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NUREG-0847
Supplement 22

Safety Evaluation Report

Related to the Operation of Watts Bar Nuclear Plant, Unit 2

Docket No. 50-391

Tennessee Valley Authority

Office of Nuclear Reactor Regulation

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ABSTRACT

This report supplements the safety evaluation report (SER), NUREG-0847 (June 1982), Supplement No. 21 (February 2009, Agencywide Documents Access and Management System (ADAMS) accession Number ML090570741), with respect to the application filed by the Tennessee Valley Authority (TVA), as applicant and owner, for a license to operate Watts Bar Nuclear Plant (WBN) Unit 2 (Docket No 50-391).

In its SER and Supplemental SER (SSER) Nos. 1 through 20 issued by the Office of Nuclear Reactor Regulation (NRR) of the U.S. Nuclear Regulatory Commission (NRC or the staff), the staff documented its safety evaluation and determination that WBN Unit 1 met all applicable regulations and regulatory guidance. Based on satisfactory findings from all applicable inspections, on February 7, 1996, the NRC issued a full-power operating license (OL) to WBN Unit 1, authorizing operation up to 100-percent power.

In SSER 21, the staff addressed TVA's application for a license to operate WBN Unit 2, and provided information regarding the status of the items remaining to be resolved, which were outstanding at the time that TVA deferred construction of WBN Unit 2, and were not evaluated and resolved as part of the licensing of WBN Unit 1. In this and future SSERs, the staff will document its evaluation and closure of open items in support of TVA's application for a license to operate WBN Unit 2.

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All WBN documents may be accessed using WBN docket numbers 05000390 and 05000391 for Units 1 and 2, respectively.

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

ABBREVIATIONS

AACC	American Association for Contamination Control
ABGTS	auxiliary building gas treatment system
ABI	auxiliary building isolation
ABSCE	auxiliary building secondary containment enclosure
AC	alternating current
ACAS	auxiliary control air system
ACR	auxiliary control room
ADAMS	Agencywide Documents Access and Management System
AFW	auxiliary feedwater
AFWP	auxiliary feedwater pump
AMSAC	anticipated transient without scram mitigation system actuation circuitry
ANSI	American National Standards Institute
ANS	American Nuclear Society, or Alerting and Notification System
AOV	air-operated valve
APS	auxiliary power system
ART	adjusted reference temperature
ASB	Auxiliary Systems Branch (of NRR)
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	anticipated transient without scram
BL	bulletin
BTP	Branch Technical Position
BWG	Birmingham Wire Gauge
BWR	boiling-water reactor
CAP	corrective action program
CAS	centrla alarm station
CCS	condensate cleanup system
CCW	circulating cooling water
CDWE	condensate demineralizer waste evaporator
CECC	Central Emergency Control Center (of TVA)
cfm	cubic feet per minute
CFR	Code of Federal Regulations
CIV	containment isolation valve
COMS	cold overpressure mitigation system
CRDM	control rod drive mechanism
CSST	common station service transformer
CST	condensate storage tank
CVCS	chemical volume and control system
CVI	containment vent isolation
DBA	design basis accident
DBT	design basis threat
DC	direct current
DCRDR	detailed control room design review
DMBW	dissimilar metal butt welds
DVR	degraded voltage relay
EAL	emergency action level

ECCS	emergency core cooling system
ECS	environmental control system
EDCR	Engineeering Document Construction Release
EDG	emergency diesel generator
EFPY	effective full power year
EGTS	emergency gas treatment system
EOF	emergency operations facility
EOL	end of life
EPA	Environmental Protection Agency or electrical penetration assemblies
EPIP	emergency plan implementing procedure
EPRI	Electric Power Research Institute
EPZ	emergency planning zone
EQ	environmental qualification
ERCW	essential raw cooling water
ERDS	emergency response data system
ERO	emergency response organization
ESF	engineered safety feature
ESW	extremely severe weather
ETA	evacuation time estimate
FEMA	Federal Emergency Management Agency
FHA	fuel handling accident
FSAR	final safety analysis report
FW	feedwater
GDC	general design criterion/criteria
GI	generic issue
GL	generic letter
gpm	gallons per minute
HDCI	High Duty Core Index
HAS	hydrogen analyzer system
HED	human engineering deficiency
HEPA	high efficiency air particulate
HMS	hydrogen mitigation system
HVAC	heating, ventilation, and air conditioning
ICS	integrated computer system
IE	Office of Inspection and Enforcement
IEB	Office of Inspection and Enforcement Bulletin
IEEE	Institute of Electrical and Electronics Engineers
IESNA	Illuminating Engineering Society of North America
IFR	Interim Finding Report
INEL	Idaho National Engineering Laboratory
IST	inservice testing
kV	kilovolt
kVA	kilovolt ampere
kW	kilowatt
LCC	lower compartment cooler
LCV	level control valve
LLEA	local law enforcement agency
LOCA	loss-of-coolant accident

LOOP	loss of offsite power
LOV	loss of voltage
LPZ	low-population zone
LTC	load tap changer
LTOP	low-temperature overpressure protection
LWR	light-water reactor
MCCB	molded case circuit breaker
MCES	main condeser evacuation system
MCR	main control room
MCRHZ	main control room habitability zone
MEB	Mechanical Engineering Branch (of NRR)
MJRERP	Multi-Jurisdictional Radiological Emergency Response Plan
MOU	memorandum of understanding
MOV	motor operated valve
mph	miles per hour
MSIV	main steam isolation valve
MSLB	main steam line break
MTEB	Materials Engineering Branch (of NRR)
MVA	megavolt-ampere
MWD/MTU	megawatt days per metric ton unit (or uranium)
MWt	megawatts thermal
NCDC	National Climatic Data Center
NDE	nondestructive examination
NEC	not elsewhere classified
NEI	Nuclear Energy Institute
NGDC	New Generation Development and Construction
NPP	Nuclear Performance Plan
NP-REP	Nuclear Power Radiological Emergency Plan
NQA	nuclear quality assurance
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSSS	nuclear steam supply system
NUREG	report prepared by NRC staff
OCA	owner controlled area
OE	operating experience
OI	operating instruction
OL	operating license
OM Code	ASME Code for Operation and Maintenance of Nuclear Power Plants
OSC	operations support center
PA	protected area
PORC	plant operations review committee
PORV	power-operated relief valve
ppb	parts per billion
PRT	pressurizer relief tank
PSHT	preservice system hydrostatic test
psia	pounds per square inch absolute
psig	pounds per square inch gauge
PSP	Physical Security Plan
PTLR	Pressure and Temperature Limits Report

PTS	pressurized thermal shock
PWR	pressurized-water reactor
PWSCC	primary water stress corrosion cracking
QA	quality assurance
RAI	request for additional information
RBPVS	reactor building purge ventilation system
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RG	Regulatory Guide
RHR	residual heat removal
RPV	reactor pressure vessel
RV	reactor vessel
RWST	refueling water storage tank
SAS	secondary alarm station
SAT	system approach to training
SBO	station blackout
SCC	stress corrosion cracking
SCP	Safeguards Contingency Plan
SE	safety evaluation
SER	safety evaluation report, NUREG-0847
SG	steam generator
SGBS	steam generator blowdown system
SI	safety injection
SNB	subcooled nucleate boiling
SP	special program
SQN	Sequoyah Nuclear Plant
SRM	Staff Requirements Memorandum
SRP	Standard Review Plan, NUREG-0800
SSC	structures, systems, and components
SSER	Supplemental SER
Std	Standard
SV	safety valve
SW	severe weather
TDAFW	turbine-driven auxiliary feedwater
TDAFWP	turbine-driven auxiliary feedwater pump
TGSS	turbine gland sealing system
TI	technical or temporary instruction
TIPTOP	Turbine Integrity Program with Turbine Overspeed Protection
TMI	Three Mile Island
TPBAR	tritium production burnable absorber rod
T&QP	Training and Qualification Plan
TS	technical specification
TSC	Technical Support Center
TSTF	Technical Specification Task Force
TVA	Tennessee Valley Authority
UHS	ultimate heat sink
UPS	uninterruptible power supply
USE	upper shelf energy

UT	ultrasonic test
WBN	Watts Bar Nuclear Plant
WBN REP	Watts Bar Nuclear Plant Radiological Emergency Plan
WBNPP	Watts Bar Nuclear Performance Plan
WCAP	Westinghouse Commercial Atomic Power (report)

1 INTRODUCTION AND DISCUSSION

1.1 Introduction

The Watts Bar Nuclear Plant (WBN or Watts Bar) is owned by the Tennessee Valley Authority (TVA) and is located in southeastern Tennessee approximately 50 miles northeast of Chattanooga. The facility consists of two Westinghouse-designed four-loop pressurized-water reactors (PWRs) within ice condenser containments.

In June 1982, the Nuclear Regulatory Commission staff (NRC staff or staff) issued a safety evaluation report (SER), NUREG-0847, "Safety Evaluation Report related to the operation of Watts Bar Nuclear Plant Units 1 and 2," regarding TVA's application for licenses to operate WBN Units 1 and 2. In Supplements 1 through 20 to the SER, the NRC staff concluded that WBN Unit 1 met all applicable regulations and regulatory guidance and on February 7, 1996, the NRC issued an operating license (OL) to Unit 1. TVA did not complete WBN Unit 2, and the NRC did not make conclusions regarding it.

On March 4, 2009, TVA submitted an updated application in support of its request for an OL for WBN Unit 2, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, "Domestic Licensing of Production and Utilization Facilities."

In SSER 21, the staff provided information regarding the status of the WBN Unit 2 items that remain to be resolved, which were outstanding at the time that TVA deferred construction of Unit 2, and which were not evaluated and resolved as part of the licensing of WBN Unit 1. In this and future SER Supplements (SSER), the staff will document its evaluation and closure of open items in support of TVA's application for a license to operate WBN Unit 2.

The format of this document is consistent with the format and scope outlined in the Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition (NUREG-0800)" dated July 1981 (SRP, NUREG-0800). The staff added additional chapters to address the overall assessment of the facility, Nuclear Performance Plan issues, and other generic regulatory topics.

Each of the sections and appendices of this supplement is numbered the same as the SER section that is being updated, and the discussions are supplementary to, and not in lieu of, the discussion in the SER, unless otherwise noted. For example, Appendix A continues the chronology of the safety review and Appendix E continues to list the principal contributors to this supplement. Appendix HH has been added to include an Action Items Table. This table provides a status of all the open items, confirmatory issues, and proposed license conditions that must be resolved prior to completion of an NRC finding of reasonable assurance on the OL application for WBN Unit 2. The staff will maintain the Action Item Table and revise Appendix HH in future SSERs, as appropriate, and new appendices will be added as necessary.

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1.7 Summary of Outstanding Issues

The staff documented its previous review and conclusions regarding the OL application for WBN Unit 1 in the SER (NUREG-847) and its supplements 1 through 20. Based on these reviews, the staff issued an OL for WBN Unit 1 in 1996. In the SER and SSERs 1 through 20, the staff also reviewed and approved certain topics for WBN Unit 2, though no final conclusions were made regarding an OL for WBN Unit 2. To establish the remaining scope and the regulatory framework for the staff's review of an OL for WBN Unit 2, the staff reviewed the SER and SSERs 1 thorough 20. Based on this review, the staff identified "resolved" topics (i.e., out of scope for review) and "open" topics (i.e., in scope for staff review) for WBN Unit 2. Where it was not clear whether the SER topic applied to Unit 2 or not, the staff conservatively identified it as "open" pending further evaluation. It should be noted that these were not technical evaluations of each topic; rather, it was a status review to determine whether the topic was "open" or "resolved." The staff documented this evaluation in SSER 21 as the baseline for resumption of the review of the OL application for Unit 2. Thus, SSER 21 reflects the status of the staff's review of WBN Unit 2 up to 1995. The staff notes that a subsequent, more detailed assessment may find some topics conservatively identified in the initial assessment as "open" that should be redefined as "closed." Conversely, the NRC staff notes that there may be circumstances that could result in the need to reopen some previously closed topic areas that may have been adequately documented and that are considered closed in SSER 21. Such cases will be identified by a foot note in future SSERs to document that previous "open" topics have been re-categorized as "closed" without requiring further review, or vice versa.

The SER and SSERs 1 through 20 evaluated the changes to the final safety analysis report (FSAR) until Amendment 91. FSAR Amendment 91 was the initial licensing basis for WBN Unit 1. At this time, the FSAR was applicable to both Units 1 and 2. As part of its updated OL application for WBN Unit 2, TVA split the FSAR Amendment 91 into two separate FSARs for WBN Units 1 and 2. TVA has submitted WBN Unit 2 FSAR Amendments 92 through 102 to address the "open" topics in support of its OL application for WBN Unit 2. These FSAR amendments reflect changes that have occurred since 1995. These FSAR amendments are currently under staff review. The staff's review of these FSAR changes will be documented in SSER 22 and future supplements.

Additional general topics, e.g., financial qualifications that were not included in SSER 21, but that should be resolved prior to issuance of an OL, are also identified in this supplement.

SSER 21 provided the table below documenting the status of each SER topic. The relevant document in which the topic was last addressed is shown in parenthesis. This table will be maintained in this and future supplements to reflect the updated status of review for each topic.

ISSUE STATUS TABLE

	Issue	Status		Section	Note
(1)	Site Envelope			2	
(2)	Geography and Demography	Resolved	(SSER 22)	2.1	
(3)	Site Location and Description	Resolved	(SER) (SSER 22)	2.1.1	3
(4)	Exclusion Area Authority and Control	Resolved	(SER) (SSER 22)	2.1.2	3
(5)	Population Distribution	Resolved	(SER) (SSER 22)	2.1.3	
(6)	Conclusions	Resolved	(SER) (SSER 22)	2.1.4	
(7)	Nearby Industrial, Transportation, and Military Facilities	Resolved	(SSER 22)	2.2	
(8)	Transportation Routes	Resolved	(SER) (SSER 22)	2.2.1	
(9)	Nearby Facilities	Resolved	(SER) (SSER 22)	2.2.2	
(10)	Conclusions	Resolved	(SER) (SSER 22)	2.2.3	
(11)	Meteorology		(SER) (SSER 22)	2.3	
(12)	Regional Climatology	Resolved	(SER) (SSER 22)	2.3.1	
(13)	Local Meteorology	Resolved	(SER) (SSER 22)	2.3.2	
(14)	Onsite Meteorological Measurements Program	Resolved	(SER) (SSER 22)	2.3.3	
(15)	Short-Term (Accident) Atmospheric Diffusion Estimates	Resolved	(SER) (SSER 14) (SSER 22)	2.3.4	
(16)	Long-Term (Routine) Diffusion Estimates	Resolved	(SER) (SSER 14) (SSER 22)	2.3.5	
(17)	Hydrologic Engineering			2.4	
(18)	Introduction	Resolved	(SER)	2.4.1	
(19)	Hydrologic Description	Resolved	(SER)	2.4.2	
(20)	Flood Potential	Resolved	(SER)	2.4.3	
(21)	Local Intense Precipitation in Plant Area	Resolved	(SER)	2.4.4	1
(22)	Roof Drainage	Resolved	(SER)	2.4.5	1
(23)	Ultimate Heat Sink	Resolved	(SER)	2.4.6	
(24)	Groundwater	Resolved	(SER)	2.4.7	1
(25)	Design Basis for Subsurface Hydrostatic Loading	Open (NRR)	(SER) (SSER 3)	2.4.8	
(26)	Transport of Liquid Releases	Resolved	(SER) (SSER 22)	2.4.9	2

	Issue	Status		Section	Note
(27)	Flooding Protection Requirements	Open (Inspection)	(SER)	2.4.10	
(28)	Geological, Seismological, and Geotechnical Engineering	Resolved	(SER)	2.5	
(29)	Geology	Resolved	(SER)	2.5.1	
(30)	Seismology	Resolved	(SER)	2.5.2	
(31)	Surface Faulting	Resolved	(SER)	2.5.3	
(32)	Stability of Subsurface Materials and Foundations	Resolved	(SER) (SSER 3) (SSER 9) (SSER 11)	2.5.4	
(33)	Stability of Slopes	Resolved	(SER)	2.5.5	
(34)	Embankments and Dams	Resolved	(SER) (SSER 22)	2.5.6	
(35)	References		(SER) (SSER 22)	2.6	
(36)	Design Criteria - Structures, Components, Equipment, and Systems			3	
(37)	Introduction			3.1	
(38)	Conformance With General Design Criteria	Resolved	(SER)	3.1.1	
(39)	Conformance With Industry Codes and Standards	Resolved	(SER)	3.1.2	
(40)	Classification of Structures, Systems and Components	Resolved	(SSER 14) (SSER 22)	3.2	
(41)	Seismic Classifications	Resolved	(SER) (SSER 3) (SSER 5) (SSER 6) (SSER 8)	3.2.1	
(42)	System Quality Group Classification	Open (NRR)	(SER) (SSER 3) (SSER 6) (SSER 7) (SSER 9) (SSER 22)	3.2.2	
(43)	Wind and Tornado Loadings			3.3	
(44)	Wind Loading	Resolved	(SER)	3.3.1	
(45)	Tornado Loading	Resolved	(SER)	3.3.2	
(46)	Flood Level (Flood) Design			3.4	
(47)	Flood Protection	Resolved	(SER)	3.4.1	
(48)	Missile Protection			3.5	
(49)	Missile Selection and Description	Resolved	(SER) (SSER 9) (SSER 14) (SSER 22)	3.5.1	

	Issue	Status		Section	Note
(50)	Structures, Systems, and Components to be Protected from Externally Generated Missiles	Resolved	(SER) (SSER 2) (SSER 22)	3.5.2	
(51)	Barrier Design Procedures	Resolved	(SER)	3.5.3	
(52)	Protection Against the Dynamic Effects Associated with the Postulated Rupture of Piping	Open (NRR)	(SER) (SSER 6) (SSER 11)	3.6	
(53)	Plant Design for Protection Against Postulated Piping Failures in Fluid System Outside Containment	Resolved	(SER) (SSER 14) (SSER 22)	3.6.1	
(54)	Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	Resolved	(SER) (SSER 14) (SSER 22)	3.6.2	3
(55)	Leak-Before-Break Evaluation Procedures	Open (NRR)	(SSER 5) (SSER 12) (SSER 22)	3.6.3	
(56)	Seismic Design	Resolved	(SER) (SSER 6)	3.7	2
(57)	Seismic Input	Resolved	(SER) (SSER 6) (SSER 9) (SSER 16)	3.7.1	2
(58)	Seismic Analysis	Resolved	(SER) (SSER 6) (SSER 8) (SSER 11) (SSER 16)	3.7.2	2
(59)	Seismic Subsystem Analysis	Resolved	(SER) (SSER 6) (SSER 7) (SSER 8) (SSER 9) (SSER 12) (SSER 22)	3.7.3	
(60)	Seismic Instrumentation	Resolved	(SER)	3.7.4	1
(61)	Design of Seismic Category I Structures	Resolved	(SER) (SSER 9)	3.8	2
(62)	Steel Containment	Resolved	(SER) (SSER 3)	3.8.1	
(63)	Concrete and Structural Steel Internal Structures	Resolved	(SER) (SSER 7)	3.8.2	
(64)	Other Seismic Category I Structures	Open (NRR)	(SER) (SSER 14) (SSER 16)	3.8.3	
(65)	Foundations	Resolved	(SER)	3.8.4	
(66)	Mechanical Systems and Components	Resolved	(SER)	3.9	

	Issue	Status		Section	Note
(67)	Special Topics for Mechanical Components	Resolved	(SER) (SSER 6) (SSER 13) (SSER 22)	3.9.1	
(68)	Dynamic Testing and Analysis of Systems, Components, and Equipment	Resolved	(SER) (SSER 14) (SSER 22)	3.9.2	
(69)	ASME Code Class 1, 2, and 3 Components, Component Structures, and Core Support Structures	Resolved	(SER) (SSER 3) (SSER 4) (SSER 6) (SSER 7) (SSER 8) (SSER 15) (SSER 22)	3.9.3	
(70)	Control Rod Drive Systems	Resolved	(SER)	3.9.4	
(71)	Reactor Pressure Vessel Internals	Resolved	(SER)	3.9.5	
(72)	Inservice Testing of Pumps and Valves	Open (NRR)	(SER) (SSER 5) (SSER 12) (SSER 14) (SSER 18) (SSER 20) (SSER 22)	3.9.6	
(73)	Seismic and Dynamic Qualification of Seismic Category I Mechanical and Electrical Equipment	Open (NRR)	(SER) (SSER 1) (SSER 3) (SSER 4) (SSER 5) (SSER 6) (SSER 8) (SSER 9)	3.10	
(74)	Environmental Qualification of Mechanical and Electrical Equipment	Open (NRR)	(SSER 15) (SSER 22)	3.11	
(75)	Threaded Fasteners — ASME Code Class 1, 2, and 3	Resolved	(SSER 22)	3.13	
(76)	Reactor			4	
(77)	Introduction		(SER)	4.1	
(78)	Fuel System Design			4.2	
(79)	Description	Open (NRR)	(SER) (SSER 13)	4.2.1	
(80)	Thermal Performance	Open (NRR)	(SER) (SSER 2)	4.2.2	
(81)	Mechanical Performance	Open (NRR)	(SER) (SSER 2) (SSER 10) (SSER 13)	4.2.3	

