21, 1995	Letter, F. J. Hebdon to O. D. Kingsley (TVA), transmitting safety evaluation on topical report TVA-NPOD89-A, regarding organizational changes.
April 6, 1995	Letter, F. J. Hebdon to O. D. Kingsley (TVA), transmitting draft report on the status of safety issues at Watts Bar.
April 7, 1995	Letter, F. J. Hebdon to O. D. Kingsley (TVA), transmitting the draft Environmental Protection Plan (Appendix B to the operating license) for comment.
April 14, 1995	Letter, A. F. Gibson to O. D. Kingsley (TVA), finding Revision 5 to the Quality Assurance Plan, TVA-NQA-PLN89-A, acceptable.
April 18, 1995	Letter, P. S. Tam to O. D. Kingsley (TVA), transmitting the environmental assessment related to the scheduler exemption regarding vehicle bomb control program.

April 18, 1995	Letter, P. S. Tam to O. D. Kingsley (TVA), finding the small-break loss-of-coolant accident analysis acceptable.
April 19, 1995	Letter, P. S. Tam to O. D. Kingsley (TVA), requesting information regarding the reactor coolant pump oil collection system.
April 20, 1995	Letter, J. P. Jaudon to O. D. Kingsley (TVA), summarizing meeting of April 19, 1995, regarding the corrective action program on vendor information.
April 21, 1995	Letter, S. C. Flanders to O. D. Kingsley (TVA), transmitting copies of Supplement 1 of the Final Environmental Statement.
April 27, 1995	Letter, P. S. Tam to O. D. Kingsley (TVA), transmitting audit report regarding Thermo-Lag fire barrier seismic adequacy.
May 4, 1995	Letter, P. S. Tam to O. D. Kingsley (TVA), informing of availability of the final version of a contractor report regarding Watts Bar's status on various safety issues.
May 10, 1995	Letter, P. S. Tam to O. D. Kingsley (TVA), requesting additional information on carbon dioxide system in the diesel generator rooms.
May 15, 1995	Letter, J. P. Jaudon to O. D. Kingsley (TVA), summarizing meeting of May 5, 1995, regarding the corrective action plan on vendor information.
May 17, 1995	Letter, P. S. Tam to O. D. Kingsley (TVA), finding the special report on employee concerns special program deviations acceptable.
May 19, 1995	Letter, P. S. Tam to O. D. Kingsley (TVA), informing of site review of fire protection issues.
June 13, 1995	Letter, F. J. Hebdon to O. D. Kingsley (TVA), transmitting the final draft version of the Unit 1 Technical Specifications, associated Bases and Technical Requirements Manual.
June 22, 1995	Letter, P. S. Tam to O. D. Kingsley (TVA), transmitting copies of Supplement 15 of the Watts Bar Safety Evaluation Report.
June 26, 1995	Letter, P. S. Tam to O. D. Kingsley (TVA), accepting supplemental response to Generic Letter 88-08 regarding thermal stresses in reactor coolant system piping.

June 27, 1995	Letter, P. S. Tam to O. D. Kingsley (TVA), requesting additional information regarding the pressure-temperature limit report.
July 10, 1995	Letter, P. S. Tam to O. D. Kingsley (TVA), informing of upcoming site audit of structure and civil engineering features.
July 19, 1995	Letter, P. S. Tam to O. D. Kingsley (TVA), informing of the need to perform a full-participation emergency preparedness exercise.
July 21, 1995	Letter, P. S. Tam to O. D. Kingsley (TVA), transmitting audit report regarding modeling of coupled internal concrete structure and nuclear steam supply system.
July 24, 1995	Letter, F. J. Hebdon to O. D. Kingsley (TVA), requesting additional information on TVA's organization topical report, Revision 6.
July 27, 1995	Letter, P. S. Tam to O. D. Kingsley (TVA), accepting revised response to Generic Letter 88-14 regarding instrument air system problems.
September 18, 1995	Letter, F. J. Hebdon to O. D. Kingsley (TVA), granting additional relief to the preservice inspection program.
September 22, 1995	Letter, F. J. Hebdon to O. D. Kingsley (TVA), approving pressure-temperature limit methodology.

TVA Letters

March 3, 1995	Letter, O. J. Zeringue to NRC, providing additional information regarding seismic capabilities of Thermo-Lag fire barriers.
March 7, 1995	Letter, O. J. Zeringue to NRC, transmitting additional information on detailed control room design review.
March 7, 1995	Letter, O. J. Zeringue to NRC, providing additional information on environmental review.
March 14, 1995	Letter, M. O. Medford to NRC, transmitting Revision 5 of the TVA Quality Assurance Plan.
March 17, 1995	Letter, R. R. Baron to NRC, providing additional information regarding severe accident mitigation design alternatives.
March 22, 1995	Letter, R. R. Baron to NRC, responding to Generic Letter 92-08 regarding Thermo-Lag fire barriers.

March 29, 1995	Letter, R. R. Baron to NRC, transmitting Revision 3 of the Unit 1 Pressure-Temperature Limit Report.
March 29, 1995	Letter, R. R. Baron to NRC, transmitting report on three-hour Thermo-Lag fire barrier tests.
April 7, 1995	Letter, R. R. Baron to NRC, providing additional information on FSAR Chapter 14, power ascension tests.
April 12, 1995	Letter, R. R. Baron to NRC, transmitting additional information on radwaste management systems.
April 13, 1995	Letter, R. R. Baron to NRC, transmitting Revisions 24 and 25 of the preservice inspection program.
April 21, 1995	Letter, R. R. Baron to NRC, transmitting information on the spent fuel storage racks.
April 21, 1995	Letter, R. R. Baron to NRC, transmitting supplemental information on compliance with RG 1.97, regarding postaccident monitoring systems.
April 25, 1995	Letter, R. R. Baron to NRC, transmitting information on ampacity derating of cables enclosed in one-hour Thermo-Lag fire barriers.
April 27, 1995	Letter, R. R. Baron to NRC, transmitting Revision 1 of Watts Bar Nuclear Plant Fire Protection Report.
April 28, 1995	Letter, R. R. Baron to NRC, transmitting additional information on thermal stresses in reactor coolant system piping.
May 12, 1995	Letter, R. R. Baron to NRC, advising of completion of Phase V of the Program for Assurance of Completion and Assurance of Quality.
May 16, 1995	Letter, R. R. Baron to NRC, responding to the staff's questions on the proof-and-review version of the Unit 1 Technical Specifications.
May 16, 1995	Letter, R. R. Baron to NRC, submitting revised information on the detailed control room design review.
May 16, 1995	Letter, R. R. Baron to NRC, submitting Amendment 89 to the FSAR.
May 18, 1995	Letter, R. R. Baron to NRC, transmitting long-term cable bend radius program plan.
May 18, 1995	Letter, R. R. Baron to NRC, notifying of recent changes to emergency core cooling system evaluation model.

May 19, 1995	Letter, R. R. Baron to NRC, transmitting the first part of a three-part response to the staff's initial assessment of safety issues.
May 26, 1995	Letter, R. R. Baron to NRC, providing additional information regarding carbon dioxide automatic fire suppression system.
May 26, 1995	Letter, R. R. Baron to NRC, providing additional information regarding reactor coolant pump lube oil collection system.
June 5, 1995	Letter, R. R. Baron to NRC, providing additional information regarding electrical separation design.
June 7, 1995	Letter, R. R. Baron to NRC, revising commitment made in original response to Item II.K.3.3 of NUREG-0737.
June 13, 1995	Letter, R. R. Baron to NRC, notifying of intent to implement radiological respiration protection program.
June 13, 1995	Letter, R.R. Baron to NRC, providing corrected pages for FSAR Amendment 89.
June 14, 1995	Letter, R. R. Baron to NRC, transmitting the second part of three-part response to the staff's initial assessment of safety issues.
June 15, 1995	Letter, R. R. Baron to NRC, submitting Revision 3 to the Fire Protection Report.
June 15, 1995	Letter, E. T. Knuettel to NRC, providing additional information in response to Generic Letter 92-08 regarding Thermo-Lag fire barrier.
June 16, 1995	Letter, R. R. Baron to NRC, revising original response to Item II.F.1.1 of NUREG-0737, regarding noble gas radiation monitors.
June 16, 1995	Letter, O. D. Kingsley to NRC, certifying of financial assurance for decommissioning Watts Bar Unit 1 at the end of life.
June 27, 1995	Letter, R. R. Baron to NRC, providing 60-day response to Generic Letter 95-03, regarding circumferential cracking of steam generator tubes.
June 28, 1995	Letter, M. O. Medford to NRC, transmitting "Reasonable Assurance Assessment Report."
June 29, 1995	Letter, O. J. Zeringue to NRC, requesting relief from code requirement regarding testing of pressurizer safety valves within six months of initial fuel load.

June 29, 1995	Letter, P. P. Carrier to NRC, transmitting Revision 6 of topical report, TVA-NPOD89-A, regarding TVA nuclear organization.
July 13, 1995	Letter R. R. Baron to NRC, transmitting revised information on preoperational testing of fuel handling and vessel servicing equipment.
July 14, 1995	Letter, R. R. Baron to NRC, providing revised response to Generic Letter 88-14 regarding instrument air supply.
July 14, 1995	Letter, R. R. Baron to NRC, transmitting the third part of a three-part response to the staff's initial assessment of safety issues.
July 18, 1995	Letter, R. R. Baron to NRC, transmitting draft change pages to FSAR Chapter 12.
July 18, 1995	Letter, R. R. Baron to NRC, providing revised information regarding the loose parts monitoring system.
July 18, 1995	Letter, R. R. Baron to NRC, transmitting minor revision to earlier submittals on RG 1.97, regarding postaccident instrumentation.
July 18, 1995	Letter, R. R. Baron to NRC, transmitting historical information on the specific seismic spectrum.
July 19, 1995	Letter, R. R. Baron to NRC, requesting exemption from certain requirement of 10 CFR 50, Appendix E, regarding emergency exercise.
July 20, 1995	Letter, R. R. Baron to NRC, transmitting topical reports WCAP-14419 and -14420 on flow measurement uncertainty methodology.
July 21, 1995	Letter, R. R. Baron to NRC, transmitting additional information regarding the radwaste systems.
July 24, 1995	Letter, R. R. Baron to NRC, responding to the staff's July 20, 1995, questions about the TVA organization.
July 24, 1995	Letter, R. R. Baron to NRC, transmitting additional information on electrical separation.
July 25, 1995	Letter, R. R. Baron to NRC, confirming complete implementation of reactor vessel head vent system.
July 26, 1995	Letter, P. P. Carrier to NRC, transmitting additional information to support an exemption from certain requirement of 10 CFR 50, Appendix E, regarding emergency exercise.

July 27, 1995	Letter, P. P. Carier to NRC, transmitting information as proof of financial protection against nuclear liabilities.
July 31, 1995	Letter, P. P. Carier to NRC, providing additional information on the TVA organization.
July 31, 1995	Letter, R. R. Baron to NRC, responding to the staff's inquiry about topical report WCAP-14040, regarding pressure-temperature limit methodology.

APPENDIX E
PRINCIPAL CONTRIBUTORS

NRC Watts Bar Project Staff

Peter S. Tam, Senior Project Manager
Thomas V. Wambach, Senior Project Manager (contributed to Appendix EE)
Michael Bugg, Project Engineer (Intern)
Beverly A. Clayton, Licensing Assistant
Rayleona Sanders, Technical Editor

NRC Technical Reviewers

Syed A. Ali, Civil Engineering and Geosciences Branch, NRR
Harry Balukjian, Reactor Systems Branch, NRR
Frederick H. Burrows, Electrical Engineering Branch, NRR
Thyagaraja Chandrasekaran, Plant Systems Branch, NRR
Thomas M. Cheng, Civil Engineering and Geosciences Branch, NRR
Phillip C. Cota, Division of Systems Safety and Analysis, NRR
Sang Bo Kim, Civil Engineering and Geosciences Branch, NRR
Eric J. Lee, Instrumentation and Controls Branch, NRR
Yueh-Li C. (Renee) Li, Mechanical Engineering Branch, NRR
William T. Lefave, Plant Systems Branch, NRR
John L. Minns, Emergency Preparedness and Radiation Protection Branch, NRR
Juan Peralta, Quality Assurance and Maintenance Branch, NRR
Robert L. Rothman, Civil Engineering and Geosciences Branch, NRR
Robert S. Wood, License Renewal and Environmental Review Project Directorate,
NRR

Contractors

T. M. Mitts, Pacific Northwest Laboratories (contributed to Chapter 14)

APPENDIX EE
DISPOSITION OF ALL GENERIC SAFETY ISSUES APPLICABLE TO WATTS BAR

DISPOSITION OF ALL GENERIC SAFETY ISSUES APPLICABLE TO WATTS BAR

Reference: Letter, F. J. Hebdon to O. D. Kingsley, April 6, 1995
(TAC No. M90068)

The term "generic issues" is used to denote any of the following actions affecting multiple nuclear plants: unresolved safety issues (USIs), generic safety issues (GSIs), Three Mile Island (TMI) issues, and multiplant actions (MPAs). The staff expended considerable efforts to assess the status of licensing action (i.e., safety evaluation), implementation, and verification of generic issues as they pertain to Watts Bar. The staff's initial efforts resulted in a Sciencetech, Inc. report. That report was transmitted to the applicant and made public by a letter, F. J. Hebdon to O. D. Kingsley, April 6, 1995 (Accession No. 9504120291).