	Issue	Status		Section	Note
(82)	Surveillance	Resolved	(SER) (SSER 2)	4.2.4	
(83)	Fuel Design Considerations	Open (NRR)	(SER)	4.2.5	
(84)	Nuclear Design			4.3	
(85)	Design Basis	Open (NRR)	(SER) (SSER 13)	4.3.1	
(86)	Design Description	Open (NRR)	(SER) (SSER 13) (SSER 15)	4.3.2	
(87)	Analytical Methods	Open (NRR)	(SER)	4.3.3	
(88)	Summary of Evaluation Findings	Open (NRR)	(SER)	4.3.4	
(89)	Thermal-Hydraulic Design			4.4	
(90)	Performance in Safety Criteria	Resolved	(SER)	4.4.1	
(91)	Design Bases	Open (NRR)	(SER) (SSER 12)	4.4.2	
(92)	Thermal-Hydraulic Design Methodology	Open (NRR)	(SER) (SSER 6) (SSER 8) (SSER 12) (SSER 13) (SSER 16) SE dated 6/13/89	4.4.3	
(93)	Operating Abnormalities	Open (NRR)	(SER) (SSER 13)	4.4.4	
(94)	Loose Parts Monitoring System	Open (NRR)	(SER) (SSER 3) (SSER 5) (SSER 16)	4.4.5	
(95)	Thermal-Hydraulic Comparison	Resolved	(SER)	4.4.6	
(96)	N-1 Loop Operation	Open (Inspection)	(SER)	4.4.7	
(97)	Instrumentation for Inadequate Core Cooling Detection (TMI Action Item II.F.2)	Open (NRR)	(SER) (SSER 10)	4.4.8	
(98)	Summary and Conclusion	Open (NRR)	(SER)	4.4.9	
(99)	Reactor Materials			4.5	
(100)	Control Rod Drive Structural Materials	Resolved	(SER)	4.5.1	1
(101)	Reactor Internals and Core Support Materials	Resolved	(SER)	4.5.2	
(102)	Functional Design of Reactivity Control Systems	Resolved	(SER)	4.6	
(103)	Reactor Coolant System and Connected Systems			5	
(104)	Summary Description	Resolved	(SER) (SSER 5) (SSER 6)	5.1	2

	Issue	Status		Section	Note
(105)	Integrity of Reactor Coolant Pressure Boundary			5.2	
(106)	Compliance with Codes and Code Cases	Resolved	(SER) (SSER 22)	5.2.1	
(107)	Overpressurization Protection	Resolved	(SER) (SSER 2) (SSER 15)	5.2.2	
(108)	Reactor Coolant Pressure Boundary Materials	Resolved	(SER) (SSER 22)	5.2.3	
(109)	Reactor Coolant System Pressure Boundary Inservice Inspection and Testing	Open (NRR)	(SER) (SSER 10) (SSER 12) (SSER 15) (SSER 16)	5.2.4	
(110)	Reactor Coolant Pressure Boundary Leakage Detection	Resolved	(SER) (SSER 9) (SSER 11) (SSER 12) (SSER 22)	5.2.5	
(111)	Reactor Vessel and Internals Modeling			5.2.6	
(112)	Reactor Vessel			5.3	
(113)	Reactor Vessel Materials	Open (NRR)	(SER) (SSER 11) (SSER 14) (SSER 22)	5.3.1	
(114)	Pressure-Temperature Limits	Open (NRR)	(SER) (SSER 16) (SSER 22)	5.3.2	
(115)	Reactor Vessel Integrity	Open (NRR)	(SER) (SSER 22)	5.3.3	
(116)	Component and Subsystem Design			5.4	
(117)	Reactor Coolant Pumps	Resolved	(SER) (SSER 22)	5.4.1	2
(118)	Steam Generators	Resolved	(SER) (SSER 1) (SSER 4) (SSER 22)	5.4.2	
(119)	Residual Heat Removal System	Open (NRR)	(SER) (SSER 2) (SSER 5) (SSER 10) (SSER 11)	5.4.3	
(120)	Pressurizer Relief Tank	Resolved	(SER) (SSER 22)	5.4.4	
(121)	Reactor Coolant System Vents (TMI Action Item II.B.1)	Open (Inspection)	(SER) (SSER 2) (SSER 12)	5.4.5	

	Issue	Status		Section	Note
(122)	Engineered Safety Features			6	
(123)	Engineered Safety Feature Materials			6.1	
(124)	Metallic Materials	Resolved	(SER)	6.1.1	
(125)	Organic Materials	Resolved	(SER) (SSER 22)	6.1.2	
(126)	Postaccident Emergency Cooling Water Chemistry	Resolved	(SER) (SSER 22)	6.1.3	
(127)	Containment Systems			6.2	
(128)	Containment Functional Design	Resolved	(SER) (SSER 3) (SSER 5) (SSER 7) (SSER 12) (SSER 14) (SSER 15) (SSER 22)	6.2.1	
(129)	Containment Heat Removal Systems	Resolved	(SER) (SSER 7) (SSER 22)	6.2.2	
(130)	Secondary Containment Functional Design	Resolved	(SER) (SSER 18) (SSER 22)	6.2.3	
(131)	Containment Isolation Systems	Resolved	(SER) (SSER 3) (SSER 5) (SSER 7) (SSER 12) (SSER 22)	6.2.4	
(132)	Combustible Gas Control Systems	Resolved	(SER) (SSER 4) (SSER 5) (SSER 8) (SSER 22)	6.2.5	
(133)	Containment Leakage Testing	Open (NRR)	(SER) (SSER 4) (SSER 5) (SSER 19) (SSER 22)	6.2.6	
(134)	Fracture Prevention of Containment Pressure Boundary	Resolved	(SER) (SSER 4)	6.2.7	1
(135)	Emergency Core Cooling System	Resolved	(SER)	6.3	1
(136)	System Design	Open (NRR)	(SER) (SSER 6) (SSER 7) (SSER 11)	6.3.1	
(137)	Evaluation	Resolved	(SER) (SSER 5)	6.3.2	1

	Issue	Status		Section	Note
(138)	Testing	Open (NRR)	(SER) (SSER 2) (SSER 9)	6.3.3	
(139)	Performance Evaluation	Resolved	(SER)	6.3.4	
(140)	Conclusions	Open (NRR)	(SER)	6.3.5	
(141)	Control Room Habitability	Resolved	(SER) (SSER 5) (SSER 11) (SSER 16) (SSER 18 (SSER 22)	6.4	
(142)	Engineered Safety Feature (ESF) Filter Systems			6.5	
(143)	ESF Atmosphere Cleanup System	Resolved	(SER) (SSER 5) (SSER 22)	6.5.1	
(144)	Fission Product Cleanup System	Resolved	(SER)	6.5.2	1
(145)	Fission Product Control System	Open (NRR)	(SER) (SSER 22)	6.5.3	
(146)	Ice Condenser as a Fission Product Cleanup System	Resolved	(SER)	6.5.4	1
(147)	Inservice Inspection of Class 2 and 3 Components	Open (NRR)	(SER) (SSER 10) (SSER 15)	6.6	
(148)	Instrumentation and Controls			7	
(149)	Introduction			7.1	
(150)	General	Open (NRR)	(SER) (SSER 13) (SSER 16)	7.1.1	
(151)	Comparison with Other Plants	Resolved	(SER) (SSER 22)	7.1.2	1
(152)	Design Criteria	Open (NRR)	(SER) (SSER 4) (SSER 15)	7.1.3	
(153)	Reactor Trip System	Resolved	(SER)	7.2	
(154)	System Description	Open (NRR)	(SER) (SSER 13) (SSER 15)	7.2.1	
(155)	Manual Trip Switches	Resolved	(SER) (SSER 22)	7.2.2	1
(156)	Testing of Reactor Trip Breaker Shunt Coils	Resolved	(SER) (SSER 22)	7.2.3	1
(157)	Anticipatory Trips	Resolved	(SER)	7.2.4	
(158)	Steam Generator Water Level Trip	Open (NRR)	(SER) (SSER 2) (SSER 14)	7.2.5	
(159)	Conclusions	Open (NRR)	(SER) (SSER 13)	7.2.6	

	Issue	Status		Section	Note
(160)	Engineered Safety Features System	Open (NRR)	(SER) (SSER 13)	7.3	
(161)	System Description	Open (NRR)	(SER) (SSER 13) (SSER 14)	7.3.1	
(162)	Containment Sump Level Measurement	Resolved	(SER) (SSER 2)	7.3.2	
(163)	Auxiliary Feedwater Initiation and Control	Resolved	(SER) (SSER 22)	7.3.3	1
(164)	Failure Modes and Effects Analysis	Resolved	(SER)	7.3.4	
(165)	IE Bulletin 80-06	Open (Inspection)	(SER) (SSER 3)	7.3.5	
(166)	Conclusions	Open (NRR)	(SER) (SSER 13)	7.3.6	
(167)	Systems Required for Safe Shutdown			7.4	
(168)	System Description	Resolved	(SER)	7.4.1	
(169)	Safe Shutdown from Auxiliary Control Room	Open (NRR)	(SER) (SSER 7)	7.4.2	
(170)	Conclusions	Resolved	(SER)	7.4.3	
(171)	Safety-Related Display Instrumentation			7.5	
(172)	System Description	Resolved	(SER)	7.5.1	
(173)	Post-accident Monitoring System	Open (NRR)	(SER) (SSER 7) (SSER 9) (SSER 14) (SSER 15)	7.5.2	
(174)	IE Bulletin 79-27	Open (Inspection)	(SER)	7.5.3	
(175)	Conclusions	Open (Inspection)	(SER)	7.5.4	
(176)	All Other Systems Required for Safety			7.6	
(177)	System Description	Resolved	(SER)	7.6.1	
(178)	Residual Heat Removal System Bypass Valves	Resolved	(SER)	7.6.2	
(179)	Upper Head Injection Manual Control	Resolved	(SER)	7.6.3	
(180)	Protection Against Spurious Actuation of Motor-Operated Valves	Resolved	(SER)	7.6.4	
(181)	Overpressure Protection during Low Temperature Operation	Resolved	(SER) (SSER 4)	7.6.5	
(182)	Valve Power Lockout	Resolved	(SER)	7.6.6	
(183)	Cold Leg Accumulator Valve Interlocks and Position Indication	Resolved	(SER)	7.6.7	
(184)	Automatic Switchover From Injection to Recirculation Mode	Resolved	(SER)	7.6.8	

	Issue	Status		Section	Note
(185)	Conclusions	Resolved	(SER) (SSER 4)	7.6.9	
(186)	Control Systems Not Required for Safety			7.7	
(187)	System Description	Resolved	(SER)	7.7.1	
(188)	Safety System Status Monitoring System	Resolved	(SER) (SSER 7) (SSER 13)	7.7.2	
(189)	Volume Control Tank Level Control System	Resolved	(SER)	7.7.3	
(190)	Pressurizer and Steam Generator Overfill	Resolved	(SER)	7.7.4	
(191)	IE Information Notice 79-22	Resolved	(SER)	7.7.5	
(192)	Multiple Control System Failures	Resolved	(SER)	7.7.6	
(193)	Conclusions	Resolved	(SER)	7.7.7	
(194)	Anticipated Transient Without Scram Mitigation System Actuation Circuitry (AMSAC)	Open (NRR)	(SSER 9) (SSER 14)	7.7.8	
(195)	NUREG-0737 Items	Resolved	(SER)	7.8	
(196)	Relief and Safety Valve Position Indication (TMI Action Item II.D.3)	Open (NRR)	(SER) (SSER 5) (SSER 14)	7.8.1	
(197)	Auxiliary Feedwater System Initiation and Flow Indication (TMI Action Item II.E.1.2)	Open (Inspection)	(SER)	7.8.2	
(198)	Proportional Integral Derivative Control Modification (TMI Action Item II.K.3.9)	Open (Inspection)	(SER)	7.8.3	
(199)	Proposed Anticipatory Trip Modification (TMI Action Item II.K.3.10)	Open (Inspection)	(SER) (SSER 4)	7.8.4	
(200)	Confirm Existence of Anticipatory Reactor Trip Upon Turbine Trip (TMI Action Item II.K.3.12)	Resolved	(SER)	7.8.5	
(201)	Data Communication Systems			7.9	
(202)	Electric Power Systems			8	
(203)	General	Open (NRR)	(SER) (SSER 22)	8.1	
(204)	Offsite Power System		(SER) (SSER 22)	8.2	
(205)	Compliance with GDC 5	Open (NRR)	(SER) (SSER 13) (SSER 22)	8.2.1	

	Issue	Status		Section	Note
(206)	Compliance with GDC 17	Open (NRR)	(SER) (SSER 2) (SSER 3) (SSER 13) (SSER 14) (SSER 15 (SSER 22)	8.2.2	
(207)	Compliance with GDC 18	Resolved	(SER) (SSER 22)	8.2.3	
(208)	Evaluation Findings	Open (NRR)	(SER) (SSER 22)	8.2.4	
(209)	Onsite Power Systems	Resolved	(SER) (SSER 10) (SSER 19) (SSER 22)	8.3	
(210)	Onsite AC Power System Compliance with GDC 17	Open (NRR)	(SER) (SSER 2) (SSER 7) (SSER 9) (SSER 10) (SSER 13) (SSER 14) (SSER 18) (SSER 20) (SSER 22)	8.3.1	
(211)	Onsite DC System Compliance with GDC 17	Open (NRR)	(SER) (SSER 2) (SSER 3) (SSER 13) (SSER 14) (SSER 22)	8.3.2	
(212)	Common Electrical Features and Requirements	Resolved	(SER) (SSER 2) (SSER 3) (SSER 7) (SSER 13) (SSER 14) (SSER 15) (SSER 16) (SSER 22)	8.3.3	

	Issue	Status		Section	Note
(213)	Evaluation Findings	Open (NRR)	(SER) (SSER 2) (SSER 3) (SSER 7) (SSER 13) (SSER 14) (SSER 15) (SSER 16) (SSER 22)	8.3.4	
(214)	Station Blackout	Open (NRR)	(SSER 22)	8.4	
(215)	Auxiliary Systems	Resolved	(SER) (SSER 10)	9	
(216)	Fuel Storage Facility			9.1	
(217)	New-Fuel Storage	Resolved	(SER)	9.1.1	1
(218)	Spent-Fuel Storage	Resolved	(SER) (SSER 5) (SSER 15) (SSER 16) (SSER 22)	9.1.2	
(219)	Spent Fuel Pool Cooling and Cleanup System	Open (NRR)	(SER) (SSER 11) (SSER 15)	9.1.3	
(220)	Fuel-Handling System	Open (NRR)	(SER) (SSER 3) (SSER 13) (SSER 22)	9.1.4	
(221)	Water Systems			9.2	
(222)	Essential Raw Cooling Water and Raw Cooling Water System	Open (NRR)	(SER) (SSER 9) (SSER 10) (SSER 18)	9.2.1	
(223)	Component Cooling System (Reactor Auxiliaries Cooling Water System)	Open (NRR)	(SER) (SSER 5)	9.2.2	
(224)	Demineralized Water Makeup System	Resolved	(SER) (SSER 22)	9.2.3	
(225)	Potable and Sanitary Water Systems	Resolved	(SER) (SSER 9) (SSER 22)	9.2.4	
(226)	Ultimate Heat Sink	Resolved	(SER)	9.2.5	
(227)	Condensate Storage Facilities	Resolved	(SER) (SSER 12) (SSER 22)	9.2.6	
(228)	Process Auxiliaries			9.3	
(229)	Compressed Air System	Resolved	(SER) (SSER 22)	9.3.1	1

	Issue	Status		Section	Note
(230)	Process Sampling System	Open (NRR)	(SER) (SSER 3) (SSER 5) (SSER 14) (SSER 16)	9.3.2	
(231)	Equipment and Floor Drainage System	Resolved	(SER) (SSER 22)	9.3.3	3
(232)	Chemical and Volume Control System	Resolved	(SER) (SSER 22)	9.3.4	3
	Heat Tracing		(SSER 22)	9.3.8	
(233)	Heating, Ventilation, and Air Conditioning Systems			9.4	
(234)	Control Room Area Ventilation System	Resolved	(SER) (SSER 9) (SSER 22)	9.4.1	
(235)	Fuel-Handling Area Ventilation System	Resolved	(SER) (SSER 22)	9.4.2	
(236)	Auxiliary Building and Radwaste Area Ventilation System	Resolved	(SER) (SSER 22)	9.4.3	
(237)	Turbine Building Area Ventilation System	Resolved	(SER) (SSER 22)	9.4.4	
(238)	Engineered Safety Features Ventilation System	Resolved	(SER) (SSER 9) (SSER 10) (SSER 11) (SSER 14) (SSER 16) (SSER 19) (SSER 22)	9.4.5	
(239)	Reactor Building Purge Ventilation System		(SSER 22)	9.4.6	
(240)	Containment Air Cooling System		(SSER 22)	9.4.7	
(241)	Condensate Demineralizer Waste Evaporator Building Environmental Control System		(SSER 22)	9.4.8	
(242)	Other Auxiliary Systems			9.5	
(243)	Fire Protection	Resolved	(SER) (SSER 10) (SSER 18) (SSER 19)	9.5.1	
(244)	Communications System	Resolved	(SER) (SSER 5)	9.5.2	1
(245)	Lighting System	Resolved	(SER) (SSER 22)	9.5.3	

	Issue	Status		Section	Note
(246)	Emergency Diesel Engine Fuel Oil Storage and Transfer System	Resolved	(SER) (SSER 5) (SSER 9) (SSER 10) (SSER 11) (SSER 12) (SSER 22)	9.5.4	2
(247)	Emergency Diesel Engine Cooling Water System	Resolved	(SER) (SSER 5) (SSER 11)	9.5.5	1
(248)	Emergency Diesel Engine Starting Systems	Resolved	(SER) (SSER 5) (SSER 10) (SSER 22)	9.5.6	2
(249)	Emergency Diesel Engine Lubricating Oil System	Resolved	(SER) (SSER 3) (SSER 5) (SSER 10) (SSER 22)	9.5.7	2
(250)	Emergency Diesel Engine Combustion Air Intake and Exhaust System	Resolved	(SER) (SSER 5) (SSER 10) (SSER 22)	9.5.8	2
(251)	Steam and Power Conversion System			10	
(252)	Summary Description	Resolved	(SER)	10.1	
(253)	Turbine Generator	Open (NRR)	(SER) (SSER 5)	10.2	
(254)	Turbine Generator Design	Resolved	(SER) (SSER 12) (SSER 22)	10.2.1	
(255)	Turbine Disc Integrity	Resolved	(SER)	10.2.2	
(256)	Main Steam Supply System	Resolved	(SER)	10.3	
(257)	Main Steam Supply System (Up to and Including the Main Steam Isolation Valves)	Resolved	(SER) (SSER 19) (SSER 22)	10.3.1	
(258)	Main Steam Supply System	Resolved	(SER) (SSER 22)	10.3.2	2
(259)	Steam and Feedwater System Materials	Resolved	(SER) (SSER 22)	10.3.3	
(260)	Secondary Water Chemistry	Resolved	(SER) (SSER 5) (SSER 22)	10.3.4	
(261)	Other Features			10.4	
(262)	Main Condenser	Resolved	(SER) (SSER 9) (SSER 22)	10.4.1	

	Issue	Status		Section	Note
(263)	Main Condenser Evacuation System	Resolved	(SER) (SSER 22)	10.4.2	
(264)	Turbine Gland Sealing System	Resolved	(SER) (SSER 22)	10.4.3	
(265)	Turbine Bypass System	Resolved	(SER) (SSER 5) (SSER 22)	10.4.4	
(266)	Condenser Circulating Water System	Resolved	(SER) (SSER 22)	10.4.5	
(267)	Condensate Cleanup System	Open (NRR)	(SER) (SSER 22)	10.4.6	
(268)	Condensate and Feedwater Systems	Resolved	(SER) (SSER 14) (SSER 22)	10.4.7	
(269)	Steam Generator Blowdown System	Open (NRR)	(SER) (SSER 22)	10.4.8	
(270)	Auxiliary Feedwater System	Resolved	(SER) (SSER 14)	10.4.9	
(271)	Heater Drains and Vents		(SSER 22)	10.4.10	
(272)	Steam Generator Wet Layup System		(SSER 22)	10.4.11	
(273)	Radioactive Waste Management			11	
(274)	Summary Description	Resolved	(SER) (SSER 16)	11.1	2
(275)	Liquid Waste Management	Resolved	(SER) (SSER 4) (SSER 16)	11.2	
(276)	Gaseous Waste Management	Resolved	(SER) (SSER 8) (SSER 16)	11.3	
(277)	Solid Waste Management System	Open (NRR)	(SER) (SSER 16)	11.4	
(278)	Process and Effluent Radiological Monitoring and Sampling Systems	Open (NRR)	(SER) (SSER 16) (SSER 20)	11.5	
(279)	Evaluation Findings	Open (NRR)	(SER) (SSER 8) (SSER 16)	11.6	
(280)	NUREG-0737 Items	Open (NRR)	(SER)	11.7	
(281)	Wide-Range Noble Gas, Iodine, and Particulate Effluent Monitors (TMI Action Items II.F.1(1) and II.F.1(2))	Open (Inspection)	(SER) (SSER 5) (SSER 6)	11.7.1	
(282)	Primary Coolant Outside Containment (TMI Action item III.D.1.1)	Open (NRR)	(SER) (SSER 5) (SSER 6) (SSER 10) (SSER 16)	11.7.2	
(283)	Radiation Protection			12	

	Issue	Status		Section	Note
(284)	General	Open (NRR)	(SER) (SSER 10) (SSER 14)	12.1	
(285)	Ensuring that Occupational Radiation Doses Are As Low As Reasonably Achievable (ALARA)	Resolved[2]	(SER) (SSER 14)	12.2	
(286)	Radiation Sources	Open (NRR)	(SER) (SSER 14)	12.3	
(287)	Radiation Protection Design Features	Open (NRR)	(SER) (SSER 10) (SSER 14) (SSER 18)	12.4	
(288)	Dose Assessment	Open (NRR)	(SER) (SSER 14)	12.5	
(289)	Health Physics Program	Open (NRR)	(SER) (SSER 10) (SSER 14)	12.6	
(290)	NUREG-0737 Items			12.7	
(291)	Plant Shielding	Open (NRR)	(SER) (SSER 14) (SSER 16)	12.7.1	
(292)	High Range In-Containment Monitor (TMI Action Item II.F.1.(3))	Open (NRR)	(SER) (SSER 5)	12.7.2	
(293)	In-Plant Radioiodine Monitor (TMI Action Item II.D.3.3)	Open (NRR)	(SER) (SSER 16)	12.7.3	
(294)	Conduct of Operations			13	
(295)	Organization Structure of the Applicant	Resolved	(SER) (SSER 16) (SSER 22)	13.1	
(296)	Management and Technical Organization	Resolved	(SER)	13.1.1	
(297)	Corporate Organization and Technical Support	Resolved	(SER)	13.1.2	
(298)	Plant Staff Organization	Open (NRR)	(SER) (SSER 8) (SSER 22)	13.1.3	
(299)	Training			13.2	
(300)	Licensed Operator Training Program	Resolved	(SER) (SSER 9) (SSER 10) (SSER 22)	13.2.1	
(301)	Training for Non-licensed Personnel	Resolved	(SER)	13.2.2	
(302)	Emergency Preparedness Evaluation			13.3	
(303)	Introduction	Open (NRR)	(SER) (SSER 13) (SSER 20)	13.3.1	

	Issue	Status		Section	Note
(304)	Evaluation of the Emergency Plan	Open (NRR)	(SER) (SSER 13) (SSER 20) (SSER 22)	13.3.2	
(305)	Conclusions	Open (NRR)	(SER) (SSER 13) (SSER 20) (SSER 22)	13.3.3	
(306)	Review and Audit	Resolved	(SER) (SSER 8) (SSER 22)	13.4	
(307)	Plant Procedures	Resolved	(SER) (SSER 22)	13.5	
(308)	Administrative Procedures	Resolved	(SER) (SSER 22)	13.5.1	
(309)	Operating and Maintenance Procedures	Resolved	(SER) (SSER 9) (SSER 10) (SSER 22)	13.5.2	
(310)	NUREG-0737 Items	Resolved	(SER) (SSER 3) (SSER 16) (SSER 22)	13.5.3	
(311)	Physical Security Plan	Resolved	(SER) (SSER 1) (SSER 10) (SSER 15) (SSER 20) (SSER 22)	13.6	
(312)	Introduction		(SSER 22)	13.6.1	
(313)	Summary of Application		(SSER 22)	13.6.2	
(314)	Regulatory Basis		(SSER 22)	13.6.3	
(315)	Technical Evaluation		(SSER 22)	13.6.4	
(316)	Conclusions		(SSER 22)	13.6.5	
(317)	Initial Test Program	Open (Inspection)	(SER) (SSER 3) (SSER 5) (SSER 7) (SSER 9) (SSER 10) (SSER 12) (SSER 14) (SSER 16) (SSER 18) (SSER 19)	14	
(318)	Accident Analyses			15	
(319)	General Discussion	Resolved	(SER)	15.1	

	Issue	Status		Section	Note
(320)	Normal Operation and Anticipated Transients	Open (NRR)	(SER)	15.2	
(321)	Loss-of-Cooling Transients	Open (NRR)	(SER) (SSER 13) (SSER 14)	15.2.1	
(322)	Increased Cooling Inventory Transients	Resolved	(SER)	15.2.2	
(323)	Change in Inventory Transients	Open (NRR)	(SER) (SSER 18)	15.2.3	
(324)	Reactivity and Power Distribution Anomalies	Open (NRR)	(SER) (SSER 4) (SSER 7) (SSER 13) (SSER 14)	15.2.4	
(325)	Conclusions	Resolved	(SER) (SSER 4)	15.2.5	
(326)	Limiting Accidents	Resolved	(SER)	15.3	
(327)	Loss-of-Coolant Accident (LOCA)	Open (NRR)	(SER) (SSER 12) (SSER 15)	15.3.1	
(328)	Steamline Break	Open (NRR)	(SER) (SSER 3) (SSER 14)	15.3.2	
(329)	Feedwater System Pipe Break	Open (NRR)	(SER) (SSER 14)	15.3.3	
(330)	Reactor Coolant Pump Rotor Seizure	Open (NRR)	(SER) (SSER 14)	15.3.4	
(331)	Reactor Coolant Pump Shaft Break	Open (NRR)	(SER) (SSER 14)	15.3.5	
(332)	Anticipated Transients Without Scram	Open (Inspection)	(SER) (SSER 3) (SSER 5) (SSER 6) (SSER 10) (SSER 11) (SSER 12)	15.3.6	
(333)	Conclusions	Resolved	(SER)	15.3.7	
(334)	Radiological Consequences of Accidents	Resolved	(SER) (SSER 15)	15.4	
(335)	Loss-of-Coolant Accident	Open (NRR)	(SER) (SSER 5) (SSER 9) (SSER 18)	15.4.1	
(336)	Main Steamline Break Outside of Containment	Open (NRR)	(SER) (SSER 15)	15.4.2	

	Issue	Status		Section	Note
(337)	Steam Generator Tube Rupture	Open (NRR)	(SER) (SSER 2) (SSER 5) (SSER 12) (SSER 14) (SSER 15)	15.4.3	
(338)	Control Rod Ejection Accident	Open (NRR)	(SER) (SSER 15)	15.4.4	
(339)	Fuel-Handling Accident	Open (NRR)	(SER) (SSER 4) (SSER 15)	15.4.5	
(340)	Failure of Small Line Carrying Coolant Outside Containment	Open (NRR)	(SER)	15.4.6	
(341)	Postulated Radioactive Releases as a Result of Liquid Tank Failures	Open (NRR)	(SER)	15.4.7	
(342)	NUREG-0737 Items			15.5	
(343)	Thermal Mechanical Report (TMI Action Item II.K.2.13)	Resolved	(SER) (SSER 4)	15.5.1	
(344)	Voiding in the Reactor Coolant System during Transients (TMI Action Item II.K.2.17)	Resolved	(SER) (SSER 4)	15.5.2	
(345)	Installation and Testing of Automatic Power-Operated Relief Valve Isolation System (TMI Action Item II.K.3.1) Report on Overall Safety Effect of Power-Operated Relief Valve Isolation System (TMI Action Item II.K.3.2)	Resolved	(SER) (SSER 5)	15.5.3	
(346)	Automatic Trip of Reactor Coolant Pumps (TMI Action Item II.K.3.5)	Open (Inspection)	(SER) (SSER 4) (SSER 16)	15.5.4	
(347)	Small-Break LOCA Methods (II.K.3.30) and Plant-Specific Calculations (II.K.3.31)	Open (Inspection)	(SER) (SSER 4) (SSER 5) (SSER 16)	15.5.5	
(348)	Relative Risk of Low-Power Operation	Resolved	(SER)	15.6	
(349)	Technical Specification	Open (NRR)		16	
(350)	Quality Assurance			17	
(351)	General	Resolved	(SER)	17.1	
(352)	Organization	Resolved	(SER)	17.2	
(353)	Quality Assurance Program	Resolved	(SER) (SSER 2) (SSER 5) (SSER 10) (SSER 13) (SSER 15) (SSER 22)	17.3	

	Issue	Status		Section	Note
(354)	Conclusions	Resolved	(SER)	17.4	
(355)	Maintenance Rule			17.6	
(356)	Control Room Design Review			18	
(357)	General	Resolved	(SER) (SSER 5) (SSER 6) (SSER 15) (SSER 16) (SSER 22)	18.1	
(358)	Conclusions	Resolved	(SER) (SSER 16) (SSER 22)	18.2	
(359)	Report of the Advisory Committee on Reactor Safeguards		(SER)	19	
(360)	Common Defense and Security		(SER)	20	
(361)	Financial Qualifications		(SER)	21	
(362)	TVA Financial Qualifications for WBN Unit 2		(SSER 22)	21.1	
(363)	Foreign Ownership, Control, or Domination		(SSER 22)	21.2	
(364)	Financial Protection and Indemnity Requirements			22	
(365)	General		(SER)	22.1	
(366)	Preoperational Storage of Nuclear Fuel		(SER)	22.2	
(367)	Operating Licenses	Open (NRR)	(SSER 22)	22.3	
(368)	Quality of Construction, Operational Readiness, and Quality Assurance Effectiveness			25	
(369)	Program for Maintenance and Preservation of the Licensing Basis for Units 1 and 2	Open (NRR)	(SSER 22)	25.9	

Notes:

1. In the process of further validating the information in the WBN Unit 2 FSAR, TVA identified minor administrative/typographical changes to sections previously considered Resolved. TVA addressed these changes to the applicable sections in their submittals and clearly indicated them to the staff. The staff has reviewed and confirmed that the changes made are administrative/typographical and do not impact the staff's conclusions as stated in previous SSERs. Based on this review, no additional review is necessary and this section remains Resolved.

2. During the assessment of the regulatory framework for completion of the project, the staff characterized certain topics as "Open" pending TVA's validation of the information contained in the section. TVA has determined that the information presented in the FSAR remained valid and only identified minor administrative or typographical changes to the section. TVA addressed the changes in their submittals and clearly indicated the changes. The staff reviewed and confirmed

that the changes made to the section are administrative/typographical and do not impact its conclusions as stated in previous SSERs. Therefore, no additional review is necessary and the staff considers this section Resolved.

3. In SSER 21, this issue was identified as "Resolved." However, TVA made changes to the Unit 2 FSAR affecting the previous staff conclusions. The staff evaluated the changes and the results are documented in this SSER.

1.8 Confirmatory Issues

At this point in the review, there are some items that have essentially been resolved to the staff's satisfaction, but for which certain confirmatory information has not yet been provided by the applicant. In these instances, the applicant has committed to provide the confirmatory information in the near future. If staff review of this information does not confirm preliminary conclusions on an item, that item will be treated as open, and the NRC staff will report on its resolution in a supplement to this report.

The confirmatory items, with appropriate references to subsections of this report, are noted in Appendix HH.

1.13 Implementation of Corrective Action Programs and Special Programs

In 1985, TVA developed a corporate Nuclear Performance Plan (NPP) 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). TVA established 18 corrective action programs (CAPs) and 11 special programs (SPs) to address these concerns.

SSER 21, Table 1.13.1 documented the status of staff review of the CAPs and SPs. This SSER and future supplements to the SER, the staff will document its evaluation and closure of open NPP items.

1.13.1 Corrective Action Programs

No.	Title	Program Review Status
(1)	Cable Issues a. Silicon Rubber Insulated Cable b. Cable Jamming c. Cable Support in Vertical Conduit d. Cable Support in Vertical Trays e. Cable Proximity to Hot Pipes f. Cable Pull-Bys g. Cable Bend Radius h. Cable Splices i. Cable Sidewall Bearing Pressure j. Pulling Cables Through 90° Condulet and Flexible Conduit k. Computer Cable Routing System Software and Database Verification and Validation	Resolved (See Appendix HH)

No.	Title	Program Review Status
(2)	Cable Tray and Tray Supports	Resolved
(3)	Design Baseline and Verification Program	Resolved
(4)	Electrical Conduit and Conduit Support	Resolved
(5)	Electrical Issues a. Flexible Conduit Installations b. Physical Cable Separation and Electrical Isolation c. Contact and Coil Rating of Electrical Devices d. Torque Switch and Overload Relay Bypass Capability for Active Safety-Related Valves e. Adhesive-Backed Cable Support Mount	 Resolved (See Appendix HH)
(6)	Equipment Seismic Qualification	Resolved
(7)	Fire protection	Resolved
(8)	Hanger and Analysis Update Program	Resolved
(9)	Heat Code Traceability	Resolved
(10)	Heating, Ventilation, and Air-Conditioning Duct and Duct Supports	Resolved
(11)	Instrument Lines	Resolved
(12)	Prestart Test Program Plan	Resolved
(13)	Quality Assurance (QA) Records	Resolved
(14)	Quality-List (Q-List)	Resolved
(15)	Replacement Items Program (Piece Parts)	Resolved
(16)	Seismic Analysis	Resolved
(17)	Vendor Information Program	Resolved
(18)	Welding	Resolved

1.13.2 Special Programs

No.	Title	Program Review Status
(1)	Concrete Quality Program	Resolved

No.	Title	Program Review Status
(2)	Containment Cooling	Resolved
(3)	Detailed Control Room Design Review	Resolved
(4)	Environmental Qualifications Program	Resolved
(5)	Master Fuse List	Resolved
(6)	Mechanical Equipment Qualification	Resolved
(7)	Microbiologically Induced Corrosion	Resolved
(8)	Moderate Energy Line Break Flooding	Resolved
(9)	Radiation Monitoring System	Resolved
(11)	Use-As-Is Condition Adverse to Quality	Resolved

1.14 Implementation of Applicable Bulletin and Generic Letter Requirements

From time to time, the NRC staff issues generic requirements or recommendations in the form of orders, bulletins (BLs), generic letters (GLs), regulatory issue summaries, and other documents to address certain safety and regulatory issues. These are generally termed "generic communications."

The table below outlines the status of the resolution of the generic communications identified in SSER 21. It should be noted that, although many of the generic communications have been documented or otherwise resolved, the NRC staff has determined that there may be circumstances that could result in the need to reopen a previously closed topic.

	Correspondence No.	Title
(1)	GL 1980-14	Light-Water Reactor Primary Coolant System Pressure Isolation Valves
	TVA Action:	Submit Technical Specifications (TSs) for NRC Review.
	NRC Action:	To be reviewed during validation of TS 3.4.14 submitted February 2, 2010.

	Correspondence No.	Title
(2)	GL 1980-77	Refueling Water Level - Technical Specifications Changes
	TVA Action:	Submit Technical Specifications for NRC Review.
	NRC Action:	To be reviewed during validation of TS 3.9.5 –TS 3.9.7 submitted February 2, 2010.
(3)	GL 1982-28	Inadequate Core Cooling Instrumentation System
	TVA Action:	Closed
	NRC Action:	Closed. Subsumed as part of NRC staff review of Instrumentation and Controls submitted April 8, 2010.
(4)	GL 1983-28	Required Actions Based on Generic Implications of Salem Anticipated Transient without Scram Events (Screened into the Items 4 through 7)
(4.a)	GL 1983-28 (item 3.1)	Post-Maintenance Testing (reactor trip system components)
	TVA Action:	Submit Technical Specifications for NRC Review
	NRC Action:	To be reviewed during validation of TS Bases 3.0.1 submitted March 4, 2009.
(4.b)	GL 1983-28 (3.2)	Post-Maintenance Testing (All Surveillance Requirement Components)
	TVA Action	Submit Technical Specifications and NRC Review
	NRC Action	To be reviewed during validation of TS Bases 3.0.1 submitted March 4, 2009.
(4.c)	GL 1983-28 (4.2)	Reactor Trip System Reliability (Preventive Maintenance and Surveillance Program for Reactor Trip Breakers)
	TVA Action	Submit Technical Specifications and NRC Review
	NRC Action	To be reviewed during staff evaluation of Item 17 of TS Table 3.3.1-1 submitted February 2, 2010.

	Correspondence No.	Title
(4.d)	GL 1983-28 (4.5)	Reactor Trip System Reliability (Automatic Actuation of Shunt Trip Attachment)
	TVA Action	Submit Technical Specifications and NRC Review
	NRC Action	To be reviewed during staff evaluation of Item 18 of TS Table 3.3.1-1 submitted February 2, 2010.
(8)	GL 1986-09	Technical Resolution of Generic Issue B-59, (N-1) Loop Operation in BWRs and PWRs
	TVA Action	Submit Technical Specifications for NRC Review.
	NRC Action	To be reviewed during validation of TS 3.4.4 - TS 3.4.8 submitted February 2, 2010.
(9)	GL 1988-20	Individual Plant Examination for Severe Accident Vulnerability
	TVA Action	Closed
	NRC Action	Open pending completion of staff review of IPE submitted February 9, 2010.
(10)	GL 1988-20s1	Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities — 10 CFR 50.54
	TVA Action	Closed
	NRC Action	Open pending completion of staff review of IPE submitted February 9, 2010.
(11)	GL 1988-20s2	Individual Plant Examination for Severe Accident Vulnerability. Accident Management Strategies for Consideration in the Individual Plant Examination Process
	TVA Action	Closed
	NRC Action	Open pending completion of staff review of IPE submitted February 9, 2010.

	Correspondence No.	Title
(12)	GL 1988-20s3	Individual Plant Examination for Severe Accident Vulnerability. Completion of Containment Performance Improvement Program and Forwarding of Insights for Use in the IPE for Severe Accident Vulnerabilities
	TVA Action	Closed
	NRC Action	Open pending completion of staff review of IPE submitted February 9, 2010.
(13)	GL 1988-20s4	Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities
	TVA Action	Closed
	NRC Action	Open pending completion of staff review of IPEEE submitted April 30, 2010.
(14)	GL 1988-20s5	Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)
	TVA Action	Closed
	NRC Action	Open pending completion of staff review of IPEEE submitted April 30, 2010.
(15)	GL 1989-04	Guidelines on Developing Acceptable Inservice Testing Programs
	TVA Action	The proposed approach has been approved for WBN Unit 1; the same approach will be proposed for use on WBN Unit 2 without change.
	NRC Action	Open
(16)	GL 1989-21	Request for Information Concerning Status of Implementation of Unresolved Safety Issue Requirements
	TVA Action	TVA provided an updated status of unresolved safety issues on September 26, 2008.
	NRC Action	Open

	Correspondence No.	Title
(17)	GL 1990-06	Resolution of Generic Issues 70, "PORV [power-operated relief valve] and Block Valve Reliability," and 94, "Additional LTOP [low-temperature overpressure] Protection for PWRs"
	TVA Action	Submit Technical Specifications for NRC Review
	NRC Action	To be reviewed during validation of TS 3.4.11 - TS 3.4.12 submitted February 2, 2010.
(18)	GL 1992-08	Thermo-Lag 330-1 Fire Barriers
	TVA Action	The proposed approach has been approved for WBN Unit 1; the same approach will be proposed for use on WBN Unit 2 without change
	NRC Action	Open. Pending NRC staff inspection verification.
(19)	GL 1995-03	Circumferential cracking of Steam Generator (SG) Tubes
	TVA Action	The proposed approach has been approved for WBN Unit 1; the same approach was submitted for use on WBN Unit 2 without change.
	NRC Action	Closed. NRC Letter dated January 21, 2010 (ADAMS Accession No. ML093631061)
(20)	GL 1995-05	Voltage –Based Repair Criteria for Westinghouse Steam Generator Tubes affected by Outside Diameter Stress Corrosion Cracking
	TVA Action	The proposed approach has been approved for WBN Unit 1; the same approach was submitted for use on WBN Unit 2 without change
	NRC Action	Closed. NRC Letter dated January 21, 2010 (ADAMS Accession No. ML093631061)
(21)	GL 1996-06	Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions
	TVA Action	The proposed approach has been approved for WBN Unit 1; the same approach will be proposed for use on WBN Unit 2 without change
	NRC Action	Closed. NRC Letter dated January 21, 2010 (ADAMS Accession No. ML100130227)

	Correspondence No.	Title
(22)	GL 1995-07	Pressure Locking and Thermal Binding of Safety- Related Power-Operated Gate Valves (Not identified in SSER 21 as "Open")
	TVA Action	The proposed approach has been approved for WBN Unit 1; the same approach will be proposed for use on WBN Unit 2 without change
	NRC Action	Closed. NRC letter dated August 12, 2010 (ADAMS Accession No. ML100190443)
(23)	GL 1997-01	Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations
	TVA Action	The proposed approach has been approved for WBN Unit 1; the same approach will be proposed for use on WBN Unit 2 without change
	NRC Action	Closed. NRC Letter dated June 30, 2010 (ADAMS Accession No. ML100539515)
(24)	GL 1997-04	Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps Integrity During Design-Basis Accident Conditions
	TVA Action	The proposed approach has been approved for WBN Unit 1; the same approach was submitted for use on WBN Unit 2 without change.
	NRC Action	Closed. NRC Letter dated February 18, 2010 (ADAMS Accession No. ML100200375)
(25)	GL 1997-05	SG Tube Inspection Techniques
	TVA Action	The proposed approach has been approved for WBN Unit 1; the same approach was submitted for use on WBN Unit 2 without change.
	NRC Action	Closed. NRC Letter dated January 21, 2010 (ADAMS Accession No. ML093631061)

	Correspondence No.	Title
(26)	GL 1997-06	Degradation of SG Internals
	TVA Action	The proposed approach has been approved for WBN Unit 1; the same approach was submitted for use on WBN Unit 2 without change.
	NRC Action	Closed. NRC Letter dated January 21, 2010 (ADAMS Accession No. ML093631061)
(27)	GL 1998-02	Loss of Reactor Coolant Inventory and Associated Potential for Loss of Emergency Mitigation Functions While in a Shutdown Condition
	TVA Action	The proposed approach has been approved for WBN Unit 1; the same approach will be proposed for use on WBN Unit 2 without change.
	NRC Action	Closed. NRC Letter dated May 11, 2010 (ADAMS Accession No. ML101200155)
(28)	GL 1998-04	Potential for Degradation of the ECCS and the Containment Spray System after a LOCA because of Construction and Protective Coating Deficiencies and Foreign Material in Containment
	TVA Action	The proposed approach has been approved for WBN Unit 1; the same approach was submitted for use on WBN Unit 2 without change.
	NRC Action	Closed. NRC Letter dated February 1, 2010 (ADAMS Accession No. ML100260594)
(29)	GL 2003-01	Control Room Habitability
	TVA Action	No action or documentation is provided to show the staff has reviewed the item for WBN Unit 2, and the resolution is through submittal of a technical specification.
	NRC Action	Closed. NRC Letter dated February 1, 2010 (ADAMS Accession No. ML100270076)

	Correspondence No.	Title
(30)	GL 2004-01	Requirements for SG Tube Inspection
	TVA Action	The proposed approach has been approved for WBN Unit 1; the same approach was submitted for use on WBN Unit 2 without change.
	NRC Action	Closed. NRC Letter dated January 21, 2010 (ADAMS Accession No. ML093631061)
(31)	GL 2004-02	Potential Impact of Debris Blockage on Emergency Recirculation during Design-Basis Accidents at PWRs
	TVA Action	The proposed approach has been approved for WBN Unit 1; the same approach was submitted for use on WBN Unit 2 without change.
	NRC Action	Open
(32)	GL 2006-01	SG Tube Integrity and Associated Technical Specifications
	TVA Action	No action or documentation is provided to show the staff has reviewed the item for WBN Unit 2, and the resolution is through submittal of a technical specification.
	NRC Action	Closed. NRC Letter dated January 21, 2010 (ADAMS Accession No. ML093631061) (See Appendix HH)
(33)	GL 2006-02	Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power
	TVA Action	The proposed approach has been approved for WBN Unit 1; the same approach was submitted for use on WBN Unit 2 without change.
	NRC Action	Closed. NRC Letter dated January 21, 2010 (ADAMS Accession No. ML093631061) (See Appendix HH)
(34)	GL 2006-03	Potentially Nonconforming Hemyc and MT Fire Barrier Configurations
	TVA Action	The proposed approach has been approved for WBN Unit 1; the same approach was submitted for use on WBN Unit 2 without change.
	NRC Action	Closed. NRC Letter February 25, 2010 (ADAMS Accession No. ML100470398)