The applicant responded to the Sciencetech report by letters dated May 19 (Accession No. 9505310202), June 14 (Accession No. 9506210279) and July 14 (Accession No. 9507200130), 1995. The staff has reviewed the applicant's response contained in those three letters, and has taken that into consideration while preparing this assessment. The staff has also factored in information that was developed after the Sciencetech report was issued, and information that was not available to (or not found by) Sciencetech.

For licensing actions, the staff's assessment results are as summarized in the eight lists (A-H) that follow. A TAC number had either already been assigned in the Workload Identification and Scheduling Program (WISP), or has been opened for each issue. An issue is now either designated as "complete" in WISP, or will continue to be tracked by the associated TAC number.

For implementation, the staff accepts the applicant's status for each issue as presented in one of the three TVA letters mentioned above, or in other letters as reported in the Sciencetech document. An issue is now either designated as "complete" in WISP, or will be tracked by the associated TAC number.

Verification is done by the Region II staff. Since each generic issue now has a TAC number, and the WISP database will be uploaded into the staff's agency-wide database, the Safety Issues Management System, Region II will track verification in accordance with current practice.

A. No Longer Applicable

Issues in this list were interim or short-term issues for operating plants, or have been superseded by other actions. Therefore, for Watts Bar, the following items ceased to exist as such:

I.A.1.1.1: Shift technical advisor (STA) program operability - Licensing action is complete on the basis of the applicant's implementation of an accredited Systems Approach to Training Program (see SSER 9).

I.A.1.1.2: STA, training and qualifications - See I.A.1.1.1.

I.A.1.1.3: STA, long-term program - See I.A.1.1.1.

I.A.1.2: Shift supervisor responsibilities - This was closed in the SER, Section 13.5.1.

II.F.1.2.A (MPA F-020): Accident monitoring, install instruments (noble gas) - This issue was superseded by Regulatory Guide (RG) 1.97, Rev. 3, May 1983.

II.F.1.2.B (MPA F-021): Accident monitoring, install instruments (iodine/particulate sample) - This issue was superseded by RG 1.97, Rev. 3., May 1983.

II.F.1.2.C (MPA F-022): Accident monitoring, install instruments (containment high radiation) - This issue was superseded by RG 1.97, May 1983.

II.F.1.2.D (MPA F-023): Accident monitoring, install instruments (containment pressure) - This issue was superseded by RG 1.97, May 1983.

II.F.1.2.E (MPA F-024): Accident monitoring, install instruments (containment hydrogen level) - This issue was superseded by RG 1.97, May 1983.

II.F.1.2.F (MPA F-025): Accident monitoring, install instruments (containment hydrogen concentration) - This issue was superseded by R.G.1.97, May 1983.

USI A-49 (MPA A-021): Pressurized thermal shock - In Section 5.2.5 of SSER 11, the staff found that Watts Bar Unit 1 reactor vessel met the requirements of 10 CFR 50.61. The applicant's January 28, 1993 request for exemption from 10 CFR 50, Appendix G, was found not needed due to a revision to the regulation such that no exemption will be needed; the staff's review and finding of acceptance of the technical issue is documented in Section 5.3.1.1 of SSER 14.

B. NUREG-0737 Issues Not Addressed to Sub-tier Levels

Issues listed here are addressed as indicated.

I.A.2.1.1: Immediate upgrading of reactor operator (RO) and senior reactor operator (SRO) training and qualification, SRO experience - The staff completed review of licensed operator training program in Section 13.2.1 of the SER and SSER 9, and specifically addressed Item I.A.2.1 in the former. The staff's review included the substance of Item I.A.2.1.1, even though it was not specifically mentioned. This issue is closed.

I.A.2.1.2: Immediate upgrading of RO and SRO training and qualification, SRO applicants must have one year of RO experience - The staff completed its review of the licensed operator training program in Section 13.2.1 of the SER and SSER 9, and specifically addressed Item I.A.2.1 in the SER. The staff subsequently verified that the review included the substance of Item I.A.2.1.2, even though it was not specifically mentioned. This issue is closed.

I.A.2.1.3: Immediate upgrading of RO and SRO training and qualification, three months training on shift - The staff completed its review of the licensed operator training program in Section 13.2.1 of the SER and SSER 9, and specifically addressed Item I.A.2.1 in the SER. The staff reviewed the substance of Item I.A.2.1.3, even though it was not specifically mentioned. This issue is closed.

I.A.2.1.4 (MPA F-004): Immediate upgrading of RO and SRO training and qualification, modify training - The staff completed its review of the licensed operator training program in Section 13.2.1 of the SER and SSER 9, and specifically addressed Item I.A.2.1 in the SER. The staff reviewed the substance of Item I.A.2.1.4, even though it was not specifically mentioned. This issue is closed.

I.A.2.1.5: Immediate upgrading of RO and SRO training and qualifications, management certification of license applications - The staff completed its review of the licensed operator training program in Section 13.2.1 of the SER and SSER 9, and specifically addressed Item I.A.2.1 in the SER. The staff reviewed the substance of Item I.A.2.1.5, even though it was not specifically mentioned. This issue is closed.

I.A.3.1.1: Revise scope and criteria for operator licensing examinations, increase scope - This issue was superseded by NUREG-1021, "Operator Licensing Examiner Standards," and 10 CFR Part 55. The staff completed its review of the subject matter in Section 13.2.1 of the SER, SSER 9, and SSER 10. The SER specifically mentioned Item I.A.3.1. The staff reviewed the substance of Item I.A.3.1.1, even though it was not specifically mentioned. This issue is closed.

I.A.3.1.2: Revise scope and criteria for operator licensing examinations, increase passing grade - This issue was superseded by NUREG-1021, "Operator Licensing Examiner Standards," and 10 CFR Part 55. The staff completed its review of the subject matter in Section 13.2.1 of the SER, SSER 9, and SSER 10. The SER specifically mentioned Item I.A.3.1. The staff reviewed the substance of Item I.A.3.1.2, even though it was not specifically mentioned. This issue is closed.

I.A.3.1.3.A: Revise scope and criteria for operator licensing examinations, simulator examination for plants with simulators - This issue was superseded by NUREG-1021, "Operator Licensing Examiner Standards," and 10 CFR Part 55. The staff completed its review of the subject matter in Section 13.2.1 of the SER, SSER 9, and SSER 10. The SER specifically mentioned Item I.A.3.1. The staff reviewed the substance of Item I.A.3.1.3.A, even though it was not specifically mentioned. This issue is closed.