	Correspondence No.	Title
(35)	GL 2007-01	Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients
	TVA Action	The proposed approach has been approved for WBN Unit 1; the same approach was submitted for use on WBN Unit 2 without change.
	NRC Action	Closed. NRC Letter dated January 26, 2010 (ADAMS Accession No. ML100120052)
(36)	GL 2008-01	Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems
	TVA Action	Open
	NRC Action	Open
(37)	BL 1992-01 and Supplement 1	Failure of Thermo-Lag 330 Fire Barrier System to Perform its Specified Fire Endurance Function
	TVA Action	The proposed approach has been approved for WBN Unit 1; the same approach will be proposed for use on WBN Unit 2 without change.
	NRC Action	Open. Pending NRC staff inspection verification.
(38)	BL 1996-01	Control Rod Insertion Problems (PWR)
	TVA Action	The proposed approach has been approved for WBN Unit 1; the same approach was submitted for use on WBN Unit 2 without change.
	NRC Action	Closed. NRC letter dated May 3, 2010 (ADAMS Accession No. ML101200035) required Confirmatory Action (See Appendix HH)
(39)	BL 1996-02	Movement of Heavy Loads Over Spent Fuel, Over Fuel In the Reactor Core, or Over Safety-Related Equipment
		The proposed approach has been approved for WBN Unit 1; the same approach was submitted for use on WBN Unit 2 without change
		Closed. NRC Letter dated March 4, 2010 (ADAMS Accession No. ML100480062)

	Correspondence No.	Title
(40)	BL 2001-01	Circumferential Cracking of Reactor Pressure Vessel (RPV) Head Penetration Nozzles
	TVA Action	The proposed approach has been approved for WBN Unit 1; the same approach was submitted for use on WBN Unit 2 without change.
	NRC Action	Closed. See NRC Letter dated June 30, 2010 (ADAMS Accession No. ML 100539515)
(41)	BL 2002-01	RPV Head Degradation and Reactor Coolant Pressure Boundary Integrity
	TVA Action	The proposed approach has been approved for WBN Unit 1; the same approach was submitted for use on WBN Unit 2 without change.
	NRC Action	Closed. See NRC Letter dated June 30, 2010 (ADAMS Accession No. ML 100539515)
(42)	BL 2002-02	RPV Head and Vessel Head Penetration Nozzle Inspection Program
	TVA Action	The proposed approach has been approved for WBN Unit 1; the same approach was submitted for use on WBN Unit 2 without change.
	NRC Action	Closed. See NRC Letter dated June 30, 2010 (ADAMS Accession No. ML100539515)
(43)	BL 2003-02	Leakage from RPV Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity
	TVA Action	The proposed approach has been approved for WBN Unit 1; the same approach was submitted for use on WBN Unit 2 without change.
	NRC Action	Closed. NRC Letter dated January 21, 2010 (ADAMS Accession No. ML093631061)

	Correspondence No.	Title
(44)	BL 2004-01	Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at PWRs
	TVA Action	The proposed approach has been approved for WBN Unit 1; the same approach was submitted for use on WBN Unit 2 without change.
	NRC Action	Closed. NRC letter dated August 4, 2010 (ADAMS Accession No. ML102080017)
(45)	BL 2007-01	Security Officer Attentiveness
	TVA Action	The proposed approach has been approved for WBN Unit 1; the same approach will be proposed for use on WBN Unit 2 without change.
	NRC Action	Closed. NRC letter dated March 25, 2010 (ADAMS Accession No. ML100770549)

NUREG-0737, TMI Action Items (TVA letter dated September 14, 1981, applies to all of the following NUREG-0737 issues)

(46)	NUREG-0737 Item I.B.1.2	Independent Safety Engineering Group
	TVA Action	The proposed approach has been approved for WBN Unit 1; the same approach will be proposed for use on WBN Unit 2 without change.
	NRC Action	Open
(47)	NUREG-0737 Item I.D.1	Control Room Design Review (CRDR)
	TVA Action	The proposed approach has been approved for WBN Unit 1; the same approach will be proposed for use on WBN Unit 2 without change.
	NRC Action	Closed in SSER 22, Section 18.2

	Correspondence No.	Title
(48)	NUREG-0737 Item II.B.3	Post-accident Sampling
	TVA Action	No action or documentation is provided to show the staff has reviewed the item for WBN Unit 2, and the resolution is through submittal of a technical specification.
	NRC Action	Open
(49)	NUREG-0737 Item II.E.4.2	Containment Isolation Dependability
	TVA Action	No action or documentation is provided to show the staff has reviewed the item for WBN Unit 2, and the resolution is through submittal of a technical specification.
	NRC Action	Open
(50)	NUREG-0737 Item II.F.2	Instrumentation for Detection of Inadequate Core-Cooling
	TVA Action	Open
	NRC Action	Open
(51)	NUREG-0737 Item II.K.3.3	Reporting SV/RV Failures/Challenges
	TVA Action	No action or documentation is provided to show the staff has reviewed the item for WBN Unit 2, and the resolution is through submittal of a technical specification.
	NRC Action	Closed in SSER 22, Section 13.5.3
(52)	NUREG-0737 Item II.K.3.10	Anticipatory Trip at High Power
	TVA Action	No action or documentation is provided to show the staff has reviewed the item for WBN Unit 2, and the resolution is through submittal of a technical specification.
	NRC Action	Open

	Correspondence No.	Title
(53)	NUREG-0737 Item III.D.1.1	Primary Coolant Outside Containment
	TVA Action	No action or documentation is provided to show the staff has reviewed the item for WBN Unit 2, and the resolution is through submittal of a technical specification.
	NRC Action	Open
(54)	NUREG-0737 Item III.D.3.4	Control-Room Habitability
	TVA Action	The proposed approach has been approved for WBN Unit 1; the same approach will be proposed for use on WBN Unit 2 without change.
	NRC Action	Closed in SSER 22, Section 6.4
(55)	IEB 75-08	PWR Pressure Instrumentation
	TVA Action	The item has been approved either for both units at WBN or explicitly for WBN Unit 2; however, a change to the original approval requires submittal of the technical specifications and staff review.
	NRC Action	Open
(56)	IEB 77-04	Calculation Error Affecting Performance of a System for Controlling pH of Containment Sump Water Following a LOCA
	TVA Action	The item has been approved either for both units at WBN or explicitly for WBN Unit 2; however, a change to the original approval requires submittal of the technical specifications and staff review.
	NRC Action	Open

1.17 <u>Financial Assurance for Decommissioning</u>

In its application, TVA acknowledged the requirements under 10 CFR 50.75 regarding the certification requirements and stated that it will provide decommissioning funding assurance in an amount of $400.3 million (2008 dollars) for decommissioning of Units 1 and 2. TVA stated that it will use an external sinking fund as the method to provide decommissioning funding assurance.

The NRC staff independently calculated the minimum funding amount required under 10 CFR 50.75(c) and found that the applicant's amount to be acceptable. The NRC staff also reviewed TVA's plan to use an external sinking fund to provide decommissioning funding assurance and finds that TVA has complied with the requirements of 10 CFR 50.75.

2 SITE ENVELOPE

2.1 Geography and Demography

The staff reviewed the Watts Bar Nuclear (WBN) Unit 2 site envelope in accordance with NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (hereafter referred to as the SRP), Sections 2.1.1, 2.1.2, and 2.1.3. The sections below discuss the results of this review.

2.1.1 Site Location and Description

The WBN facility is located on a tract of approximately 1,770 acres in Rhea County on the west bank of the Tennessee River at River Mile 528. The site is approximately 1.25 miles south of the Watts Bar Dam and approximately 31 miles north-northeast of the Sequoyah Nuclear Plant.

The United States Government owns the 1,770-acre reservation, and the land is in the custody of the Tennessee Valley Authority (TVA). Also located within the reservation are the Watts Bar Dam and hydroelectric plant, the Watts Bar (fossil-fired) Steam Plant, the TVA Central Maintenance Facility, and the Watts Bar resort area.

The resort area buildings and improvements have been sold to private individuals and the associated land mass leased to the Watts Bar Village Corporation, Inc. Because of this sale and leasing arrangement, WBN provides no services to the resort area.

The location of each reactor is given below:

Longitude and Latitude (degrees/minutes/seconds)	
UNIT 1 35°36 10.430" N	84°47 24.267" W
UNIT 2 35°36 10.813" N	84°47 21.398" W

Universal Transverse Mercator (Meters)	
Northing	Easting
UNIT 1 N3, 941,954.27	E 700,189.94
UNIT 2 N3, 941,967.71	E 700,261.86

2.1.2 Exclusion Area Authority and Control

Because the WBN site is so large, the site boundary completely contains the exclusion area, which is determined by a circle with a radius of 1,200 meters (3,940 feet) centered on a point 20 feet from the north wall of the turbine building along the building centerline. The exclusion area boundary will be clearly marked on all access roads. Figure 2.1-4b of the final safety analysis report (FSAR) shows the exclusion area.

The United States Government owns all of the land inside the exclusion area, which is in the custody of TVA. TVA controls all activities within the reservation. TVA states that there will be no residences, unauthorized commercial operations, or recreational areas within the exclusion area. No public highways or railroads cross the exclusion area; however, a portion of the Tennessee River crosses the eastern portion of the exclusion area. This portion of the river is accessible for fishing, pleasure boating, and commercial transportation.

2.1.3 Population Distribution

In FSAR Amendment 94, TVA provided updated population data and a projection based on the following sources: (1) U.S. Census Bureau, Census of Population, 2000, including block group, block, and census track data, (2) county projections by Woods & Poole, (3) subcounty population estimates using a constant share of the 1990 county total from the Economic Analysis Division, Bureau of Economic Analysis, U.S. Department of Commerce, 1992, and (4) county census maps and 1:250,000 topographic maps used to desegregate subcounty population data into the annular segments. Considerations included municipal limits, topography, road system, land ownership (e.g., National Forest), and land use (e.g., strip mines).

Table 2.1-1 Resident Population in Watts Bar Vicinity

Year	0–1 mi	1–2 mi	2–3 mi	3–4 mi	4–5 mi	5–10 mi	0–10 mi
1970	35	155	350	610	600	8,985	10,735
1980	45	165	390	695	715	10,325	12,335
2000	22	102	532	1,375	2,595	14,302	18,928
2010	27	113	619	1,599	3,085	16,556	21,999
2020	35	126	735	1,895	3,790	19,531	26,112
2030	42	141	854	2,205	4,520	22,648	30,410
2040	48	152	945	2,427	5,050	24,883	33,505
2050	55	163	1,045	2,689	5,673	27,534	37,159
2060	61	175	1,151	2,955	6,294	30,183	40,819

About 18,900 people lived within 10 miles of the WBN site in 2000, with more than 75 percent of them between 5 and 10 miles from the site. Two small towns, Spring City and Decatur, which in 2007 had populations of 2,002 and 1,456, respectively, are located between 5 and 10 miles from the site. Decatur is south, while Spring City is northwest of the site. Most of the remainder of the area is sparsely populated, especially within 5 miles of the site. TVA expects this pattern to continue.

The area between 10 and 50 miles from the site lies mostly in the lower and middle portions of east Tennessee, with small areas in southwestern North Carolina and in northern Georgia. The population of this area is projected to increase by about 62 percent, or 660,000 persons, between 2000 and 2060. About 71 percent of this total increase is expected to be in the area between 30 and 50 miles from the site.

The largest urban concentration between 10 and 50 miles is the city of Chattanooga, located to the southwest and south-southwest. This city had a population in 2007 of 169,884; about 80 percent located between 40 and 50 miles from the site, while the rest is located beyond 50 miles. The city of Knoxville is located to the east-northeast of the site and is slightly larger than Chattanooga. However, less than 10 percent of its population of 183,546 is located between 40 and 50 miles of the site, with the remainder located beyond 50 miles.

Three smaller urban concentrations in this area have populations greater than 20,000. The city of Oak Ridge, which had a population in 2007 of 27,514, is located about 40 miles to the northeast. The twin cities of Alcoa and Maryville, which had a combined population in 2007 of about 35,300, are located between 45 and 50 miles to the east-northeast. Cleveland, with a population in 2007 of 39,200, is located about 30 miles to the south. TVA expects most of the population growth to occur around these and the larger population centers.

TVA has chosen the low-population zone (LPZ) distance, as defined in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 100, "Reactor Site Criteria," to be 4,828 meters (3 miles). The population of the LPZ (2,976 in 2010) and the population density (105 people per square mile in 2010) are both low. The population includes estimates for both permanent residents (759) and transients (2,217) for 2010. Transients are defined as "peak hour recreation visitors." In addition, this area is of such size that, in the unlikely event of a serious accident, there is a reasonable probability that appropriate measures could be taken to protect the health and safety of the residents. The WBN site emergency plan includes specific provisions to protect this area.

The nearest population center distance (as defined by 10 CFR Part 100) is approximately 30 miles south of the Watts Bar site in Cleveland, Tennessee. In 2007, Cleveland had a population of 39,200.

2.1.4 Conclusions

As discussed above, TVA has presented and substantiated information to establish (1) the site location and description, (2) its control of all activities within the designated exclusion area, and (3) a description of current and projected population densities in and around the site. The NRC staff has reviewed the information provided by TVA and concludes that the information is acceptable regarding geography and demography on the basis that the applicant's definitions of the exclusion area, low population zone, and population center distance conform to 10 CFR Part 100. In addition, the NRC staff evaluated (1) the onsite meteorological data from which the relative concentration factors (X/Q) were calculated by TVA (see Section 2.3 of this report), and (2) the calculated potential radiological dose consequences of the design basis accidents (see Section 15 of this report). The NRC staff concludes that that the exclusion area, low population zone, and population center distance meet the criteria of 10 CFR 100 and are acceptable.

2.2 Nearby Industrial, Transportation, and Military Facilities

The staff reviewed WBN site characteristics in accordance with SRP Sections 2.2.1, 2.2.2, and 2.2.3. The following sections discuss the results of this review.

2.2.1 Transportation Routes

The only significant nearby industrial facility is the Watts Bar Steam Plant. The nearest land transportation route is State Route 68, about 1 mile north of the site. The Tennessee River is navigable past the site. A main line of the Cincinnati, New Orleans and Texas Pacific Railway (Norfolk Southern Corporation) is located approximately 7 miles west of the site. A TVA railroad spur track connects to this main line and serves the Watts Bar Steam Plant and WBN. The spur has fallen into disuse and would need to be repaired before reuse. No other significant industrial land use, military facilities, or transportation routes are in the vicinity of the nuclear plant.

2.2.2 Nearby Facilities

The Watts Bar Steam Plant is a coal-fired electric generating facility with a total capacity of 240,000 kilowatts (kW) and, during normal operation, has about 100 employees. The plant is not currently operating but could be reactivated in the future.

The Tennessee River is a major barge route in which a 9-foot navigation channel is maintained.

Table 2.2-1 shows the total amount of certain hazardous materials shipped past WBN from 2002 to 2007 on a yearly basis.

Table 2.2-1 Waterborne Hazardous Material Traffic (Tons)
(U.S. Army Corps of Engineers)
2002–2007

Commodities	2002	2003	2004	2005	2006	2007
Ammonium Nitrate Fertilizers			3,110			
Carbon (Including Carbon Black), NEC	15,232	7,605	1,348	1,518		
Ethyl Alcohol (Not Denatured) 80% or More Alcohol	137,147	118,594	137,464	133,412	76,993	8,947
Fuel Oils, NEC			3,400			7,209
Lubrication Petroleum Oils from Petrol & Bitum Min				12,732		
Other Light Oils from Petroleum & Bitum Minerals						9,120
Petro. Bitumen, Petro. Coke, Asphalt, Bitumen Mixes NEC	1,531	12,708	25,183	11,437	3,148	71,061
Petroleum Oils/Oils from Bituminous Minerals, Crude				6,674		

Commodities	2002	2003	2004	2005	2006	2007
Pitch & Pitch Coke from Coal Tar/Oth Mineral Tars	248,986	258,584	236,716	254,001	235,381	164,752
Vermiculite, Perlite, Chlorites			1,642		1,643	
Grand Total	**402,896**	**397,491**	**408,863**	**419,774**	**317,165**	**261,089**

Total traffic past the site was 670,716 tons in 2008, compared to 1,294,959 tons in 1990 and 760,000 tons in 1975.

When in operation, traffic on the TVA railroad spur consisted of heavy components for the nuclear plant. If the Watts Bar Steam Plant were reactivated, the spur would also be used to deliver heavy components and coal to it.

No pipelines carrying petroleum products are located in the vicinity of the nuclear plant.

The WBN site is located on a 9-foot navigable channel on Chickamauga Reservoir. Its intake structure is located approximately 2 miles downstream of Watts Bar Lock and Dam. Watts Bar lock is 60 feet wide by 360 feet long and is located on the east bank of the Tennessee River. Towboat sizes vary from 1,500 to 1,800 horsepower for this section of the Tennessee River (Chattanooga to Knoxville). The most common type of barge using the waterway is the 35 foot by 195 foot jumbo barge with a 1,500 ton capacity. Other ships using the waterway include numerous liquid cargo (tank) barges of varying sizes with capacities up to 3,000 tons.

No airports are located within 10 miles of the site. Mark Anton Airport is about 11 miles southwest of the site. Its longest runway is 4,500 feet and is hard surfaced. The airport has no commercial facilities. Lovell Field, about 45 miles south-southwest of the site, is the nearest airfield with commercial facilities. The annual number of movements per year is about 62,000 for Lovell Field and about 4,000 at Mark Anton Airport, of which about 1,300 are student pilots executing "touch and go's." Traffic on VOR Federal Airway V51 totals fewer than 2,000 flights per year, based on 2008 data.

Two major potential industrial sites are located within 5 miles of WBN. About 4 miles southwest of the plant is a 3,000-acre tract, and about 3 miles north is a 200-acre tract. The 3,000-acre site is currently owned by the Mead Corporation. A site impact analysis was performed for the possible construction of a paper plant. However, the Mead Corporation has withdrawn its application to build the plant, and there are no immediate or future plans for development. The 200-acre tract is undeveloped, and there are no immediate or future plans for development of the site.

A study by TVA of the products and materials transported past the site by barge concluded that no potential explosive hazard exists. Other than the intake pumping station, which is discussed below, the worst potential hazard to onsite essential safety features from an accident involving the products transported near the site (coal, fuel oil, asphalt, tar and pitches), would be the generation of smoke by the burning of these products. The hazard to the Main Control Room from the generation of smoke from these products is discussed in Section 6.4.4.2.

Gasoline is supplied to Knoxville via pipeline, but TVA stated that this pipeline is not in the vicinity of the WBN. As of 1974, when the pipeline began full operation, no future gasoline barge shipments past the WBN site are expected. Therefore, the potential for damage to the WBN from a gasoline barge explosion is negligible.

Fuel oil is shipped by barge past the WBN Site. In case of a fuel oil barge accident, fire and dense smoke may result. However, neither fire nor dense smoke will affect plant safety.

The intake pumping station is protected against fire by virtue of its design and location. Pump suction is taken from the bottom of the channel. All pumps, essential cables, and instruments are protected from fire by being enclosed within concrete walls. Also, the pumping station embayment is just downstream of the Watts Bar Dam, which is locked on the opposite side of the Tennessee River. Consequently, any oil released to the river would be swept by the current past the embayment that leads to the intake pumping station, due to the fact that the embayment is located on the inside of a bend in the Tennessee River.

Even if fuel oil from a spill should enter the embayment and reach the intake pumping station, the oil would have no significant effect on the water intake system or the systems it serves. Entry of oil in the intake is unlikely since the oil will float on water. A concrete skimmer wall exists at the pumping station, and the pumps take suction approximately 20 feet below the minimum normal water level. The pump suction would be approximately 10 feet below the water surface even in the event of failure of the downstream dam. Any oil that did enter the pumps would be highly diluted and in such a state would have a minor effect on system piping losses and heat exchanger capabilities.

2.2.3 Conclusions

Based on its review of the information provided by TVA, the NRC staff concludes that none of the activities being performed in the vicinity of the site are considered to be a potential hazard to the plant.

As discussed in Sections 2.2.1 and 2.2.2 above, TVA has identified potential hazards in the site vicinity. The NRC staff has reviewed the information provided by TVA based upon the criteria given in GDC 4 and in SRP section 2.2.3. The NRC staff concludes that WBN Unit 2 is adequately protected and can be operated with an acceptable degree of safety as a result of activities at nearby transportation, industrial, and military facilities.

2.3 Meteorology

Paragraph (c)(2) of 10 CFR 100.20, "Factors To Be Considered when Evaluating Sites," states that "meteorological characteristics of the site that are necessary for safety analysis or that may have an impact upon plant design (such as maximum probable wind speed and precipitation) must be identified and characterized." Furthermore, General Design Criterion 2, "Design Bases for Protection against Natural Phenomena," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," states the following, with regard to meteorology:

> Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as...tornadoes,

hurricanes...without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and (3) the importance of the safety functions to be performed.

Thus, climate and meteorological data are needed as inputs to engineering design and safety assessments.

In the NRC staff's original safety evaluation report (SER) for WBN Units 1 and 2, issued June 1982 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML072060490), and in subsequent supplements to the SER (SSERs), the NRC staff reviewed and evaluated climatology and meteorology in the site vicinity, as well as the WBN meteorological program. To address subsequent meteorological phenomena, TVA has revised WBN Unit 2 FSAR Section 2.3 to update selected topics of the Watts Bar regional climatology and local meteorology that may be pertinent to engineering design and safety assessments.

TVA confirmed that it had reviewed and updated the relevant meteorological parameters and analysis, as appropriate. In general, it obtained the information used in the review from recent data summaries from the U.S. Department of Commerce National Climatic Data Center (NCDC), analysis reports based upon NCDC data, and measurements made at the WBN site. TVA primarily used NCDC data summaries for the following locations in Tennessee: Chattanooga; Knoxville; Dayton; Decatur; Oak Ridge; Rhea County, in which the WBN site is located; and the six counties surrounding Rhea County. The review included consideration of changes in the limiting values resulting from the addition of recently measured data to the compiled historic climate data base and resultant changes in locations that were found to be more limiting than locations identified in prior FSAR amendments.

2.3.1 Regional Climatology

In Section 2.3.1 of FSAR Amendment 101 (ADAMS Accession No. ML103140314), TVA provided revised information on average and limiting values associated with tornadoes, strong winds and storms, hail, lightning, and snowfall resulting from consideration of the more recently measured NCDC and WBN site data.

TVA also updated the assessment of the probability that a tornado would strike the WBN site and the associated recurrence interval. TVA's current estimate of tornado strike probability, based on a longer period of record and a larger area of tornado occurrence than previously used, is 0.00074 per year (74 chances out of 100,000 of a tornado striking the WBN site in any given year) with a recurrence interval of 1,351 years, which is more limiting than the previous estimate. The NRC staff performed a comparison calculation and confirmed the TVA tornado strike probability estimate; therefore, the estimate is acceptable.

In addition, TVA stated that it updated the design conditions assumed for the WBN reactor shield building and other safety-related structures to be consistent with the appropriate guidance in Regulatory Guide (RG) 1.76, Revision 1, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," issued March 2007. This is a revision from the planned original design basis for WBN Unit 2, which was consistent with Revision 0 of RG 1.76, "Design-Basis Tornado for Nuclear Power Plants," issued April 1974. The change reflects the use of the updated guidance and is acceptable.

With regard to short-duration high winds, TVA generated maximum 3-second gust equivalent approximations, in addition to providing updates to include consideration of recent high winds. This facilitated a comparison and use of measurements made with variable averaging periods (e.g., fastest mile wind speed, 2-minute average, hourly average) for Chattanooga, Knoxville, and the WBN site to identify the highest case. Use of the maximum 3-second gust is consistent with current NCDC reporting practice and more up-to-date engineering analysis applications, and is acceptable.

Based on sampling the revised information provided by TVA, the NRC staff has concluded that TVA used acceptable references and information to develop the updates.

2.3.2 Local Meteorology

In Section 2.3.2 of WBN FSAR Amendment 101, dated October 29, 2010, TVA revised information on average and limiting values associated with temperature, precipitation, snowfall, atmospheric water vapor content, fog, and onsite wind measurements resulting from consideration of the more recently measured NCDC and WBN site data. Based on sampling the revised information provided, the NRC staff has concluded that TVA used acceptable references and information to develop the updates.

2.3.3 Onsite Meteorological Measurements Program

TVA described several updates in equipment and procedures. TVA also stated that it developed the WBN onsite meteorological program to be consistent with the guidance given in RG 1.23, Revision 1, "Meteorological Monitoring Programs for Nuclear Power Plants," issued March 2007, which is a revision from the previous phase of the program, developed to be consistent with the guidance in RG 1.23, Revision 0, "Onsite Meteorological Programs," issued February 1972. The NRC staff finds the use of this RG version acceptable.

In addition, TVA provided tables of joint windspeed, wind direction, and atmospheric stability data for onsite meteorological measurements made from 1974 through 1993. SSER 15 (ADAMS Accession No. ML072060488) discussed these data, but the tables, which are an update of previous tables for 1974 through 1988, were not included in prior amendments because of an oversight. The NRC staff finds this replacement acceptable.

2.3.4 Short-Term (Accident) Diffusion Estimates

The NRC staff previously addressed this section in SSER 15. TVA revised the reference number for Table 2.3-64a to Table 2.3-65. The NRC staff finds this change to be editorial and, therefore, acceptable.

2.3.5 Long-Term (Routine) Diffusion Estimates

TVA did not identify any revisions to this section. The NRC staff previously addressed this section in SSER 14 (ADAMS Accession No. ML072060486).

2.4 Hydrologic Engineering

2.4.9 Transport of Liquid Releases

2.4.9.2 Dispersion, Dilution, and Travel Times of Accidental Releases of Liquid Effluents

The WBN Unit 2 FSAR provides details on the accidental release of radiological or nonradiological liquid into the Tennessee River. The FSAR describes the model and calculations that TVA developed to determine the reduction in concentration of a release as it progresses downstream, with particular emphasis on the concentrations at the surface water intakes downstream of the plant. The model predicted the maximum concentration observed on the Tennessee River at two downstream locations. FSAR Amendment 92 identified these two intakes as Dayton and the Volunteer Army Ammunition Plant. In FSAR Amendment 93, TVA updated the name of the "Volunteer Army Ammunition Plant" to "East Side Utility."

The NRC staff has concluded that the change to the name of the intake is administrative and did not affect the location or relative concentration result associated with the intake. Since the change does not affect the conclusions identified in the FSAR, the staff finds it acceptable.

2.5 Geological, Seismological, and Geotechnical Engineering

2.5.6 Embankments and Dams

The site has no embankments that are used for plant flood protection or for impounding the cooling water required to operate the nuclear power plant.

2.6 References

SER Section 2.6 contains a list of references used in the NRC staff's original evaluation. The staff reviewed Chapter 2 of the original WBN FSAR, dated September 27, 1976 (ADAMS Legacy Library Accession No. 4005002724), and subsequent FSAR amendments, and determined that the FSAR has never contained a Section 2.6. References used in the WBN FSAR and in the staff's subsequent SSERs are included within the applicable sections.

3 DESIGN CRITERIA

3.2 Classification of Structures, Systems, and Components

During its review of WBN Unit 2 FSAR Amendment 95, dated November 24, 2009, the staff of the U.S. Nuclear Regulatory Commission (NRC) identified a number of editorial and typographical errors in Section 3.2, Table 3.2-2. By letter dated November 9, 2010, Tennessee Valley Authority (TVA) acknowledged the errors and stated that the errors will be corrected in a future FSAR amendment. TVA's response is acceptable to the staff.

3.2.2 System Quality Group Classification

General Design Criterion (GDC) 1, "Quality Standards and Records," of Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," requires that nuclear power plant systems and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. These fluid system pressure-retaining components are part of the reactor coolant pressure boundary (RCPB) and other fluid systems important to safety, where reliance is placed on these systems to conduct the following functions: (1) prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary, (2) permit shut down of the reactor and maintain it in a safe shutdown condition, and (3) retain radioactive material.

In FSAR Amendment 95, dated November 24, 2009, TVA changed FSAR Tables 3.2-2 and 3.2-5 to reclassify the Unit 2 Primary Water Storage Tank (PWST) as TVA Safety Class G (NRC Quality Group D, Safety Class NNS). The tank was initially procured to ASME Section III, Class 3 requirements and has subsequently been downgraded to API-650. This change is consistent with the similar change for Unit 1, as described in EDC E-51860-A, wherein it was determined that the PWST is not required to mitigate any design basis events. Therefore, downgrading of this tank to Safety Class G does not create the potential for changes to the probability of an accident or malfunction and does not contribute to accident radiological consequences.

The NRC staff concludes that the PWST has been classified in an acceptable manner and has been constructed to quality standards commensurate with the importance of its function. The specification of API-650 is consistent with the guidance of RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," for the construction of Quality Group D atmospheric storage tanks, as stated in the SER.

3.5 Missile Protection

3.5.1 Missile Selection and Description

3.5.1.3 Turbine Missiles

During its review, the NRC staff identified an open item to review TVA's testing frequency of once every 6 months for turbine valves. The staff identified the open item during its review of Sections 3.5.1 and 3.5.3 of WBN Unit 2 Final Safety Analysis Report (FSAR) Amendment 99, regarding the test frequency for turbine valves documented in References 18, 19, and 20 of

FSAR Section 3.5. Reference 18 is topical report WCAP-16501-P, Revision 0, "Extension of Turbine Valve Test Frequency Up to 6 Months for BB-296 Siemens Power Generation (Westinghouse) Turbines with Steam Chests," issued February 2006, Westinghouse Electric Company, LLC, Westinghouse Proprietary Class 2. Reference 19 is Technical Instruction (TI)-227, "Turbine Integrity Program with Turbine Overspeed Protection (TIPTOP)," Revision 3, dated October 24, 2008, TVA Watts Bar Nuclear (WBN). Reference 20 is EC-02262, "Missile Generation Risk Assessment for Original and Retrofit Nuclear HP Rotors," dated December 17, 2002, Siemens Westinghouse Power Corporation. FSAR Section 3.5.1.3 cited WCAP-16501-P and EC-02262 to support the use of a test frequency of once every 6 months for turbine valves. TI-227 is related to the test frequency for turbine valves.

The NRC staff evaluated the acceptability of using a test frequency of once every 6 months for turbine valves. In Section 3.5.1.3 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (hereafter referred to as the SRP), the staff accepted the approach of protecting essential structures, systems, and components (SSCs) against turbine missiles by limiting the missile generating frequency, denoted as P1, to under $1x10^{-4}$ per year for a favorably oriented unit. TVA based its calculations on the probability of turbine overspeed in WCAP-16501-P, documented in the Seimens Energy, Inc., report CT-27467, "Missile Report, TVA Watts Bar 2, BB281-13.9m2," dated October 6, 2009 (Reference 13 of FSAR Section 3.5). The staff reviewed CT-27467 and confirmed that the calculations are in accordance with NRC-approved methodology, as documented in a safety evaluation dated March 30, 2004, for the Siemens 13.9 M^2 turbine (Agencywide Documents Access and Management System (ADAMS) Accession No. ML040930616). In addition, the staff determined that the calculated P1 value based on a test frequency of 6 months for turbine valves is acceptable, based on a margin of two orders of magnitude between the calculated P1 value and the NRC criterion of $1x10^{-4}$ per year for a favorably oriented unit. This margin is large enough to cover the uncertainty of using the probability of turbine overspeed in WCAP-16501-P as input to the plant-specific P1 calculation.

Since TVA's calculations used NRC-approved methodology and had a large margin of safety between the calculated P1 value and the NRC criterion, the NRC staff finds that the proposed test frequency of once every 6 months for turbine valves is acceptable, and the open item is closed.

3.5.2 Structures, Systems, and Components To Be Protected from Externally Generated Missiles

The NRC staff used the following regulatory requirements and guidance in its review of FSAR Section 3.5.2.

- GDC 4, "Environmental and Dynamic Effects Design Bases," states, in part, that "Structures, systems, and components important to safety...shall be appropriately protected against dynamic effects, including the effects of missiles..."

- Regulatory Guide (RG) 1.13, "Spent Fuel Storage Facility Design Basis," as it relates to the spent fuel pool systems and structures being capable of withstanding the effects of externally generated missiles and preventing missiles from contacting stored fuel assemblies

- RG 1.27, "Ultimate Heat Sink for Nuclear Power Plants," as it relates to the ultimate heat sink (UHS) and connecting conduits being capable of withstanding the effects of externally generated missiles

- RG 1.115, "Protection Against Low-Trajectory Turbine Missiles," as it relates to the protection of SSCs important to safety from the effects of turbine missiles

- RG 1.117, "Tornado Design Classification," as it relates to the protection of SSCs important to safety from the effects of tornado missiles

TVA has identified safety-related structures, systems, and components, including plant outdoor features and air intakes and exhausts, which may be required to perform a safety function coincident with or following the occurrence of a tornado, and that require protection from externally generated missiles. These structures are designed to withstand the effects of postulated tornado-generated missiles, including vertical missiles, without damage to safety-related equipment. Section 3.5.1.4 of this report discusses the tornado spectrum. Safety-related systems and components and stored fuel are within tornado-missile protected structures or are provided with tornado-missile barriers, with the exception of the refueling water storage tank (RWST). Failure of the RWST will not compromise the safe controlled shutdown of the plant because the required makeup water for shutdown is usually obtained from the boric acid storage tanks through the chemical volume and control system (CVCS), which is located in the auxiliary building. Section 9.3.4 of this report discusses further the reliability and capability of the CVCS to supply borated water and makeup water to the reactor coolant system in the event of small breaks or leaks, and to function as part of the emergency core cooling system.

Essential piping from the outdoor and essential raw cooling water (ERCW) intake structure is protected from missiles throughout its length by missile protection slabs or seismic Category I pipe tunnels. The UHS, which is the Tennessee River waterway complex, does not require protection from externally generated missiles. Therefore, TVA has satisfied the requirements of GDC 4 with respect to missile protection and the specific guidance of RGs 1.13, 1.27, 1.115, and 1.117 concerning tornado-missile protection for safety-related SSCs, including stored fuel and the UHS. Sections 9.2.1 and 9.2.5 of this report further discuss the UHS and the ERCW systems.

Based on its review of Section 3.5.2 of Amendment 97 to the WBN FSAR, the NRC staff concludes that those SSCs identified by TVA as requiring protection from externally generated missiles conform to the relevant regulatory requirements and are, therefore, acceptable.

3.6 Protection against the Dynamic Effects Associated with the Postulated Rupture of Piping

3.6.1 Plant Design for Protection against Postulated Piping Failures in Fluid Systems Outside Containment

The NRC staff reviewed Section 3.6A, "Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping (Excluding Reactor Coolant System Piping)," of Amendment 100 to the WBN Unit 2 FSAR, and concluded that there are no substantive differences from WBN Unit 1 regarding plant design for protection against postulated failures in fluid systems outside containment.

Based on the above and prior staff evaluations documented in NUREG-0847 and its supplements, the staff concludes that the plant design and protection of those areas and systems required for safe shutdown following a postulated event, including the combination of pipe failures and single active failures, meets the criteria of NRC Branch Technical Positions Auxiliary Systems Branch (ASB) 3-1 and Mechanical Engineering Branch (MEB) 3-1, where appropriate. Therefore, the staff concludes that the design meets the requirements of GDC 4 regarding protection against pipe failures in fluid systems outside containment and is acceptable.

3.6.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

In FSAR Amendment 95, TVA modified Section 3.6B.2 to state that it used the computer code MULTIFLEX 3.0 for the loss-of-coolant accident (LOCA) analysis to replace the SATANV computer code. The NRC staff approved the use of MULTIFLEX 3.0 for WBN Unit 1 on September 30, 2003, in License Amendment No. 46. In FSAR Amendment 95, TVA inserted the exact wording describing the use of MULTIFLEX 3.0 from Section 3.6B.2 of the WBN Unit 1 FSAR into Section 3.6B.2 of the WBN Unit 2 FSAR. In response to the NRC staff's request for additional information (RAI), TVA indicated that the inputs to and results from MULTIFLEX 3.0, which were the same for WBN Units 1 and 2 for the reactor vessel internals, are also applicable to FSAR Section 3.6B.2. However, during a conference call on August 27, 2010, in response to an additional RAI, TVA explained that the inputs to and results from MULTIFLEX 3.0 were the same for both WBN Unit 1 and Unit 2, except for the inputs for the steam generators, because the WBN Unit 1 steam generators have been replaced. The TVA clarification is acceptable. The staff confirmed that all other changes in Section 3.6B.2 of the WBN Unit 2 FSAR Amendment 95 were editorial or clarifying in nature. Therefore, the staff finds TVA's changes and modifications to Section 3.6B.2 of FSAR Amendment 95 to be acceptable.

3.6.3 Leak-Before-Break Evaluation Procedures

In FSAR Amendment 95, Section 3.6B, TVA addressed the design bases, analytical methods, and dynamic analysis used to determine the dynamic response of the reactor coolant loop associated with postulated pipe breaks in the loop piping. The NRC staff used the following regulatory requirements in its review of this section:

- GDC 4 states, in part, that "[SSCs] important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents." It further states that "dynamic effects associated with pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping."

- 10 CFR 50.55a(a)(1) states that "[SSCs] must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed."

The NRC staff's review of Section 3.6B focused on resolving variations between the WBN Unit 1 and Unit 2 baseline FSARs and items found in the supplemental safety evaluation reports

(SSERs) that are applicable to this section. In SSER 5, the staff approved the elimination of the dynamic effects of postulated primary loop ruptures from the design basis of WBN Units 1 and 2, using "leak-before-break" technology, as permitted by GDC 4.

In its review, the NRC staff noted that TVA had modified the analysis methodology related to the evaluation of the WBN Unit 2 reactor coolant hydraulic forcing functions and response model. This modification was made to reflect the use of the MULTIFLEX 3.0 computer code to model the complex, nonlinear thermal-hydraulic loadings. The MULTIFLEX program calculates the pressure, velocity, and force transients in reactor primary coolant systems during the subcooled, transition, and early saturation portions of the blowdown caused by a LOCA. The NRC staff approved the use of this computer code for these evaluations for WBN Unit 1 in License Amendment No. 46 (ADAMS Accession No. ML032740199). The NRC staff concluded that TVA's changes to the thermal-hydraulic analysis methodologies are acceptable, based on the prior acceptance of these methodologies for WBN Unit 1, and because the changes do not affect the leak-before-break results.

The NRC staff has determined that other changes made in Section 3.6B are not substantive or are editorial in nature and do not affect the original leak-before-break analysis results.

The NRC staff determined that it is possible that the primary water stress-corrosion cracking (PWSCC) degradation mechanism exists in the Alloy 600 dissimilar metal butt welds (DMBWs) in the primary loop piping. This issue is Open Item 15 (Appendix HH) until TVA confirms to the NRC staff the completion of PWSCC mitigation activities on the Alloy 600 DMBWs in the primary loop piping.

Based on the above discussion, the NRC staff concludes that the leak-before-break evaluation methodology used in WBN Unit 2 FSAR Amendment 95 is consistent with the NRC-approved methodology documented in the safety evaluation report (SER) and its supplements. The leak-before-break evaluation methods are consistent with SRP Section 3.6.3 and are, therefore, acceptable, pending the resolution of Open Item 15 regarding the completion of PWSCC mitigation activities.

3.7 Seismic Design

3.7.3 Seismic Subsystem Analysis

3.7.3.18 Seismic Qualification of Main Control Room Suspended Ceiling and Air Delivery Components

The NRC previously reviewed and approved the seismic qualification methodology for the suspended ceiling and air delivery components in the main control room in License Amendment No. 50, dated February 12, 2004 (ADAMS Accession No. ML040430624). Since WBN Units 1 and 2 share a common control room, TVA has applied to Unit 2 the Unit 1 methodology of qualifying the main control room components. The NRC staff has reviewed TVA's submittal and confirmed that the methodology and results pertaining to Unit 1 are applicable to Unit 2. Therefore, the staff considers this section resolved.

3.9 Mechanical Systems and Components

3.9.1 Special Topics for Mechanical Components

The NRC staff used the following regulatory requirements in its review of Section 3.9.1 of FSAR Amendment 97:

- GDC 1, "Quality Standards and Records," states, in part, that "[SSCs] important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function."

- GDC 2 states, in part, that "[SSCs] important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes...without loss of capability to perform their safety functions."

- GDC 14, "Reactor Coolant Pressure Boundary," states that "[t]he reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture."

- GDC 15, "Reactor Coolant System Design," states that "[t]he reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences."

- Criterion III, "Design Control," in Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 states, in part, that "[m]easures shall be established to assure that applicable regulatory requirements and the design basis, as defined in § 50.2 and as specified in the license application, for those [SSCs] to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions."

SRP Section 3.9.1 contains the guidance the NRC staff used to determine whether TVA has met the above regulations.

The NRC staff's review focused on variations between the WBN Unit 1 and Unit 2 baseline FSARs and resolving open items found in the SER and SSERs that are applicable to this section. SSER 6 identified an issue related to TVA's piping evaluation for a postulated main feedwater (FW) header rupture transient, which results in a water hammer event caused by a rapid check valve closure. The analyses originally performed by TVA included an assumption that certain FW piping system supports failed when the loads exceeded their calculated capacities; this was listed as an open item in SSER 6 (tracked as Outstanding Issue (OI) 20(a) for SSER 6, Section 3.9.1).

In SSER 13, the NRC staff accepted TVA's analytical approach and noted that the analysis performed, which postulated pipe support failures, was acceptable, based on the difficulty in making subsequent pipe support modifications and on the low probability of a water hammer transient. The NRC staff requested additional information regarding the validity of the conclusions reached in SSER 13 regarding OI 20(a), as those conclusions relate to the current WBN Unit 2 refurbishment efforts. In response, TVA confirmed, by letter dated July 31, 2010 (ADAMS Accession No. ML102290258), that the main FW piping analyses performed for WBN Unit 2 are similar to the analyses performed for WBN Unit 1. TVA also stated that the WBN Unit 2 pipe support designs and pipe support stiffness values used in the analyses were the same as those used for the WBN Unit 1 piping analyses. TVA stated that six snubbers that were included in the main FW analyses for WBN Unit 1 are not present in WBN Unit 2. However, in accordance with the analyses discussed in SSER 13, these snubbers were assumed to fail in the check valve slam transient analyses. Therefore, the absence of these snubbers is insignificant. Since TVA performed the main FW piping analyses for WBN Units 1 and 2 in a similar fashion with no significant variations, the NRC staff concludes that the piping analyses documented in SSER 13 are valid for the WBN Unit 2 refurbishment and are, therefore, acceptable.

Based on the review of Section 3.9.1 of Amendment 97 to the WBN Unit 2 FSAR, as described above, the NRC staff concludes that TVA complies with the regulatory requirements relevant to this section. Therefore, the open item (SSER 6 OI 20(a) for Section 3.9.1) is closed.

3.9.2 Dynamic Testing and Analysis of Systems, Structures, and Components

The NRC staff used the following regulatory requirements in its review of Section 3.9.2 of FSAR Amendment 97:

- 10 CFR 50.55a(a)(1) states that "[SSCs] must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed."

- GDC 1 states, in part, that "[SSCs] important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed."

- GDC 2 states, in part, that "[SSCs] important to safety shall be designed to withstand the effects of natural phenomena...without loss of capability to perform their safety functions."

- GDC 4 states, in part, that "[SSCs] important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents."

- GDC 14 states that "[t]he reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture."

- GDC 15 states that "[t]he reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences."

- 10 CFR Part 50, Appendix B, Criterion III, states, in part, that "[m]easures shall be established to assure that applicable regulatory requirements and the design basis, as defined in § 50.2 and as specified in the license application, for those [SSCs] to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions."

SRP Section 3.9.2 contains the guidance used by the NRC staff to evaluate whether an applicant has met the above regulations.