I.C.1.1: Upgrade emergency operating procedures, small-break LOCA - As confirmed by the Sciencetech report, the staff and TVA did a large amount of work on Item I.C.1. See Section 13.5.1 of the SER and Section 13.5.2.1 of SSER 9. The latter stated that the staff no longer reviews emergency operating procedure generating packages but has made such review part of the inspection program under Inspection Procedure 42001. Inspection Reports 50-390, 391/94-86 and 95-42 are examples. On such basis, this issue is closed.

I.C.1.2.A (MPA F-004): Upgrade emergency operating procedures, inadequate core cooling guidelines - As confirmed by the Sciencetech report, the staff and TVA did a large amount of work on Item I.C.1. See Section 13.5.1 of the SER and Section 13.5.2.1 of SSER 9. The latter stated that the staff no longer reviews emergency operating procedure generating packages but has made such review part of the inspection program under Inspection Procedure 42001. Inspection Reports 50-390, 391/94-86 and 95-42 are examples. On such basis, this issue is closed.

I.C.1.2.B (MPA F-005): Upgrade emergency operating procedures, revise ICC emergency operating procedures - As confirmed by the Sciencetech report, the staff and TVA did a large amount of work on Item I.C.1. See Section 13.5.1 of the SER and Section 13.5.2.1 of SSER 9. The latter stated that the staff no longer reviews emergency operating procedure generating packages but has made such review part of the inspection program under Inspection Procedure 42001. Inspection Reports 50-390, 391/94-86 and 95-42 are examples. On such basis, this issue is closed.

I.C.1.3.A (MPA F-004): Upgrade emergency response procedures, reanalyze guidelines for transients and accidents - As confirmed by the Sciencetech report, the staff and TVA did a large amount of work on Item I.C.1. See Section 13.5.1 of the SER and Section 13.5.2.1 of SSER 9. The latter stated that the staff no longer reviews emergency operating procedure generating packages but has made such review part of the inspection program under Inspection Procedure 42001. Inspection Reports 50-390, 391/94-86 and 95-42 are examples. On such basis, this issue is closed.

I.C.1.3.B (MPA F-005): Upgrade emergency operating procedures, revise emergency operating procedures for transients and accidents - As confirmed by the Sciencetech report, the staff and TVA did a large amount of work on Item I.C.1. See Section 13.5.1 of the SER and Section 13.5.2.1 of SSER 9. The latter stated that the staff no longer reviews emergency operating procedure generating packages but has made such review part of the inspection program under Inspection Procedure 42001. Inspection Report 50-390, 391/94-86 is an example of such an inspection. On such basis, this issue is closed.

I.D.2.1 (MPA F-074): Plant safety parameter display system (SPDS), basis for parameter selection - The staff completed its review in SSER 5, SSER 6, and SSER 15. This issue is closed.

II.B.3.1: Post-accident sampling system (PASS) design review - The staff completed its design review in Section 9.3.2 of the SER, SSER 3, SSER 5, and SSER 14. Although none of these documents specifically mentioned "II.B.3.1," it is obvious that the design review has been completed, and this issue is closed.

II.B.3.2 (MPA F-076): Post-accident sampling corrective actions - The staff completed its design review in Section 9.3.2 of the SER, SSER 3, SSER 5, and SSER 14. Although none of these documents specifically mentioned "II.B.3.2," it is obvious that the design review has been completed, and this issue is closed.

II.B.3.3 (MPA F-077): Post-accident sampling procedures - By letter dated June 10, 1994, and May 25, 1995, the applicant submitted the postaccident core damage estimate procedure. This was acknowledged in Section 9.3.2 of SSER 14. The staff reported its review of the procedure in Section 9.3.2 of SSER 16. The staff confirms that Item II.B.3.3 was adequately covered and the issue is closed.

II.B.4.1: Training for mitigating core damage, development of a training program - This issue was subsumed by NUREG-1021, "Operator Licensing Examiner Standards," and 10 CFR Part 55. Item II.B.4 was closed in Section 13.2.1 of the SER, even though "II.B.4.1" was not explicitly mentioned. Inspection Report 50-390,391/95-01 concluded that the Watts Bar training program was

adequate to meet 10 CFR 50.120. Although not specifically mentioned, the staff has confirmed that the substance of Item II.B.4.1 was adequately covered and the issue is closed.

II.B.4.2 (MPA F-013): Training for mitigating core damage, completion of training - In Section 13.2.2 of the SER, the staff stated that the mitigating core damage training program was acceptable. Item II.B.4.2 is specifically concerned with implementation of the program, and is not a licensing review issue. This issue is thus considered closed.

II.F.1.1 (MPA F-081): Accident monitoring procedures - Item II.F.1, "Accident Monitoring Instrumentation, was evaluated and found complete in various sections of the SER (see Table 1.1 of the SER), and in SSER 5 and SSER 6. II.F.1.1 is concerned with development of procedures to use those instruments. There is no requirement for the staff to review those procedures, except to verify their existence by inspection. This issue is thus closed without additional documentation.

II.F.2.2 (MPA F-082): Instrumentation for detection of inadequate core cooling, install primary coolant saturation meter - In Section 4.4 of SSER 10, the staff accepted the design, but stated that final acceptance is contingent upon the staff's review of the implementation letter. Implementation is scheduled to be before fuel load. The staff will track this to closure by TAC No. M93146.

II.F.2.3 (MPA F-083): Instrumentation for detection of inadequate core cooling, describe other instrumentation, install reactor vessel level instrumentation - In Section 4.4 of SSER 10, the staff accepted the design, but stated that final acceptance is contingent upon the staff's review of the implementation letter. Implementation is scheduled to be before fuel load. The staff will track this to closure by TAC No. M93147.

II.K.1.5: I&E bulletins, assure proper ESF functioning - In Section 7.3.5 of the SER and SSER 3, the staff found the applicant's response to Bulletins 79-06A and 80-06 acceptable, but made no explicit reference to II.K.1.5. Since the technical information was found acceptable, even though there was no mention of Item II.K.1.5, the item is closed.

II.K.1.10: I&E bulletins, operability of safety-related systems - In NUREG-0660, Item II.K.1.10 references I.C.2 and I.C.6 as the implementation items; both have been closed in Section 13.5.1 of the SER. The staff has subsequently confirmed that Item II.K.1.10 is closed even though it was not specifically mentioned.