The NRC staff's review focused on variations between the WBN Unit 1 and Unit 2 baseline FSARs and resolving open items found in the SER and SSERs that are applicable to this section. In its review of Section 3.9.2, the NRC staff noted that the analysis methodology related to the evaluation of the WBN Unit 2 reactor vessel internals under faulted loading conditions had been modified in its entirety. TVA made this modification to reflect the use of the MULTIFLEX, LATFORCE, FORCE-2, and WECAN computer codes to model the complex, nonlinear thermal-hydraulic loadings induced on the reactor internals to verify their structural adequacy. In License Amendment No. 46 for WBN Unit 1, the NRC staff approved the use of these codes for these evaluations. TVA confirmed, in a letter dated July 31, 2010 (ADAMS Accession No. ML102290258), that no variations existed between the analyses for WBN Units 1 and 2. Since the NRC staff previously accepted, for WBN Unit 1, the changes to the analysis methodologies found in FSAR Section 3.9.2.5, the staff concludes that the changes are also acceptable for WBN Unit 2.

In its review, the NRC staff also identified variations in the WBN Units 1 and 2 criteria for the maximum deflections of the reactor vessel internals under design-basis events, found in Table 3.9-5. This table provides the maximum allowable and no-loss-of-function limits for the reactor vessel internals under design-basis loading conditions, which are based in part on the WCAP-5890 report. In response to an NRC staff RAI regarding the variations in the WBN Units 1 and 2 deflection limits, TVA revised the limits in FSAR Amendment 100. Therefore, the limits for WBN Units 1 and 2 are identical for the reactor vessel internal deflection limits during design-basis events. Since it previously reviewed and approved the limits and the derivations for the limits for WBN Unit 1, the staff concludes that the deflection limits are also acceptable for WBN Unit 2.

Based on the review of Section 3.9.2 described above, the NRC staff concludes that TVA complies with the regulatory requirements relevant to this section.

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

The NRC staff used the following regulatory requirements in its review of Section 3.9.3 of FSAR Amendment 97:

- 10 CFR 50.55a(a)(1) states that "[SSCs] must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed."

- GDC 1 states, in part, that "[SSCs] important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed."

- GDC 2 states, in part, that "[SSCs] important to safety shall be designed to withstand the effects of natural phenomena...without loss of capability to perform their safety functions."

- GDC 4 states, in part, that "[SSCs] important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents."

- GDC 14 states that "[t]he reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture."

- GDC 15 states that "[t]he reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences."

SRP Section 3.9 contains the guidance used by the NRC staff to evaluate whether an applicant has met the aforementioned regulations.

The NRC staff's review focused on variations between the WBN Unit 1 and Unit 2 baseline FSARs and resolving open items found in the SER and SSERs that are applicable to this section. In SSER 4, the NRC staff identified Open Item 2 regarding compressive stresses imposed on short column pipe supports at WBN Units 1 and 2, which exceeded the buckling criteria margin established by the NRC. By letter dated May 14, 1984, TVA provided details of a sampling program used to conclude that the Class 2 and 3 pipe supports at WBN comply with the applicable NRC design criteria. By letter dated July 31, 2010 (ADAMS Accession No. ML102290258), TVA confirmed that the Class 2 and 3 pipe supports at WBN Unit 2 continue to meet the NRC acceptance criteria discussed in SSER 4.

The NRC staff also reviewed OI 19(h) from SSER 6. This outstanding issue was related to TVA's use of earthquake experience data to seismically qualify Category I(L) piping at WBN. The NRC staff noted in SSER 8 that TVA had developed a screening criterion to identify items in Category I(L) piping systems that may require further evaluation, based on this earthquake experience data. The NRC staff found this screening criterion adequate for demonstrating the seismic ruggedness of Category I(L) piping. In its letter dated July 31, 2010 (ADAMS Accession No. ML102290258), TVA stated that it used the same screening and acceptance criterion for Category I(L) piping at WBN Unit 2 that it used for WBN Unit 1.

Coupled with the Category I(L) piping seismic qualification, the NRC staff noted in SSER 8 that TVA had indicated it would use a safety factor of three in its evaluation of concrete expansion anchor bolts for Category I(L) pipe supports. In SSER 8, the NRC staff accepted the use of this safety factor value to validate the existing design of concrete expansion anchors used for Category I(L) piping systems, based on TVA's implementation of recommendations, including additional concrete inspections, anchor spacing, and concrete edge distance in conjunction with the existing anchor bolts. However, the NRC staff noted in SSER 8 that, for future Category I(L) piping, TVA should use the required safety factors for these piping systems found in NRC Office of Inspection and Enforcement (IE) Bulletin 79-02, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts, dated March 8, 1979. In its letter dated July 31, 2010 (ADAMS Accession No. ML102290258), TVA confirmed that it conducted the evaluation of the WBN Unit 2 Category I(L) piping supports in the same fashion as those in WBN Unit 1. Additionally, TVA confirmed that it designs new Category I(L) piping supports using the safety factors described in the aforementioned IE Bulletin 79-02 (Watts Bar Design Criteria 20-32).

In addition to the items associated with previously identified SSER issues, the NRC staff reviewed the changes made to Sections 3.9.3.2 and 3.9.3.3 as part of Amendment 97 to the WBN Unit 2 FSAR. The NRC staff finds that the changes made to these sections are only editorial and are, therefore, acceptable. However, the NRC staff's review also found that TVA deleted a number of safety-related valves from Tables 3.9-17 and 3.9-25. In its letter dated August 6, 2010 (ADAMS Accession No. ML102210440), TVA provided its justification for deleting the valves from the tables. Generally, the affected valves were no longer in the plant, had been retagged, or had no active safety function. The NRC staff finds that TVA provided adequate justification for deleting the valves from the tables, and the deletions are, therefore, acceptable.

Based on its review of Section 3.9.3 of Amendment 97 to the WBN Unit 2 FSAR, as described above, the NRC staff concludes that TVA complies with the regulatory requirements relevant to this section.

3.9.6 Inservice Testing of Pumps and Valves

The inservice testing (IST) program for pumps and valves is intended to demonstrate that they will maintain operational readiness at all times during the plant's lifetime. The purpose of these tests and parameter measurements is to detect long-term degradation, and TVA is required to perform them in accordance with 10 CFR 50.55a. According to 10 CFR 50.55a(f)(4)(i), TVA must perform the IST of certain American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 pumps and valves at initial and subsequent 120-month (10-year) intervals, in accordance with the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code), incorporated by reference in the regulations.

WBN Unit 2 FSAR Amendment No. 97, Section 3.9.6, states that "IST of ASME Code Class 1, 2, and 3 pumps and valves will be conducted to the extent practical in accordance with 2001 Edition of ASME OM Code with Addenda through 2003." This statement is inconsistent with the requirements of 10 CFR 50.55a(f)(4)(i). In its letter to NRC dated July 31, 2010 (ADAMS Accession No. ML102290258), TVA noted that the above sentence in FSAR Amendment 97 was in error. TVA corrected the error in FSAR Amendment 100, which reads, "IST of ASME Code Class 1, 2, and 3 pumps and valves will be conducted to the extent practical in accordance with latest Edition and Addenda of the ASME OM Code incorporated by reference in 10 CFR 50.55a(b) on the date 12 months before the date of issuance of OL [operating license] for Unit 2 as required by 10 CFR 50 55a(f)." In its letter of July 31, 2010, TVA further stated that, once the date for issuance of an OL becomes more certain, the licensee will update the above sentence in Section 3.9.6, in accordance with 10 CFR 50.55a(f)(4)(i). Therefore, the NRC staff finds FSAR Amendment 100 acceptable for complying with the regulation.

The NRC staff review under SRP Section 3.9.6 covered TVA's program for preservice and inservice testing of pumps and valves and emphasized those areas of the test program for which TVA requested relief from the requirements of the OM Code. To complete the staff's review on schedule, TVA is expected to submit an IST program and specific relief requests for WBN Unit 2 nine months before the projected date of OL issuance. Currently, the development and submittal of an acceptable IST program for the WBN Unit 2 is Open Item 13 (Appendix HH). The NRC will include its evaluation of the IST program in a future supplement to the SER before it issues an OL for WBN Unit 2.

3.11 Environmental Qualification of Mechanical and Electrical Equipment

The NRC staff reviewed FSAR Amendments 95 and 100 (ADAMS Accession Nos. ML093370275 and ML102530216, respectively), and supplemental information provided by TVA in letters dated August 6, August 30, and September 1, 2010 (ADAMS Accession Nos. ML102210440, ML102510580, and ML102500170, respectively), as well as in Section 3.11 of the original SER supplement prepared for the environmental qualification (EQ) of mechanical and electrical equipment (SSER 15).

The NRC based its acceptance criteria for this review on the requirement that equipment used to perform a necessary safety function must be demonstrated capable of maintaining functional operability under all service conditions postulated to occur during its installed life, and for the time it is required to operate in response to any postulated accident conditions. GDC 1, 2, 4, and 23, "Protection System Failure Modes," and Appendix B to 10 CFR Part 50, Criteria III, XI, "Test Control," and XVII, "Quality Assurance Records," contain this requirement, which applies to safety-related equipment located both inside and outside containment. More detailed requirements and guidance relating to the methods and procedures for demonstrating this capability appear in 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," and in NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," Revision 1, issued July 1981.

RG 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," and NUREG-0588 supplement the Institute of Electrical and Electronics Engineers (IEEE) Standard 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," which contains detailed information about the qualification of electrical equipment. For EQ, equipment at WBN must meet the Category II criteria in NUREG-0588, Revision 1, until TVA modifies or replaces a Category II component, in which case, TVA must comply with 10 CFR 50.49(l). This regulation requires that replacement equipment be qualified in accordance with the provisions of 10 CFR 50.49 (i.e., meet Category I criteria), unless there are sound reasons to the contrary.

3.11.1 Background

The NRC staff issued NUREG-0588 in December 1979 to promote a more orderly and systematic implementation of EQ programs by the industry and to provide guidance to the NRC staff to use in ongoing licensing reviews. The positions contained in the NUREG provide guidance on (1) how to establish environmental service conditions, (2) how to select methods considered appropriate for qualifying equipment in different areas of the plant, and (3) other specific topics, such as margin, aging, and documentation.

In February 1980, the staff requested that applicants for near-term OLs review and evaluate the EQ documentation for each item of safety-related electric equipment and identify the degree to which their qualification programs complied with the staff positions discussed in NUREG-0588.

NRC Bulletin 79-01b, "Environmental Qualification of Class IE Equipment," dated January 14, 1980, and its supplements provided EQ criteria and guidance for operating reactors and license applicants. The final rule on EQ for electrical equipment important to safety for nuclear power plants became effective on February 22, 1983. This rule, 10 CFR 50.49, specifies the requirements for demonstrating the EQ of electrical equipment that is important to safety and that is located in a harsh environment.

In June 1982, the NRC staff issued NUREG-0847, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2," which is the SER for WBN Units 1 and 2. In Section 3.11 of the SER, the staff cited the requirements of NUREG-0588 and stated that it would review information on the EQ of electrical and mechanical equipment once TVA submitted it. In this regard, the EQ program for WBN Unit 1 began in 1983, when TVA first submitted information. Between February 14 and February 16, 1984, the NRC staff and its consultant (Idaho National Engineering Laboratories (INEL)) audited the EQ files for WBN Unit 1. As documented in its audit report of March 14, 1984, the NRC staff determined that the EQ files were incomplete.

The staff documented more information about the evolution of the WBN Unit 1 EQ program in NUREG-1232, Volume 4, "Safety Evaluation Report on Tennessee Valley Authority: Watts Bar Nuclear Performance Plan, Watts Bar Unit 1," issued January 1990. In Section 3.3.4 of that report, the NRC staff stated that, in July and August 1985, TVA conducted a management review of the EQ programs at the Sequoyah and Browns Ferry Nuclear Plants and at WBN Unit 1. TVA found that much of the qualification documentation was not fully auditable and, in some cases, the available documentation did not demonstrate full qualification. On the basis of this review, TVA established EQ projects at the three plants with the responsibility for developing and implementing EQ programs at each site. These programs were to establish controls to ensure a consistent approach to EQ at all TVA sites.

TVA submitted "Watts Bar Nuclear Plant Unit 1, Summary Status Update Report of TVA's Compliance to 10 CFR 50.49—Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," to the NRC for review on September 30, 1986, after it established the WBN Unit 1 EQ project. TVA supplemented this submittal by letter to the NRC dated April 30, 1991. The NRC staff and its contractor, INEL, reviewed these submittals and identified deficiencies and open items in the WBN Unit 1 EQ program. The staff met with TVA on March 5, 1992 (meeting summary dated March 13, 1992; ADAMS Accession No. ML073550432), and resolved some of the deficiencies and open items. The staff issued an RAI on May 1, 1992. TVA sent additional information in a letter to the NRC dated February 17, 1993. The staff inspected the WBN Unit 1 EQ program to resolve any remaining open items before fuel load.

3.11.2 Evaluation

The NRC staff notes that the evaluation of EQ documented in the SER was only applicable to WBN Unit 1. Based on the unique aspects of EQ, the staff must perform a detailed inspection and evaluation before fuel load to determine how the WBN Unit 2 EQ program complies with the requirements of 10 CFR 50.49. This is Open Item 16 (Appendix HH). The inspection must examine TVA's overall EQ organization and interfaces, EQ program procedures, EQ program documentation (e.g., 10 CFR 50.49 equipment and cable lists, EQ qualification files, environmental drawings, category and operating time calculations, and essentially mild calculations), EQ engineering support, EQ maintenance program, EQ training, and quality assurance interfaces with EQ. The scope of the inspection should include evaluating the qualification criteria and the environments in which the equipment must function, assessing the qualification documentation, and examining the physical installation of the equipment. The area of review should focus on the qualification of safety-related electrical equipment that must function to prevent or mitigate the consequences of a LOCA or a high-energy-line break inside or outside the containment while subjected to the harsh environments associated with these accidents.

To complete its review of the WBN Unit 2 EQ program, the NRC staff must perform the following tasks to verify that full EQ of all equipment within the scope of 10 CFR 50.49 is achieved by fuel load:

- Examine electrical equipment on site.
- Complete an audit of TVA's qualification documentation and environmental qualification documentation binders.
- Review the acceptability of the components, qualification methods, and accident environments.

These tasks, to be completed by the NRC staff before fuel load at WBN Unit 2, are considered part of Open Item 16 (Appendix HH).

3.11.2.1 Completeness of Environmental Qualification List

The regulations at 10 CFR 50.49 require that TVA or the licensee prepare a list of electric equipment important to safety, as defined in the section. In general, this includes electrical equipment important to safety located in harsh environments. The regulation requires that the list be auditable and current, and the equipment must remain on the list for the entire period during which it is installed in the plant.

TVA has not yet compiled a complete EQ list for WBN Unit 2. Therefore, the verification of the accuracy of the EQ list before fuel load is Open Item 17 (Appendix HH).

3.11.2.2 Qualification Methods

In its FSAR amendments and supplemental letters, TVA stated that it conducted qualification tests and analyses for safety-related electrical equipment in accordance with the requirements of 10 CFR 50.49 and the guidelines of NUREG-0588, Revision 1. In NUREG-0588, Revision 1, the NRC staff presents detailed procedures for qualifying safety-related electrical equipment in a harsh environment. The NUREG-0588 criteria apply to equipment that is important to safety, as defined in 10 CFR 50.49.

The staff concluded that the TVA qualification methods are consistent with NRC regulations and are, therefore, acceptable. However, based on the extensive layup period of equipment within WBN Unit 2, the NRC staff must review, before fuel load, the assumptions used by TVA to reestablish a baseline for the qualified life of the equipment. The purpose of the staff's review is to ensure that TVA has addressed the effects of environmental conditions on equipment during the layup period. This is Open Item 18 (Appendix HH), and the NRC should address it before fuel load at WBN Unit 2.

3.11.2.2.1 Electrical Equipment in a Harsh Environment

The NRC staff has not yet completed its review of TVA's EQ program procedures for WBN Unit 2. This is Open Item 19 (Appendix HH), which the NRC should complete before fuel load.

During its review, the NRC staff identified the following issues that require further discussion and review:

- The NRC staff asked TVA to confirm that equipment being replaced or refurbished will be qualified as Category I, as required by 10 CFR 50.49. In its response, dated September 1, 2010, TVA stated that "replaced equipment is being procured qualified to the NUREG-0588, Category I requirements." TVA also stated, however, that it is refurbishing a small number of components. The NRC staff reviewed these components as noted below:

    - 6.9 kV motors (all to be refurbished but only one to be rewound)

      TVA considers the refurbishment activities to be routine maintenance that does not affect the qualification of the motors. The staff considers that routine maintenance activities should result in increasing the EQ of the motors to Category I status, in accordance with 10 CFR 50.49. Therefore, the staff considers resolution of this issue to be Open Item 20 (Appendix HH).

      TVA intends to rewind one containment spray pump motor. TVA stated that the materials for this motor will be those qualified in the Electric Power Research Institute's motor rewind test program, which meets the NUREG-0588, Category I criteria. The NRC staff considers this approach to be acceptable for WBN Unit 2.

    - electrical penetration assemblies (EPAs)

      TVA stated that the EPAs are qualified to NUREG-0588, Category I. The WBN Units 1 and 2 EPAs have the identical design made by Conax Nuclear and were procured and installed around the same time.

      TVA is refurbishing the EPAs by replacing the missing and damaged feedthrough modules with new ones of identical design and type and by adding some new feedthrough modules qualified to NUREG-0588, Category I.

      The NRC staff should confirm that the EPAs are installed in the tested configuration and that the feedthrough module is manufactured by the same company and is consistent with the EQ test report for the EPA. This is Open Item 21 (Appendix HH).

    - terminal blocks

TVA stated that it noted during a field walkdown that the terminal blocks installed in the EQ application for WBN Unit 2 are General Electric-made CR151B. According to TVA, these terminal blocks are qualified to NUREG-0588, Category I. TVA further noted that it issued design modifications to inspect the EQ terminal blocks and replace those that are damaged or degraded with equivalent CR151B blocks.

TVA must clarify its use of the term "equivalent" (e.g., identical, similar) regarding the replacement terminal blocks. If the blocks are similar, then TVA should complete and present a similarity analysis to the NRC for review. This issue is Open Item 22 (Appendix HH).

- main steam isolation valves (MSIVs)

TVA stated that the original WBN Unit 2 MSIVs are qualified to NUREG-0588, Category II. TVA is refurbishing the MSIVs by replacing the missing subcomponents, including the solenoid valves, terminal blocks, and manufacturing wiring, with components of the same make, model, and type as supplied on the original purchase order. With the exception of the solenoid valves, the subcomponents will be qualified to NUREG-0588, Category I requirements. According to TVA, the solenoid valves cannot be upgraded because equivalent subcomponents qualified to NUREG-0588, Category I, requirements are not available.

The NRC staff does not consider TVA's justification for not upgrading the MSIV solenoid valves to Category I, as specified in 10 CFR 50.49(l), to be a sound reason. Resolution of this issue is Open Item 23 (Appendix HH).

- Cables purchased under Contract 81K5-830078

TVA stated that it purchased identical cables under Rockbestos Contracts 80K7-826542 and 81K5-830078. TVA installed the cable purchased under Contract 80K7-826542 before the February 22, 1983, cutoff date stipulated in RG 1.89, Revision 1, for Category II qualification, and therefore it meets the full RG 1.89 and NUREG-0588 requirements for Category II qualification. TVA's verification of the purchasing and installation documentation for cables purchased under Rockbestos Contract 81K5-830078 determined that it purchased the cables in 1982 but installed them in 1985. RG 1.89, Revision 1, Regulatory Position C.6, states that "replacement electric equipment installed subsequent to February 22, 1983, must be qualified in accordance with the provisions of §50.49 unless there are sound reasons to the contrary...." TVA stated that it had identical equipment (i.e., Rockbestos cable) on hand as part of the utility's stock before February 22, 1983. Therefore, TVA chose not to upgrade this cable to Category I when installing it in 1985.

The NRC staff concludes that, since the cable from Contract 81K5-830078 is of identical design, and the installation of the cable is the same as that supplied in Contract 80K7-826542, and since TVA installed the cable shortly after February 22, 1983, the cable purchased under Contract 81K5-830078 is equally qualified and acceptable under the NRC EQ requirements (i.e., the cable is acceptable in accordance with 10 CFR 50.49(l)). Any future EQ cable replacements must meet the Category I requirements of 10 CFR 50.49. Therefore, the staff concludes that the cable is acceptable for WBN Unit 2.

- The NRC staff requested that TVA clarify that, for a mild environment, the threshold for electronic components such as semiconductors or electronic components containing organic material is a total integrated dose of less than 1×10^3 rads, and that a threshold for a mild radiation environment for other equipment is less than 1×10^4 rads. In its response, dated August 30, 2010, TVA stated that the upper threshold for the EQ mild environment is 1×10^4 rads for all components, which is consistent with WBN Unit 1. Since certain components may have a lower threshold for impacts from radiation exposure, TVA stated that it requires the following at WBN:

  - Per design standards, any device, whether in a mild, essentially mild, or harsh environment, must be purchased to the environmental conditions where it is to be located. That includes the usual guidelines (e.g., radiation, temperature, humidity). An environmental conditions data sheet with the stated environmental parameters is sent to the manufacturer, along with the purchase request, and TVA asks the vendor to meet those conditions.

  - For metal oxide semiconductors, complementary metal oxide semiconductors, or like circuitry, the same holds true. But the design standard for the qualification of electrical equipment in a harsh environment specifically requires that any circuitry of this type be evaluated on a case-by-case basis for any gamma dose exceeding 1×10^3 rads.

  The NRC staff requires supporting documentation from TVA to justify its establishment of a mild environment threshold for a total integrated dose of less than 1×10^3 rads for electronic components, such as semiconductors, or electronic components containing organic material. This issue is Open Item 24 (Appendix HH).

- The NRC staff asked TVA to explain the basis for reducing the period of exposure after an accident from 1 year to 100 days. In its response, TVA noted that the 100-day postaccident operating time is consistent with the WBN Unit 1 EQ program. Since the NRC previously reviewed and accepted this criterion for WBN Unit 1 (SSER 15), the staff considers this issue resolved.

3.11.3 Conclusion

To determine the adequacy of the WBN Unit 2 EQ program, the NRC staff reviewed FSAR Amendments 95 and 100; the supplemental information provided by TVA in letters dated August 6, August 30, and September 1, 2010; and Section 3.11 of the original SSER prepared for the EQ of mechanical and electrical equipment (SSER 15).