II.K.1.17: I&E bulletins, low pressurizer pressure reactor trip - Neither II.K.1 nor II.K.1.17 was mentioned in the SER. However, the staff has confirmed that Sections 7.2.1 in SER addressed the pressurizer low pressure signal as one of the reactor trips, and Section 7.3.1 addressed the same as a safety injection signal. Hence the substance of II.K.1 and II.K.1.17 is present in the SER, even though the issues are not explicitly mentioned. The issue is thus closed.

II.K.3.5.A (MPA F-039): Final recommendations, B&O Task Force, propose modifications for auto trip of reactor coolant pumps - In SSER 4 (Section 15.5.4) and a letter, P. S. Tam to O. D. Kingsley, June 8, 1990, the

staff approved Item II.K.3.5, but did not specifically mention II.K.3.5.A. The staff has subsequently confirmed that no further review need be performed. Item II.K.3.5.A is considered closed.

II.K.3.5.B (MPA F-039): Final recommendations, B&O Task Force, perform modifications for auto trip of reactor coolant pumps - In SSER 4 (Section 15.5.4) and a letter, P. S. Tam to O. D. Kingsley, June 8, 1990, the staff approved Item II.K.3.5, but did not specifically mention II.K.3.5.B. The staff has subsequently confirmed that no further review need be performed. Item II.K.3.5.A is considered closed.

II.K.3.30.A: Final recommendations, B&O Task Force, program outline and schedule for small-break LOCA (SBLOCA) model - The applicant has submitted a current SBLOCA analysis, and the staff has completed its review (SSER 15, Chapter 15). Although Item II.K.3.30.A was not specifically mentioned, the staff has subsequently confirmed that the issue is closed by the staff's evaluation.

II.K.3.30.B (MPA F-057): Final recommendations, B&O Task Force, revise SBLOCA analysis model - TVA has submitted an SBLOCA analysis using the staff-approved NOTRUMP code (Section 15.5.5 of SSER 5), and the staff has completed its review of the current SBLOCA analysis (SSER 15, Chapter 15). This issue is closed.

II.K.3.30.C: Final recommendations, B&O Task Force, plant-specific SBLOCA analysis - The applicant has submitted an SBLOCA analysis using the staff-approved NOTRUMP code, and the staff has completed its review (SSER 15, Chapter 15). This issue is closed.

III.D.3.4.1: Control room habitability, identify and evaluate potential hazards - In Section 6.4 of the SER, the staff concludes that the applicant has satisfied the requirements of Item III.D.3.4 and 10 CFR Part 50, Appendix A, General Design Criterion 19. The staff confirms that the applicant has identified and evaluated potential hazards. This issue is closed.

III.D.3.4.2: Control room habitability, propose modifications - In Section 6.4 of the SER, the staff concludes that protection was acceptable, and that no modification was necessary. Although Item III.D.3.4.2 was not explicitly mentioned, the staff has confirmed that no proposal for modification need be forthcoming. This issue is closed.

C. Approved With Conditions

II.B.1.2 (MPA F-010): Reactor coolant system vents, install the vents - In SSER 12, Section 5.4.5, the staff approved the design, but stated that TVA should submit a letter upon implementation. The applicant's letter of July 25, 1995, certified complete implementation. This issue is closed.

D. Licensing Action (Safety Review) Not Applicable Because Issues Involve Implementation Only

Since the following issues are concerned only with implementation, licensing actions were neither needed nor done:

I.D.2.2 (MPA F-075): Safety parameter display system (SPDS), install the SPDS - See SSER 15, Chapter 18, for SPDS installation (implementation) audit. Since the console was installed, this issue is closed.

II.B.3.4: Post-accident sampling, complete actions - In SSER 3, Section 9.3.2, the staff stated that TVA met all 11 criteria, but needed to submit a procedure to estimate core damage. In SSER 14, Section 9.3.2, the staff acknowledged TVA's submittal of a procedure to estimate core damage, and deleted proposed License Condition 19. In SSER 16, Section 9.3.2, the staff documents the safety evaluation of the staff's review of that procedure. This issue is closed.

II.E.1.1.2 (MPA F-015): Auxiliary feedwater system evaluation, modifications - In Section 10.4.9 of the SER, the staff concluded that the system was reliable. Per TVA's May 19, 1995, letter, implementation will be complete by August 20, 1995. This issue is closed.

II.F.2.4 (MPA F-026): Instrumentation for detection of inadequate core cooling, install instrumentation - The staff completed its review in Section 4.2.3 of SSER 10. Per TVA's May 19, 1995, letter, implementation will be complete by August 13, 1995. This issue is closed.

III.D.3.4.3 (MPA F-070): Control room habitability, install modifications - In Section 6.4 of the SER, the staff concludes that protection was acceptable, and that no modification was necessary. Although III.D.3.4.2 was not explicitly mentioned, the staff has confirmed that no modification is required. This issue is closed.

E. Technical Specification Issues

Watts Bar Technical Specifications (TSs) follow the new Westinghouse Standard Technical Specifications (STSS). As of this time, Watts Bar Unit 1 TSs only exist in draft form, as transmitted to TVA by letter, F. J. Hebdon to O. D. Kingsley, January 18, 1995). As a result, requirements will exist as entries in the TSs, or in the Technical Requirements Manual (TRM), or have been eliminated generically through the STSS. All the following issues are considered complete since pertinent requirements will be imposed:

I.A.1.3.1 (MPA F-002): Shift manning, overtime limit - Requirement will be imposed in TS Section 5.2.2.

I.A.1.3.2 (MPA F-002): Shift manning, minimum shift crew - This issue was superseded by plant TSs and 10 CFR 50.54(m)(2)(i). Requirement will be imposed in TS Sections 5.1.2 and 5.2.2.

USI A-09: Anticipated transient without scram - This issue was resolved for Watts Bar (see SSER 9, Appendix W, "Safety Evaluation Report, Watts Bar Nuclear Plant, Units 1 and 2, Compliance With ATWS Rule, 10 CFR 50.62"). There is currently no existing guidance on ATWS equipment TSs. When the guidance is developed, Watts Bar TSs will be modified accordingly. This issue is closed.

GSI B-63: Isolation of low-pressure systems from reactor coolant system - The staff imposed this requirement in TS Section 3.4.14. This issue is closed.

GSI 70: PORV and block valve reliability - The staff completed its review in a letter, P. S. Tam to O. D. Kingsley, January 9, 1991 (TAC M77469). Requirement will be imposed in TS Section 3.4.11.

GSI-94 (MPA B-115): Additional low-temperature overpressure protection for light water reactors - The staff completed its review in a letter, P. S. Tam to O. D. Kingsley, January 9, 1991. Requirement will be imposed in TS Section 3.4.12.

GSI A-13 (MPA B-017): Snubber operability assurance, hydraulic snubbers - In Generic Letter (GL) 84-13, the staff informed licensees that the TS can be deleted. Requirement has been relocated to the TRM, Section 3.7.3. This issue is closed.

GSI A-13 (MPA B-022): Snubber operability assurance, mechanical snubbers - In GL 84-13, the staff informed licensees that the TS can be deleted. Requirement has been relocated to the TRM, Section 3.7.3. This issue is closed.

MPA A-023: Regulatory Guide 1.99, Revision 2 (pressurized thermal shock rule) (GL 88-11) - In TS Section 3.4.3, the staff addressed pressure/temperature limits. This issue is closed.

MPA B-024: Containment purging and venting during normal operation, guidelines for valve operability (GL 79-46) - The staff completed its review in Section 6.2.4 of SSER 5. Requirement was imposed in TS Section 3.6.3.7. This issue is closed.

F. To Be Accomplished After OL Is Issued

I.D.2.3 (MPA F-009): Safety parameter display system (SPDS), implement the system - The staff completed its review of the SPDS in SSERs 5, 6, and 15. By letter dated July 11, 1989, the applicant committed to have the SPDS operational, meeting guidance of GL 89-06, NUREG-0737, and NUREG-1342, by startup from the first refueling outage. As stated in SSER 5, the staff will impose a license condition to assure acceptable implementation. The initial fuel load, and throughout the first fuel cycle, the applicant committed to have the SPDS "functional."

MPA B-118: Individual plant examination, external events (IPEEE) (GL 88-20, Supplement 4) - By letter dated March 24, 1994, the staff accepted IPEEE submittal 120 days after 1st refueling. TAC M83693 tracks IPEEE.

MPA F-072: Safety parameter display system, response to GL 89-02. - The staff accepted this issue in Chapter 18 of SSER 5, SSER 6, and SSER 15. A proposed license condition, introduced in SSER 5, will ensure full operability by startup from first refueling outage.

G. Documentation or Other Staff Actions Needed

I.C.7.1: Nuclear steam supply system vendor review of procedures, low-power test program - In its letter of May 19, 1995, the applicant stated that the power ascension test, which is described in Section 14.2.12.2 of the FSAR and approved by the staff in SSER 14, included review of procedures by the vendor. This issue is thus closed without additional action.

I.D.1.2 (MPA-F008, F-071): Detailed control room design review (DCRDR), followup to MPA F-008 - In Chapter 18 of SSER 5, SSER 6, and SSER 15, the staff found all DCRDR actions acceptably complete. This issue is closed.

II.D.1.1: Safety and relief valve test requirements, description of test program and schedule - The staff reviewed the applicant's July 19, 1994, submittal, and closed this issue in SSER 15.

II.D.1.2 (MPA F-014): Safety and relief valve test requirements, complete testing of safety and relief valves - The staff reviewed the applicant's July 19, 1994, submittal, and closed this issue in SSER 15.

II.D.1.3 (MPA F-084): Safety and relief valve test requirements, test block valves - The staff reviewed the applicant's July 19, 1994, submittal, and closed this issue in SSER 15.

II.E.4.2.1-4 (MPA F-78): Containment isolation dependability, implement diverse isolation - The staff completed its evaluation in a letter, P. S. Tam to O. D. Kingsley, July 12, 1990 (TAC M63633), and Section 6.2.4 of the SER, SSER 5, and SSER 12. The applicant's letters of February 20 and February 25, 1985, have been incorporated into the FSAR and thus reviewed by the staff. This issue is closed.

USI A-01: Water hammer - In Appendix C of the 1982 SER, the staff specifically resolved this issue for Watts Bar. USI A-01 was resolved generically in 1984, but the generic resolution does not have an impact on the staff's previous resolution for Watts Bar. Appendix C in SSER 15 closed this USI.

USI A-11 (MPA S-007): Reactor vessel material toughness - This issue was subsumed by equivalent margin analysis, which was evaluated and closed in Section 5.3.1.1.1 of SSER 14. This USI is thus closed.

USI A-17: Systems interaction - This USI was generically resolved with publication of GL 89-18, and required no licensee action. Appendix C in SSER 15 closed the USI.

USI A-24: Equipment qualification - In Section 3.11 and Appendix C of SSER 15, the staff closed this USI.

USI A-47 (MPA B-113): Safety implications of control systems - The staff closed this technical issue by letter, P. S. Tam to O. D. Kingsley, dated October 24, 1990 (TAC No. M75017). In Appendix C of SSER 15, the staff closed this USI.

GS1 75 (MPA B-078): GL 83-28, Items 3.1.1 and 3.1.2 - This was considered complete in Section 15.3.6 of SSER 5. The applicant's letter of November 1, 1993 (after SSER 5), removed details not required by GL 83-28. The staff agrees that the letter should not reopen this issue. The issue remains closed.

GS1 75 (MPA B-087): GL 83-28, Items 3.2.1 and 3.2.2 - As reported in SSER 5, the staff first completed its review in Inspection Report 50-390, 391/86-04. TVA modified its position by letter of November 1, 1993. The staff accepted the modified position in IR 94-73; this closes the issue.

MPA A-016: Qualifications of inspection, examination, and testing and audit personnel (GL 81-01) - In SSER 5, Chapter 17, the staff clearly stated that the TVA QA program is in compliance with Regulatory Guides 1.58 and 1.146, the main substance of GL 81-01. Subsequent revisions to the TVA QA program have also been approved, and are listed in Chapter 17 of SSER 15. Although MPA A-016 or GL 81-01 was not specifically mentioned, the staff has determined that the substance is addressed in the issuances listed in SSER 15. This issue is thus closed.

MPA B-031: Deep draft pump deficiencies (Bulletin 79-15) - In Section 3.10 of SSER 4, the staff approved TVA's response and closed the issue. TVA then submitted another letter on January 24, 1992, providing additional information on preventive maintenance. The staff reviewed this in Inspection Report 50-390, 391/94-45; this issue is closed.

MPA B-063: Emergency procedures and training for station blackout events (GL 81-04) - All station blackout issues were subsumed by 10 CFR 50.63 (MPA A-44). The staff's letter dated September 9, 1993 (TAC No. M68624), found all Watts Bar station blackout fixes acceptable, with full implementation by fuel load. This issue is thus closed.

MPA B-110: Motor-operated valve testing and surveillance - By letter dated September 14, 1990 (TAC No. M75736), the staff reviewed TVA's response to Supplement 2 to GL 89-10. This issue is closed.

MPA L-208: Thermo-Lag 330-1 fire barriers (GL 92-08) - The staff is reviewing TVA's response under TAC No. M85622. Furthermore, TVA has undertaken plant-specific design and testing of Thermo-Lag; such review is ongoing under TAC No. M63648.