Based on its review, the staff has insufficient information to conclude that the TVA EQ program conforms to the requirements of 10 CFR 50.49; the relevant parts of GDC 1, 2, 4, and 23; Criteria III, XI, and XVII of Appendix B to 10 CFR Part 50; and the criteria specified in NUREG-0588. The staff will update this SSER upon satisfactory closure of the open items identified in Appendix HH, consistent with the staff's approach to the review and acceptance of the WBN Unit 1 EQ program.

3.13 Threaded Fasteners

In SSER 21, Section 1.7, the NRC staff identified Section 3.13.0 as an issue but did not list the issue status. NRC Bulletin 82-02, "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants," dated June 2, 1982, addressed threaded fasteners. In its letter dated March 20, 2008, TVA committed to implementing the actions of NRC Bulletin 82-02 in WBN Unit 2, using the same approach as it used on Unit 1. NRC Inspection Report 50-390/85-08 and 50-391/85-08, dated March 29, 1985, documented receipt and review of TVA's response to Bulletin 82-02, and documented closure of the Bulletin for WBN Unit 1, based upon the NRC's verification of TVA's actions.

The NRC staff concludes that TVA's approach to addressing this issue for WBN Unit 2 is acceptable, based upon its commitment to implement Bulletin 82-02 for WBN Unit 2, using the same approach as at Unit 1.

5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1 Compliance with Codes and Code Cases

5.2.1.4 Applicable Code Cases

In the Watts Bar Nuclear (WBN) Unit 2 FSAR Section 5.2.1.4, "Applicable Code Cases," the Tennessee Valley Authority (TVA) identified specific American Society of Mechanical Engineers (ASME) Code cases whose requirements have been applied in the construction of pressure-retaining ASME Code, Section III, Class 1 components within the reactor coolant pressure boundary (RCPB). As noted by the U.S. Nuclear Regulatory Commission (NRC) staff in SER Section 5.2.1.2, "the basis for acceptance in the staff review has been the Code cases found to be acceptable in Regulatory Guides 1.84...and 1.85 [withdrawn in 2003]...and the Code cases previously found to be acceptable by the staff for plants similar to Watts Bar, before publication of the Regulatory Guides. The staff concludes that compliance with the requirements of these Code cases will result in a component quality level that is commensurate with the importance of the safety function of the RCPB, constitutes an acceptable basis for satisfying the requirements of GDC 1, and is, therefore, acceptable."

During its review of TVA's WBN Unit 2 Final Safety Analysis Report (FSAR) Amendment 97, dated January 11, 2010, the NRC staff questioned TVA's use of American Society of Mechanical Engineers (ASME) Code Case 1423-2, "Wrought Type 304 and 316 with Nitrogen Added, Sections I, III, VIII, Division 1 and 2," without committing to the limitations and modifications listed in Regulatory Guide (RG) 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III," for this Code case. By letter dated November 9, 2010, TVA responded to the staff, stating the following:

> Amendment 97 to the Unit 2 FSAR inadvertently incorporated Code Case 1423-2 into Table 5.2-8...A future amendment to Unit 2 FSAR Table 5.2-8 will remove the reference to Code Case 1423-2 for the branch nozzles material specifications. A change to Section 5.2.1.4 will not be necessary because the future amendment will reconcile Table 5.2-8 and Section 5.2.1.4.

TVA's response is acceptable to the staff.

5.2.3 Reactor Coolant Pressure Boundary Materials

The NRC staff reviewed FSAR Section 5.2.3, "Reactor Coolant Pressure Boundary Materials," in accordance with Section 5.2.3, Revision 1, dated July 31, 1981, of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (hereafter referred to as the SRP). The staff reviewed the reactor coolant pressure boundary (RCPB) with regard to material specifications and compatibility with the reactor coolant and evaluated the RCPB in terms of its ability to meet the following regulatory requirements:

- Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," General Design Criterion (GDC) 1, "Quality Standards and Records," and GDC 30, "Quality of Reactor Coolant Pressure Boundary," as they relate to quality standards for design, fabrication, erection, and testing of structures, systems, and components (SSCs)

- GDC 4, "Environmental and Dynamic Effects Design Bases," as it relates to the compatibility of components with environmental conditions

- GDC 14, "Reactor Coolant Pressure Boundary," and GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," as they relate to minimizing the probability of rapidly propagating fracture and gross rupture of the RCPB

- Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50, Criterion XIII, "Handling, Storage and Shipping," as it relates to onsite material cleaning control

- Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50, as it relates to materials testing and acceptance criteria for fracture toughness of the RCPB

- 10 CFR 50.55a, "Codes and Standards," as it relates to quality standards applicable to the RCPB

Materials selected for the reactor coolant system (RCS) components must be compatible with reactor coolant water chemistry, thermal insulation materials, and the atmosphere. The specific processes (including heat treatment and welding practices) used to fabricate the RCS components must maximize the corrosion resistance and fracture toughness of the components.

SRP Section 5.2.3 contains the relevant NRC regulatory requirements for this area of review and the associated acceptance criteria.

Materials Specifications

In WBN Unit 2 FSAR Amendment 98, dated May 7, 2010, TVA made changes to the RCPB materials specifications of Class 1 primary components listed in FSAR Table 5.2-8. In general, the changes consist of the addition of other material types or grades within the same ASME Code material specification (e.g., the addition of ASME SA182, "Standard Specification for Forged or Rolled Alloy-Steel Pipe Flanges, Forged Fittings, and Valves and Parts for High-Temperature Service," and Type F316 for reactor vessel (RV) nozzle safe ends, which previously listed SA 182 Type F304L only). For some components, additional ASME specifications for the same material type, but a different product form, were added (e.g., SB-166 was added for the control rod drive mechanism or emergency core cooling system appurtenances on the RV upper head, where previously only SB-167 was listed). For a few components, material meeting various ASME Code cases was added.

The NRC staff verified that all of the changes and additions to the materials specifications in FSAR Table 5.2-8 are ASME Code-approved materials (i.e., those listed in the ASME Code, Section II), with the exception that SA182 Grade 316 (ASME Code Cases 1423-1 and 1423-2) is specified for branch nozzles in the reactor coolant piping. RG 1.84, Revision 34, lists ASME Code Case 1423-2 as having been formerly conditionally approved by the staff but annulled by ASME in 1977. Before its annulment by ASME, Code Case 1423-2 was acceptable, subject to the condition of compliance with the recommendations contained in RGs 1.31, "Control of

Ferrite Content in Stainless Steel Weld Metal," and 1.44, "Control of the Use of Sensitized Stainless Steel." In general, ASME annuls Code cases when their provisions are incorporated into the current edition of the ASME Code. The staff verified that the 2004 Edition of SA182, which is part of the most recent ASME Code edition to be incorporated by reference in 10 CFR 50.55a, allows Type 316 stainless steel with nitrogen added (Grade F316N). Therefore, the provisions of ASME Code Case 1423-2 have been incorporated into the ASME Code, and WBN has met the conditions that were previously required for use of this Code case. Since the code of record for fabrication of the WBN Unit 2 reactor coolant piping is ASME Code, Section III, 1971 Edition through Winter 1971 Addenda, Code Case 1423-2 was not yet annulled at that time. In addition, although the conditions previously required by the staff for use of this AMSE Code case, specifically compliance with RGs 1.31 and 1.44, were not incorporated into the ASME Code, the staff concluded in the safety evaluation report (SER) (NUREG-0847, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2," issued June 1982) that the controls imposed upon austenitic stainless steel are either consistent with RGs 1.31 and 1.44, or, if they are not consistent with these RGs, the staff has previously accepted the positions and actions taken. Since the provisions of ASME Code Case 1423-2 have been incorporated into the current ASME Code, and TVA has met the conditions previously required by the staff for use of this Code case for all austenitic stainless steels, the NRC staff finds the use of this ASME Code case acceptable.

The NRC staff finds that the changes made by TVA to the materials specifications meet the requirements of either a version of the ASME Code incorporated by reference in 10 CFR 50.55a or ASME Code cases that have been accepted by the staff and therefore conform to the requirements of 10 CFR 50.55a. Thus, the staff finds the materials specifications acceptable.

Compatibility with Reactor Coolant

In the SER, the NRC staff concluded that the RCPB met regulatory requirements, including, in part, GDC 14. Accordingly, in SSER 21, the staff found no open items for Section 5.2.3 of the SER. However, in FSAR Amendment 97, TVA included a description of the process for zinc addition to FSAR Section 5.2.3.4, "Chemistry of Reactor Coolant," for the purpose of reducing radionuclide content in the primary system corrosion films. The residual zinc content would be maintained at a concentration of from 2 to 8 parts per billion. FSAR Table 5.2-10 also indicates that zinc is limited to less than 40 parts per billion during normal power operation.

SRP Section 5.2.3 refers to SRP Section 9.3.4, "Chemical and Volume Control System," for the review of reactor coolant chemistry as it relates to corrosion control and compatibility with RCPB materials. Revision 2 of SRP Section 9.3.4 does not provide any guidance addressing zinc addition. However, SRP Section 9.3.4, Revision 3, recommends, in part, that the reviewer evaluate the proposed chemistry program with respect to that described in the latest version in the Electric Power Research Institute (EPRI) report series entitled, "Pressurized Water Reactor Primary Water Guidelines." With respect to zinc, the EPRI pressurized-water reactor (PWR) primary water chemistry guidelines consider zinc to be a diagnostic parameter, meaning that no specific range of zinc concentration is required. The EPRI PWR primary water chemistry guidelines also state that the sampling frequency for zinc is based on the plant-specific zinc injection program and that nickel analysis may also be required for high-duty plants, defined as plants with cores that have a relatively high tendency for subcooled nucleate boiling (SNB). The EPRI guidelines refer to another EPRI document—the PWR zinc application guidelines.

Based on the industry experience with zinc addition, the NRC staff has no concerns with pressure boundary degradation from zinc addition. Industry experience has shown that zinc reduces general corrosion rates, may have a beneficial effect on reducing primary water stress-corrosion cracking (PWSCC) initiation, and may reduce PWSCC growth rates. Industry experience has also demonstrated the beneficial effects of zinc on reducing radiation dose.

Because of limited experience with zinc addition to high-duty cores, industry guidelines recommend that plants with high-duty cores take additional measures to ensure that there are no adverse effects on the fuel, such as excessive or uneven crud buildup that can lead to crud-induced power shift, also known as axial offset anomaly (Refs. 1 and 2). Core duty is a measure of the potential for SNB to occur in the core and is a function of several thermal-hydraulic parameters. The tendency for SNB can be quantified by means of the High Duty Core Index (HDCI), calculated in accordance with Appendix F to Reference 3. Because it is not clear whether the WBN Unit 2 core is considered high duty, and, if so, what additional measures will be taken to ensure fuel performance, such as a fuel surveillance program, the NRC staff requested additional information (Request for Information (RAI) 5.2.3-1) from TVA. In Enclosure 1 of its response letter dated July 31, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML102290258), TVA stated that the WBN Unit 2 core would be considered high duty if the bounding values of power, flow, and outlet temperature listed in the FSAR are used. However, TVA stated that, if the best-estimate values are used in the core-duty calculation as recommended by Appendix F to Reference 3, the resulting HDCI for WBN Unit 2, Cycle 1, is in the "Medium Duty Core" category.

TVA's response to RAI 5.2.3-1 also stated that zinc injection at WBN Unit 2 is bounded by the existing experience base in Westinghouse-designed reactors, including WBN Unit 1. Therefore, TVA considers a fuel surveillance program specifically for WBN Unit 2 to be unnecessary. However, TVA stated that Westinghouse performs for each operating plant cycle a fuel crud risk analysis that includes the consideration of zinc addition, if zinc is used. TVA further stated that Westinghouse also provides zinc addition guidelines to each PWR that adds zinc. These guidelines describe the chemistry monitoring requirements needed for zinc addition.

Based on TVA's consideration of operating experience related to zinc and the consideration of zinc addition in cycle-specific crud risk analyses, the NRC staff concludes that TVA has taken adequate measures to prevent adverse effects on fuel from zinc addition; therefore, TVA's actions are acceptable.

GDC 14, "Reactor coolant pressure boundary," states that "the [RCPB] shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture." In Section 5.2.3 of the SER, the NRC staff concluded that TVA met the requirements of GDC 14. Based on the staff's review of the information provided by TVA in FSAR Amendment 97, as supplemented by letter dated July 31, 2010, regarding zinc addition to the primary system, the staff concludes that the changes to the reactor coolant chemistry are compatible with the RCPB materials and that the integrity of the RCPB will not be adversely affected. Therefore, the requirements of GDC 14 continue to be met, and TVA's proposed changes are acceptable.

The staff also concludes the changes to the materials specifications proposed by TVA in WBN Unit 2 FSAR Amendment 98 meet 10 CFR 50.55a, since the specifications are either ASME-approved or the materials meet NRC staff-approved code cases.

5.2.3.1 References for Section 5.2.3

1. Electric Power Research Institute, "Overview Report on Zinc Addition in Pressurized Water Reactors—2004," 1009568 Final Report, December 2004.

2. Electric Power Research Institute, "Pressurized Water Reactor Primary Water Zinc Application Guidelines," 1013420 Final Report, December 2006.

3. Electric Power Research Institute, "PWR Axial Offset Anomaly (AOA) Guidelines," 1008102 Final Report, June 2004.

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

The NRC staff reviewed Section 5.2.7 of WBN Unit 2 FSAR Amendment 101, dated October 29, 2010, and concluded that TVA had made no substantive changes to the reactor coolant leak detection systems from those previously reviewed by the staff in the SER and its supplements.

Based on the above and the previous staff evaluations, as documented in the SER and its supplements, the NRC staff concludes that the RCPB leakage detection systems are diverse and provide reasonable assurance that identified and unidentified primary system leakage will be detected in a timely manner.

The systems meet the requirements of GDC 30 with respect to RCPB leakage detection and identification, as well as the guidelines of RG 1.45, "Guidance on Monitoring and Responding to Reactor Coolant System Leakage," Revision 1, issued May 2008, with respect to the RCPB leakage detection system design. Therefore, the staff finds these systems acceptable.

5.3 <u>Reactor Vessel</u>

5.3.1 Reactor Vessel Materials

The NRC staff reviewed FSAR Section 5.3.1, "Reactor Vessel Materials," in accordance with SRP Section 5.3.1, Revision 1, "Reactor Vessel Materials," issued July 1981. The staff reviewed the RV material specifications, fracture toughness properties, RV materials surveillance program, and special processes used in manufacturing and fabrication of the RV (such as welding and nondestructive examination (NDE)) using the following regulatory requirements:

- GDC 1 and 30, as they relate to quality standards for design, fabrication, erection, and testing of SSCs
- GDC 4, as it relates to the compatibility of components with environmental conditions
- GDC 14, as it relates to prevention of rapidly propagating fractures of the RCPB
- GDC 31, as it relates to material fracture toughness
- GDC 32, "Inspection of Reactor Coolant Pressure Boundary," as it relates to the requirements for a materials surveillance program

- 10 CFR 50.55a, as it relates to quality standards for design and determination and monitoring of fracture toughness

- 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation," as it relates to RCPB fracture toughness and the material surveillance requirements of Appendix G and Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50

- Appendix B to 10 CFR Part 50, Criterion XIII, as it relates to onsite material cleaning control

- Appendix G to 10 CFR Part 50, as it relates to materials testing and acceptance criteria for fracture toughness

- Appendix H to 10 CFR Part 50, as it relates to the determination and monitoring of fracture toughness

- 10 CFR 50.61, "Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events," as it relates to fracture toughness criteria for PWRs relevant to pressurized thermal shock (PTS) events

Fracture Toughness

In Section 5.3.1.1 of the SER, the NRC staff approved two exemptions to the requirements of paragraph IV.A.1 of Appendix G to 10 CFR Part 50, regarding determination of the initial material nil ductility reference temperature (RT_{NDT}). Paragraph IV.A.1 of Appendix G requires that RV beltline materials be tested to the requirements of Appendix G of Section XI of the ASME Code. These include the RCPB materials. These exemptions were related to the requirements of paragraph NB 2330 of the ASME Code as applied to the RCPB materials. TVA (1) did not perform drop-weight tests and instead performed Charpy V-Notch (C_v) impact tests at a single temperature for the intermediate-to-lower shell beltline weld, and (2) oriented C_v specimens for the RV materials outside the beltline in the longitudinal rather than the transverse direction. The staff granted the exemptions based on data for similar welds provided by TVA that established RT_{NDT} at 0 degrees Fahrenheit (F) for the intermediate-to-lower shell beltline weld and adding 30 degrees F to the temperature at which the C_v values for absorbed energy and lateral expansion were acceptable. (Note: Reference 2 provides a different value for the unirradiated RT_{NDT} of this weld; see discussion of PTS in this section of the SSER).

In Section 5.3 of the SER, the NRC staff also approved an exemption to the minimum upper shelf energy (USE) requirements of paragraph IV.B of Appendix G to 10 CFR Part 50 for the intermediate shell forging of WBN Unit 1 only. This exemption was not required for WBN Unit 2. Subsequently, the NRC changed Appendix G to 10 CFR Part 50 to allow licensees to submit an equivalent margins analysis for RVs with low USE. By letter dated October 15, 1993, TVA withdrew its request for exemption and submitted a low USE evaluation. In SSER 14, issued December 1994, the NRC staff evaluated and found acceptable TVA's analysis demonstrating equivalent margins of safety to those required by Appendix G to 10 CFR Part 50 for the material with low USE and determined that the previously approved exemptions were no longer needed (see Section 5.3.1.1.1 of SSER 14).

Reactor Vessel Materials Surveillance Program

In WBN Unit 2 FSAR Amendment 97, dated January 11, 2010, TVA made several revisions, including the following:

(1) In Section 5.4.3.6, TVA revised the surveillance capsule withdrawal to indicate that the tentative schedule for removal of the capsules for postirradiation testing is shown in Table 4.0-1 of the pressure and temperature limits report (PTLR). However, the PTLR does not contain this information. The PTLR for WBN Unit 2 is contained in report WCAP-17035-NP, "Watts Bar Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation" (Ref. 1).

(2) In Section 5.2.4.2, TVA changed the description of the C_v specimen orientation. The original description stated that, "Initial upper shelf fracture energy levels for materials of the reactor vessel beltline region (including welds), as determined by Charpy-V-Notch Test on specimens oriented in the transverse direction of the base material, are established for the reactor vessel irradiation surveillance test program." In FSAR Amendment 97, TVA changed the orientation from "transverse" to "tangential and axial."

(3) In Section 5.2.4, TVA changed the stated lead factor of the surveillance capsules. The FSAR states that the surveillance program will comply with American Society for Testing and Materials (ASTM) E185-82, "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels," and Appendix H to 10 CFR Part 50, with the exception that all four RV irradiation surveillance capsules will receive a fluence which is at least four times the maximum RV fluence (i.e., the lead factor for all the capsules will be at least 4). The previous lead factor was 3.6.

Since the PTLR does not provide the schedule for removal of the capsules for postirradiation testing, and neither the FSAR nor PTLR describe the RV materials surveillance program, the NRC staff asked TVA to provide additional information in Questions 1 and 2 of RAI 5.3.1-1 (see Enclosure 1 of TVA's letter dated July 31, 2010, ADAMS Accession No. ML102290258). The staff requested that TVA include the surveillance capsule withdrawal schedule and a description of the RV material surveillance program in the PTLR, including a discussion of how the specimen examinations will be used to update the pressure-temperature (P-T) curves.

In response to RAI 5.3.1-1, Question 1, TVA stated that it would add the following proposed surveillance capsule withdrawal schedule to the PTLR (as Table 4.01):

Table 5.3.1-1
Watts Bar Unit 2 Surveillance Capsule Removal Schedule[a]

Capsule	Orientation of Capsule	Lead Factor	Removal Time	Expected Capsule Fluence (n/cm^2, E > 1.0 MeV)
U	Dual 34°	5.13	1st refuel outage	0.50×10^{19}
W	Single 34°	5.18	6.1 EFPY	3.17×10^{19} (b)
X	Dual 34°	5.13	6.2 EFPY to 12.5 EFPY(c)	3.17×10^{19} to 6.34×10^{19} (c)
Z	Single 34°	5.18	Standby	-
V	Dual 31.5°	4.40	Standby	-
Y	Dual 31.5°	4.40	Standby	-

(a) This information is taken from the withdrawal schedule contained in WCAP-9455, Revision 3 (Ref. 3).

(b) Approximate fluence at vessel inner wall at end of life (EOL) (32 effective full-power years (EFPYs)).

(c) Capsule X should be withdrawn between 6.2 EFPY and 12.5 EFPY, which corresponds to a capsule fluence of not less than once (3.17×10^{19} neutrons per square centimeter (n/cm^2) (E > 1.0 million electron volts (MeV)) or greater than twice (6.34×10^{19} n/cm^2 (E > 1.0 MeV)) the peak EOL vessel fluence. This is consistent with the recommendations of ASTM E185-82.

Additionally, TVA provided a description of the surveillance program, indicating the following:

- The RV material irradiation surveillance specimens shall be removed and examined to determine changes in material properties. The results of these specimen examinations shall be used to update the P-T curves.
- The RV surveillance program is in compliance with Appendix H to 10 CFR Part 50.
- The material test requirements and the acceptance standard utilize RT_{NDT}, which is determined in accordance with ASTM E208, "Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels."
- The empirical relationship between RT_{NDT} and the fracture toughness of the RV steel is developed in accordance with Appendix G, "Fracture Toughness Criteria for Protection Against Failure," to Section XI of the ASME Code.
- The surveillance capsule removal schedule meets the requirements of ASTM E185-82.

In its response to RAI 5.3.1-1, TVA further stated that Westinghouse provided a proposed revision of the PTLR and that, in accordance with this markup, it had updated the RV material surveillance program as indicated above. TVA also stated that the WBN Unit 2 PTLR is included in the Unit 2 system description for the RCS (TVA document WBN2-68-4001), which will be revised to reflect required revisions to the PTLR by September 17, 2010.