MPA L-304: Rod control system failures and withdrawal of rod control cluster assemblies (GL 93-04) - The staff found, by letter dated December 9, 1994, that TVA's response to the GL is acceptable (TAC No. M86877), predating the Westinghouse Owners Group (WOG) recommendations. By its letter of July 14, 1995, the applicant stated that its response was consistent with the WOG's recommendation, and thus there is no need to submit a revised response. This issue is thus closed.

MPA X-802 (BL-88-02): Rapidly propagating fatigue crack in steam generator tubes - The staff closed out this issue in a June 7, 1990, letter (TAC No. M67329). TVA stated in an August 16, 1990, letter that the staff did not consider TVA's March 1, 1989, letter, which proposed a more conservative approach. On page 29 of Inspection Report 50-390, 391/90-24, the staff reviewed all TVA submittals, including the one sent on March 29, 1989. On such basis, this issue is considered closed.

MPA X-808 (BL-88-08): Thermal stress in piping - Issue was originally closed by staff letter of September 19, 1991 (TAC No. M69706). The issue was reopened by TVA's submittal of March 29, 1994; the staff completed its review by letter dated June 26, 1995 (TAC No. M89581).

MPA X-809 (Bulletin 88-09): Thimble tube thinning in Westinghouse reactors - TVA's response of March 11, 1994, fully complies with the bulletin (TAC No. M72693). On the basis of instructions from the lead project manager

(memo, B. Buckley to PMs, September 27, 1989), there is no need for additional action. This issue is closed.

H. Licensing Action Considered Complete in Sciencetech, Inc. Report

The staff agrees with Sciencetech's assessment, and the following issues remain closed:

I.A.2.3: Administration of training program

I.B.1.2: Independent safety engineering group

I.C.2: Shift and relief turnover procedures

I.C.3: Shift supervisor responsibility

I.C.4: Control room access

I.C.5 (MPA F-006): Feedback of operating experience

I.C.6 (MPA F-007): Verifying correct performance of operating activities

I.C.7.2: Nuclear steam supply system vendor review of procedures, power ascension, and emergency procedures

I.C.8: Pilot monitoring of selected emergency procedures for near-term operating licenses

I.D.1.1 (MPA F-008): Detailed control room review (DCRDR), program plan

I.G.1.1: Preoperational and low-power testing, propose tests

I.G.1.2: Preoperational and low-power testing, analyses and procedures

I.G.1.3: Preoperational and low-power testing, perform training and evaluate results

II.B.1.1: Reactor coolant system vents design and analyses

II.B.1.3: Reactor coolant system vents, procedures governing use of the vents

II.B.2.1: Design review of plant shielding and environmental qualification of equipment for spaces/systems which may be used in post-accident operations, and radiation and shielding review

II.B.2.2: Design review of plant shielding and environmental qualification of equipment for spaces/systems which may be used in post-accident operations, corrective actions to assure access

II.B.2.3 (MPA F-011): Design review of plant shielding and environmental qualification of equipment for spaces/systems which may be used in post-accident operations, complete modifications

II.D.3: Valve position indication, install in control room

II.E.1.1.1: Auxiliary feedwater system evaluation analysis

II.E.1.2.1.A: Auxiliary feedwater system initiation and flow, short-term changes

II.E.1.2.1.B (MPA F-016): Auxiliary feedwater system initiation and flow, long-term changes

II.E.1.2.2.A: Auxiliary feedwater system initiation and flow, short-term flow indication

II.E.1.2.2.B (MPA F-017): Auxiliary feedwater system initiation and flow, long-term flow indication

II.E.3.1: Emergency power for pressurizer heaters

II.E.4.2.5: Containment isolation dependability, adjust setpoints for non-essential system isolation

II.E.4.2.6 (F-079): Containment isolation dependability, containment purge valve operability

II.E.4.2.7 (MPA F-019): Containment isolation dependability, close purge valves on high radiation signal

II.G.1: Power supplies for pressurizer relief valves, block valves, and level indicators

II.K.2.13 (MPA F-030): Orders on B&W plants, thermal/mechanical analysis

II.K.2.17 (MPA F-033): Orders on B&W plants, voiding in reactor coolant system during anticipated transients

II.K.2.19 (MPA F-034): Orders on B&W plants, benchmark analysis of sequential auxiliary feedwater flow

II.K.3.1 (MPA F-036): Final recommendations, B&O Task Force, automatic isolation of the pressurizer pilot-operated relief valve

II.K.3.2 (MPA F-037): Final recommendations, B&O Task Force, report on safety effect of pilot-operated valve isolation system

II.K.3.3 (MPA F-038): Final recommendations, B&O Task Force, report safety and relief valves failures and challenges

II.K.3.9 (MPA F-040): Final recommendations, B&O Task Force, proportional integral derivative (PID) controller modification

II.K.3.10 (MPA F-041): Final recommendations, B&O Task Force, proposed anticipatory trip modifications

II.K.3.12.A: Final recommendations, B&O Task Force, proposed modifications to install anticipatory trip

II.K.3.17 (MPA F-047): Final recommendations, B&O Task Force, report ECCS component outages

II.K.3.25.A: Final recommendations, B&O Task Force, propose modifications to pump seal cooling systems

II.K.3.25.B (MPA F-053): Final recommendations, B&O Task Force, install modifications to pump seal cooling systems

II.K.3.31 (MPA F-058): Final recommendations, B&O Task Force, compliance with 10 CFR 50.46

III.A.1.2.1 (MPAs F-063, -064, -065): Upgrade emergency support facilities

III.A.2.1 (MPA F-067): Emergency preparedness, upgrade emergency plans

III.A.2.2 (MPA F-068): Emergency preparedness; meteorological data

III.D.1.1: Primary coolant outside containment

III.D.3.3.1: In-plant iodine radiation monitoring, provide equipment to determine presence of radioiodine

III.D.3.3.2 (MPA F-069): In-plant iodine radiation monitoring, provide equipment, training, and procedures

USI A-02: Asymmetric blowdown loads on reactor primary coolant systems

USI A-03: Westinghouse steam generator tube integrity

USI A-26: Reactor vessel pressure transient protection

USI A-31: Residual heat removal shutdown requirements

USI A-36 (MPA C-010 and -015): Control of heavy loads near spent fuel

USI A-40 (MPA B-109): Seismic design criteria

USI A-44 (MPA A-022): Station blackout

USI A-45: Shutdown decay heat removal requirements (In its letter of June 14, 1995, TVA believes that since this issue was subsumed by IPEEE, which is yet to be done, this issue is still open. The staff agrees that while the technical issue may continue under IPEEE, it no longer exists under USI A-45, which is considered closed.)

USI A-48 (MPA S-003): Hydrogen control measures and effects of hydrogen burns on safety equipment

GSI 43 (MPA B-107): Reliability of air systems (In its letter of July 14, 1995, the applicant identified this issue as open because of an impending revision. The staff completed its review of that revision on July 27, 1995; thus this issue remains closed.)

GSI 51 (MPA L-913): Proposed requirements for improving the reliability of open-cycle service water systems

GSI 67.3.3 (MPA A-17): Improved accident monitoring

GSI 75 (MPA B-076): GL 83-28, Item 1.1, post-trip review program description and procedure

GSI 75 (MPA B-077): GL 83-28, Item 2.1, equipment classification and vendor interface, reactor trip system components

GSI 75 (MPA B-079): GL 83-28, Item 3.1.3, post-maintenance testing, changes to test requirements (reactor trip system components)

GSI 75 (MPA B-080): GL 83-28, Item 4.1, reactor trip system reliability, vendor-related modifications

GSI 75 (MPA B-081): GL 83-28, Items 4.2.1 and 4.2.2, reactor trip system reliability, maintenance and testing

GSI 75 (MPA B-082): GL 83-28, Item 4.3, reactor trip system reliability, design modifications - automatic actuation of shunt trip attachment

GSI 75 (MPA B-085): GL 83-28, Item 1.2, post-trip data and information capability

GSI 75 (MPA B-086): GL 83-28, Item 2.2.1, equipment classification for safety-related components

GSI 75 (MPA B-088): GL 83-28, post-maintenance testing, changes to test requirements (all other safety-related components)

GSI 75 (MPA B-090): GL 83-28, Item 4.3, reactor trip system reliability - technical specification changes, automatic actuation of shunt trip attachment

GSI 75 (MPA B-092): GL 83-28, Item 4.5.1, reactor trip system reliability - diverse trip features (system functional testing)

GSI 75 (MPA B-093): GL 83-28, Items 4.5.2 and 4.5.3, reactor trip system reliability, test alternatives, and intervals (system functional testing)

GSI 75 (MPA L-003): GL 83-28, Item 2.2.2, vendor interface for safety-related components

GSI 93 (MPA B-098): Steam binding of auxiliary feedwater pumps (GL 88-03)

GSI 99 (MPA L-817): Reactor coolant system/residual heat removal suction line valve interlock on PWRs

MPA A-025: Inservice testing reviews and schedules (GL 89-04)

MPA B-043: Cracking in feedwater system piping (Bulletin 79-13)

MPA B-066: Natural circulation cooldown (GL 81-21)

MPA B-089: GL 83-28, Items 4.2.3 and 4.2.4, life testing and replacement of reactor trip breakers

MPA B-095: Loss of residual heat removal (GL 87-12)

MPA B-096: Thinning of pipe walls in nuclear power plants (Bulletin 87-01)

MPA B-101: Boric acid corrosion of carbon steel reactor pressure boundary components in PWR plants

MPA B-111: Individual plant examinations (GL 88-20)

MPA B-117 (Bulletin 89-01, Supplement 2): Failure of Westinghouse steam generator tube mechanical plugs

MPA B-120: Reactor vessel structural integrity (GL 92-01)

MPA B-122 (Bulletin 90-01, Supplement 1): Loss of fill oil in transmitters manufactured by Rosemount

MPA B-123: Inaccuracy of motor-operated valve diagnostic equipment

MPA L-907: Power reactor safeguards contingency planning for surface vehicle bombs (GL 89-07)

MPA L-908 (GL 89-08): Erosion/corrosion-induced pipe wall thinning

MPA X-001 (Bulletin 90-01): Loss of fill oil in transmitters manufactured by Rosemount

MPA X-801 (Bulletin 88-01): Defects in Westinghouse circuit breakers (Bulletin 88-01)

MPA X-803 (Bulletin 88-01): General Electric HFA relays

MPA X-804 (Bulletin 88-04): Potential safety-related pump loss

MPA X-805 (Bulletin 88-05): Nonconforming materials supplied by Piping Supplies, Inc. at Folsom, New Jersey, and West Jersey Manufacturing Company at Williamstown, New Jersey

MPA X-810: Circuit breaker material problems (Bulletin 88-10)

MPA X-811: Thermal stratification in pressurizer surge line (Bulletin 88-11)

MPA X-901 (Bulletin 89-01): Failure of Westinghouse steam generator tube mechanical plugs

MPA X-902: Stress corrosion cracking of Anchor Darling valve bolting (Bulletin 89-02)

MPA X-903: Potential loss of required shutdown margin during refueling operations (Bulletin 89-03)

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER
(Assigned by NRC. Add Vol., Supp., Rev.,
and Addendum Numbers, if any.)

NUREG-0847
Supplement No. 16

2. TITLE AND SUBTITLE

Safety Evaluation Report Related to the Operation of
Watts Bar Nuclear Plant, Units 1 and 2

3. DATE REPORT PUBLISHED

MONTH | YEAR

September | 1995

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

Peter S. Tam et al.

6. TYPE OF REPORT

Technical

7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 00555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Same as 8. above.

10. SUPPLEMENTARY NOTES

Docket Nos. 50-390 and 50-391

11. ABSTRACT (200 words or less)

Supplement No. 16 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. 15 was issued, and (2) matters that the staff had under review when Supplement No. 15 was issued.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

Safety Evaluation Report (SER)
Watts Bar Nuclear Plant
Docket Nos. 50-390/50-391

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page)

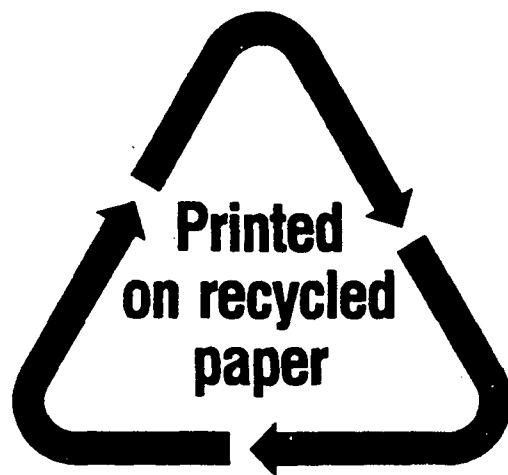
Unclassified

(This Report)

Unclassified

15. NUMBER OF PAGES

16. PRICE



Federal Recycling Program

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

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, \$300

SPECIAL FOURTH-CLASS RATE
POSTAGE AND FEES PAID
USNRC
PERMIT NO. G-67