The NRC staff reviewed TVA's proposed surveillance capsule withdrawal schedule using the guidance in ASTM E185-82. The recommended number of capsules and withdrawal schedule, consistent with ASTM E185-82, depends on the magnitude of the predicted transition temperature (RT_{NDT}) shift at the vessel inside surface. For vessels having a predicted shift of less than or equal to 100 degrees F, such as WBN Unit 2, ASTM E185-82 recommends that a minimum of three capsules be withdrawn. ASTM E185-82 recommends that the first capsule be withdrawn at approximately 6 EFPY or an accumulated neutron fluence of 5×10^{18} n/cm^2 (E > 1.0 MeV), whichever comes first; the second capsule be withdrawn at approximately 15 EFPY or a neutron fluence corresponding to the EOL fluence, whichever comes first; and the third capsule be withdrawn at a neutron fluence not less than once or greater than twice the peak EOL RV fluence. The staff reviewed the proposed withdrawal EFPY and neutron fluence for WBN Unit 2 and concludes that the proposed surveillance schedule, which will be included in the PTLR, meets the recommendations of ASTM E185-82 and is, therefore, acceptable. Thus, TVA's response to RAI 5.3.1-1, Question 1, is acceptable.

In response to RAI 5.3.1-1, Question 2 (Enclosure 1 of TVA's letter dated July 31, 2010), TVA stated that the RV materials surveillance program will comply with Appendix H to 10 CFR Part 50. Compliance with 10 CFR Part 50, Appendix H, is based on compliance with an edition of ASTM E185 that is acceptable to the staff, since 10 CFR Part 50, Appendix H, incorporates by reference several editions of ASTM E185 up to 1982.

However, TVA's proposed program description also stated that the material test requirements and the acceptance standard utilize the RT_{NDT}, which is determined in accordance with ASTM E208. While ASTM E208 is the standard for drop-weight tests, which are part of the determination of initial RT_{NDT}, C_v tests are also required in accordance with ASME Code, Section III, Subarticle NB-2300. FSAR Section 5.2.4.1 indicates that, since the WBN Unit 2 RV was designed to an edition of the ASME Code that pre-dates the present requirements of ASME Code, Section III, Subarticle NB-2300, as augmented by Appendix G to 10 CFR Part 50, the fracture toughness of the RV materials were assessed by using methods described in NRC Branch Technical Position (BTP) Materials Engineering Branch (MTEB) 5-2, "Fracture Toughness Requirements for Older Plants." BTP MTEB 5-2 allows the use of a combination of drop-weight and C_v tests and provides procedures that may be used if drop-weight tests were not performed. However, RT_{NDT} is not determined by drop-weight tests alone, and drop-weight tests are typically only performed for initial RT_{NDT} determination, not irradiated RT_{NDT}. TVA should provide additional information to clarify how the initial and irradiated RT_{NDT} was determined. This is Open Item 44 (Appendix HH).

TVA's description of the RV materials surveillance program also stated that the empirical relationship between RT_{NDT} and the fracture toughness of the RV steel is developed in accordance with the ASME Code, Section XI, Appendix G. The NRC staff concludes that use of ASME Code, Section XI, Appendix G, for the relationship between fracture toughness and RT_{NDT} is acceptable because 10 CFR Part 50, Appendix G, references ASME Code, Section XI, Appendix G, as the acceptable methodology for determining P-T limits, which take into account fracture toughness.

Regarding TVA's change to the stated lead factor of the surveillance capsules, FSAR Section 5.2.4 now indicates that the surveillance program will comply with ASTM E185-82 and Appendix H to 10 CFR Part 50, with the exception that all four RV irradiation surveillance capsules have lead factors of at least 4. ASTM E185-82 defines the lead factor as the ratio of the peak neutron fluence (E > 1 MeV) of the specimens in a surveillance capsule to the peak neutron fluence (E > 1 MeV) at the reactor pressure vessel inside surface. The previous lead factor was 3.6. Since ASTM E185-82 recommends that the capsule lead factors be in the range of 1 to 3, the NRC staff asked in RAI 5.3.1-2, Question 1, that TVA provide a justification and verification of the validity of the surveillance data for the capsules with the higher lead factors. The staff also asked TVA to confirm that all capsules with lead factors greater than 3 will provide metallurgically meaningful data, in terms of the expected design life or licensed life of the RV, including possible license renewal terms, based on the fluences these capsules are projected to receive (RAI 5.3.1-2, Question 2).

In its response letter dated July 31, 2010, TVA stated that it will assess the validity of the accelerated irradiation data by comparing the Unit 2 surveillance data to RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," predictions. TVA further stated that it will assess the validity of the data by comparing the Unit 2 surveillance data to results for similar forging and weld material to ascertain that the observed trends are consistent. TVA noted that the predictions in RG 1.99, Revision 2, and the available data on similar materials from other plants represent data irradiated under a wide range of lead factors; thus, the Unit 2 surveillance data will be verified by trending it against the RG 1.99, Revision 2, predictions and the data for similar surveillance materials. The NRC staff finds this response acceptable because TVA is comparing the surveillance program results for the WBN Unit 2 materials to similar materials irradiated in other reactors with a wide range of lead factors. This should ensure that the trends in the changes in fracture toughness properties for the WBN Unit 2 materials are not radically different than similar materials irradiated with lower lead factors. The staff also notes that more recent versions of ASTM E185, such as ASTM E185-10, "Standard Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels," allow lead factors of up to 5. The staff therefore finds that TVA's response to RAI 5.3.1-2, Question 1, is acceptable.

In response to RAI 5.3.1-2, Question 2, TVA stated that, because all six of the WBN Unit 2 lead factors are greater than 3, all capsules will be removed early in life. TVA further stated that Unit 2 will be able to store capsules in the spent fuel pool for future reinsertion to ensure that the RV continues to be monitored throughout the licensed life (and during potential license extensions). The NRC staff finds that TVA's response to RAI 5.3.1-2, Question 2, is acceptable because removal early in life and storage in the spent fuel pool will ensure that the standby capsules are available later for future reinsertion, if necessary.

Regarding C_v specimen orientation, the original description in the FSAR stated that, "Initial upper shelf fracture energy levels for materials of the Reactor Vessel Beltline Region (including welds), as determined by Charpy V-Notch Test on specimens oriented in the transverse direction of the base material, are established for the reactor vessel irradiation surveillance test program." TVA has changed the orientation from "transverse" to "tangential and axial."

ASTM E185-82 states the following:

> [T]he tension and Charpy specimens from base metal shall be oriented so that the major axis of the specimen is parallel to the surface and normal to the major working direction for plates, or normal to the major working direction for forgings as described in Section III of the ASME Code. The axis of the notch of the Charpy specimen for base metal and weld metal shall be oriented perpendicular

to the surface of the material; for the HAZ [heat-affected zone] specimens, the axis of the notch shall be as close to perpendicular to the surface as possible so long as the entire length of the notch is located within the HAZ.

In RAI 5.3.1-3, the NRC staff asked TVA to clarify the C_v specimen orientation in terms of the ASTM E185-82 language. In its response letter dated July 31, 2010, TVA clarified that the orientation of specimens described as being in the "tangential" orientation for forgings are those that would typically be described as oriented in the "longitudinal" or "strong" direction for plate material, while those whose orientation is described as "axial" for forgings are those that would be described as oriented in the "transverse" or "weak" direction for plate material. TVA's response provided the necessary clarification and is, therefore, acceptable to the staff.

The table in FSAR Section 5.4.6.3 provides the number and orientation of specimens (C_v, tensile, fracture toughness, and bend) for each material type. The notes to the table indicate that, for the limiting base materials, each capsule contains specimens oriented both in the major working direction (longitudinal) and normal to the major working direction (transverse). TVA did not change the table in FSAR Amendment 97.

Since the RV materials surveillance program description provided by TVA states that the program meets the requirements of 10 CFR Part 50, Appendix H (thereby meeting the fracture toughness monitoring requirements of 10 CFR 50.60), and determines the relationship between RT_{NDT} and fracture toughness of the RV in accordance with 10 CFR Part 50, Appendix G, the NRC staff concludes, pending resolution of Open Item 44 (Appendix HH), that the program description meets the applicable NRC regulatory requirements and is, therefore, acceptable. The staff also concludes that the surveillance capsule withdrawal schedule proposed by TVA meets ASTM E185-82 requirements (thereby meeting 10 CFR Part 50, Appendix H, requirements) and is, therefore, acceptable.

Pressurized Thermal Shock

In SSER 11, issued April 1993, the NRC staff evaluated information provided by TVA on reactor vessel fluence for compliance with 10 CFR 50.61 for WBN Units 1 and 2. The staff concluded that the WBN RVs satisfied the requirements of 10 CFR 50.61; however, the staff stated that, based on surveillance data from other reactor vessels, the margin term for base metal in the PTS rule may be nonconservative and future flux reduction may be needed. The staff also stated that the need for flux reduction would be confirmed by the withdrawal and testing of surveillance specimens for 10 CFR Part 50, Appendix H requirements, and was outside the scope of SSER 11.

In WBN Unit 2 FSAR Amendment 97, TVA modified the table in Section 5.2.4.3 to add the PTS reference (RT_{PTS}) values for WBN Unit 2. TVA also added the fluence values for the RV inner surface at EOL and per year for WBN Unit 2 to this section. TVA provided the details of the RT_{PTS} evaluation in Appendix C to the PTLR.

The acceptance criteria for RT_{PTS} values are based on the provisions of 10 CFR 50.61. In accordance with 10 CFR 50.61(b), values of RT_{PTS} projected, using the procedures outlined in 10 CFR 50.61, for the period from the initial application submittal to the projected expiration date of the operating license must not exceed the screening criteria of 132 degrees Celsius (C) (270 degrees F) for plates, forgings, and axial weld materials and 149 degrees C (300 degrees F) for circumferential weld materials. For WBN Unit 2, TVA provided projected EOL RT_{PTS} values of 79 degrees F for lower shell forging 04, 88 degrees F for intermediate shell forging 05, and 95 degrees F for intermediate-to-lower shell circumferential weld seam 05. These values are well below the screening criteria of 270 degrees F for plates and 300 degrees F for circumferential welds and are, therefore, acceptable.

The NRC staff checked the copper, nickel, and unirradiated RT_{NDT} ($RT_{NDT(u)}$) values against the values submitted in TVA's response to Generic Letter (GL) 92-01, "Reactor Vessel Structural Integrity," dated February 28, 1992 (Ref. 2). The values used in TVA's PTS evaluation are identical to the values submitted to the NRC in Reference 2. TVA's responses to GL 92-01, Revision 1, Supplement 1 (Refs. 3 and 4), did not result in changes to the copper, nickel, or $RT_{NDT(u)}$ values reported for WBN Unit 2. The staff closed its review of TVA's response to GL 92-01 in Reference 5.

On January 4, 2010 (75 FR 13), the NRC amended its regulations at 10 CFR 50.61 to provide alternate fracture toughness requirements for protection against PTS events for PWR pressure vessels. The final rule provides alternate PTS requirements based on updated analysis methods. The NRC amended the rule because the existing requirements were based on unnecessarily conservative probabilistic fracture mechanics analyses. TVA is not using the alternate requirements allowed by 10 CFR 50.61a, which are optional. However, the staff confirmed that the margin terms used by TVA in determining RT_{PTS} meet the requirements of the current version of 10 CFR 50.61. Therefore, the concern noted in SSER 11, that the margin term for base metal in the PTS rule is nonconservative, is resolved.

The NRC staff performed a confirmatory calculation of the RT_{PTS} values using TVA's projected 32 EFPY fluence and the copper and nickel content provided for the limiting beltline materials. The staff's calculation confirmed TVA's RT_{PTS} values. However, in RAI 5.3.1-4, the staff questioned whether the assumption of 32 EFPY as the EOL fluence was conservative, since many nuclear plants are now operating at capacity factors in excess of 90 percent, which would result in a higher EFPY value at 40 calendar years of operation.

In its response letter dated July 31, 2010, to RAI 5.3.1-4, TVA stated that, for reactors with actual operating experience, a plant-specific calculation is performed for fuel cycles that have been completed to provide a best-estimate fluence, and fluence projections for future operation are generated on an assumed mode of operation. TVA further stated that for reactors with no actual operating experience, such as WBN Unit 2, design-basis fluence calculations are completed based on the assumption of operation with a conservative out/in (nonlow-leakage) fuel loading pattern for the entire licensed lifetime of the reactor. Furthermore, this design-basis vessel fluence at EOL, whether based on the assumed 80-percent capacity factor or a higher capacity factor, will be conservative compared with best-estimate vessel fluence analyses. TVA also stated that, therefore, the design-basis EOL fluence values are appropriately conservative and do not need to be adjusted to account for a higher capacity factor. The NRC staff finds the applicant's response acceptable because the conservatism built into the design-basis fluence calculation should result in a fluence that would bound any increases in neutron fluence associated with higher capacity factors. The staff also notes that WBN Unit 2 will have to revise its EOL neutron fluence projections as necessary to account for dosimetry data obtained through the RV materials surveillance program, which should ensure that the RT_{PTS} projection remains accurate. Therefore, the staff concludes that TVA's response to RAI 5.3.1-4 is acceptable.

The NRC staff performed an RT_{PTS} calculation for 40 EFPY and found that the RT_{PTS} values remain below the screening criteria. Table 5.3.1-2 summarizes the staff's RT_{PTS} evaluation.

Since the RT_{PTS} values at EOL for the WBN Unit 2 beltline materials are below the screening criteria of 10 CFR 50.61, the staff finds that TVA's WBN Unit 2 PTS evaluation is acceptable, pending resolution of Open Item 44 (Appendix HH).

Table 5.3.1-2
Staff RT_{PTS} Evaluation Summary for WBN 2

Material	$RT_{NDT(u)}$, °F	CF, °F	Fluence, 32 EFPY, n/cm^2 (E > 1.0 MeV) ×10^{19}	Fluence, 40 EFPY, n/cm^2 (E > 1.0 MeV) ×10^{19}	σ_Δ °F	σ_u °F	M °F	ΔRT_{NDT}, 32 EFPY °F	ΔRT_{NDT}, 40 EFPY °F	RT_{PTS}, 32 EFPY °F	RT_{PTS}, 40 EFPY °F
Intermediate Shell Forging 05	14	31	3.17[1]	3.96[2]	17	0	34	40.3	41.9	88.3	89.9
Lower Shell Forging 04	5	31	3.17[1]	3.96[2]	17	0	34	40.3	41.9	79.3	80.9
Intermediate-to-Lower Shell Circumferential Weld Seam W05	-50	68	3.17[1]	3.96[2]	28	0	56	88.4	91.8	94.4	97.8

1 Fluence Factor = 1.30

2 Fluence Factor = 1.35

<u>Upper Shelf Energy</u>

In Appendix B to the PTLR, TVA provided the projected USE values for the EOL, defined as 32 EFPY. USE values were provided for the three limiting beltline materials.

The limiting beltline materials are an intermediate shell forging, a lower shell forging, and an intermediate-to-lower shell circumferential weld seam. The applicant used Position 1.2 in RG 1.99, Revision 2, to project the USE values, since surveillance data are unavailable. For the initial (unirradiated) USE values, TVA stated that the C_v specimens were tested in the strong (tangential) direction. ASTM E185-82, which is incorporated by reference in Appendix H to 10 CFR Part 50, requires that such specimens be oriented normal to the major working direction of the material for forgings (transverse orientation). However, TVA reduced the measured values to 65 percent of their value in accordance with the guidance in the SRP BTP 5-3, Revision 3, "Fracture Toughness Requirements." The projected USE values are all greater than the requirement of 10 CFR Part 50, Appendix G, that the USE be greater than or equal to 50 foot-pounds (ft-lbs). The initial USE values are also greater than the requirement of 10 CFR Part 50, Appendix G, for unirradiated USE values of 75 ft-lbs, after being reduced to 65 percent of the value measured in the longitudinal direction.

In Reference 6, TVA provided revised initial (unirradiated) USE values for WBN Unit 2 in response to the staff's RAI related to TVA's initial response to GL 92-01. The USE values provided for intermediate shell forging 05 and intermediate-to-lower shell circumferential weld seam W05 in Reference 6 differ from those given in the PTLR. Specifically, in Reference 6, the values are from direct measurements of transversely oriented C_v specimens. The initial USE values given in the PTLR are lower than the values given in Reference 6, except for lower shell forging 04, for which the value is identical (105 ft-lbs). Although the USE values provided in the PTLR are different than USE values previously provided to the staff for the same materials, the staff concludes that the revised values are acceptable, because they are either lower or the same as the original values and are, therefore, conservative.

The NRC staff performed a confirmatory evaluation of TVA's projected USE values using TVA's copper content for the limiting materials and projected fluence, in conjunction with RG 1.99, Revision 2. TVA provided a 32-EFPY projected fluence at the vessel inner diameter of 3.17×10^{19} n/cm^2 (E > 1.0 MeV) and a copper content of all of the limiting materials of 0.05 weight-percent. Based on TVA's RV inner diameter fluence, the NRC staff calculated the 1/4t fluence using Equation 3 in RG 1.99, Revision 2. For the base materials, the staff conservatively used the line on Figure 2 of RG 1.99, Revision 2, for 0.10-percent copper since there is no line for 0.05 percent. TVA used the same line. For both the base and weld metals, the staff determined the predicted decrease in USE to be 23 percent, which was the same as TVA's determination. The staff's predicted USE values for all three limiting materials at 32 EFPY are the same as the values predicted by TVA. As stated in its response letter dated July 31, 2010, to RAI 5.3.1-4, TVA considers the use of 32 EFPY as the EOL fluence to be conservative, even though many nuclear plants are now operating at capacity factors in excess of 90 percent, because of the conservative assumptions made for the core design to ensure that the 32-EFPY fluence will bound the expected actual EOL fluence, even with higher capacity factors (see the section of this SSER on PTS for details). The NRC staff also performed the USE projections with the 1/4t fluence projected to 40 EFPY. The predicted decrease in USE is 24 percent, so the USE would still remain above 50 ft-lbs at 40 EFPY. Table 5.3.1-3 summarizes the staff's USE evaluation for WBN Unit 2.

Table 5.3.1-3 Staff USE Evaluation Summary for WBN Unit 2

Material	Unirradiated USE, ft-lbs	Projected USE Decrease, % (32 EFPY)[2]	Projected USE Decrease, %, (40 EFPY)[3]	Projected USE, 32 EFPY, ft-lbs	Projected USE, 40 EFPY, ft-lbs
Intermediate Shell Forging 05	(138) 90[1]	23	24	69	68
Lower Shell Forging 04	(162) 105[1]	23	24	81	80
Intermediate-to-Lower Shell Circumferential Weld Seam W05	127	23	24	98	96.5

1 Longitudinal value in parentheses; transverse value was used as basis for projection

2 32 EFPY fluence at 1/4t is 1.91×10^{19} n/cm^2 (E > 1.0 MeV)

3 40 EFPY fluence at 3/4t is 2.38×10^{19} n/cm^2 (E > 1.0 MeV)

The NRC staff concludes that the USE values for WBN Unit 2 are acceptable because the values will remain above 50 ft-lbs at EOL and therefore meet the requirements of Appendix G to 10 CFR Part 50.

Special Processes Used in Manufacturing and Fabrication

SRP Section 5.3.1, "Reactor Vessel Materials," states that:

> Information submitted by the applicant for any special process used in the manufacture of the product forms supplied and for their fabrication into the reactor vessel or any of its appurtenances is reviewed, and the capability of these processes to provide components with suitable mechanical and physical properties is assessed. The effects of such special processes on the stress-corrosion characteristics of the material, and any aspect of the process which could cause special requirements for nondestructive examinations, are reviewed.
>
> Information on special processes used for manufacture and fabrication of the reactor vessel and its appurtenances is reviewed to (1) identify each special process, (2) determine whether there are any Code restrictions on its use, (3) establish the adequacy of the process in providing components with suitable mechanical and physical properties, (4) establish the effects of such processes on the stress-corrosion characteristics of the material, and (5) identify whether special requirements for nondestructive examination are needed if the process is used. Since there are no specific Code requirements on the use of special processes, the suitability of a process is assessed on the basis of service

experience with similar parts fabricated by the process being reviewed.

A special process or method is one that deviates from or is not covered by the ASME Code. With respect to special processes used to fabricate the RV (e.g., welding), SRP Section 5.3.1 states that the requirements of GDC 1 and 30 and 10 CFR 50.55a regarding quality standards are met by compliance with the provisions of ASME Code, Section III, for fabrication of components when the appropriate ASME Code symbols are affixed and appropriate certifications are made by the manufacturer or installer.

With respect to special methods for NDE, SRP Section 5.3.1 further states that the requirements of GDC 1 and 30 and 10 CFR 50.55a regarding quality standards are met by compliance with the ASME Code, Section III, for fabrication nondestructive testing and that the acceptance criteria for examination of the RV and its appurtenances by NDE are those specified in ASME Code, Section III, Subarticle NB-5000.

In WBN Unit 2 FSAR Amendment 97, Section 5.4.4.2, "Penetrant Examinations," TVA changed the description of the liquid penetrant examinations of the core support block attachment welds. The description previously stated that the core support block attachment welds were inspected by dye penetrant after the first layer of weld metal and after each 2 inches of weld metal. TVA changed the description to state that the core support block attachment welds were inspected by dye penetrant after each one-half inch of weld metal.

The code of record for the WBN Unit 2 RV is ASME Code, 1971 Edition through Winter 1971 Addenda. ASME Code, Section III, Subarticle NB-4433, of the code of record requires that structural attachment welds be full-penetration welds, except for temporary attachments and minor supports. ASME Code, Section III, Subarticle NB-5260, states that structural attachment welds to pressure-retaining material shall be examined by either the magnetic particle or liquid penetrant method. The ASME Code does not provide any requirements for the increment of liquid penetrant examination, if performed, for structural attachment welds. However, inspecting the core support block attachment welds after each one-half inch of weld is consistent with the requirements of ASME Code, Section III, Subarticle NB-5245, for fillet welded and partial penetration welded joints. ASME Code, Section III, Subarticle NB-5245, requires such welds to be examined progressively using either the magnetic particle or liquid penetrant methods, with the increments of examination being the lesser of one-half of the maximum welded joint dimension measured parallel to the center line of the connection or one-half inch (13 millimeters).

Provided that the core support block attachment welds are full-penetration welds, all ASME Code requirements are met, and the specified NDE requirements do not conflict with the ASME Code requirements. Since it was not clear whether the core support blocks are full-penetration welds, and the basis for the NDE requirements are unclear, the staff requested additional information (RAI 5.3.1-6). In its response letter dated July 31, 2010, to RAI 5.3.1-6, TVA confirmed that the core support block attachment welds are full-penetration welds. TVA also explained the basis for the NDE requirements. TVA stated that the progressive liquid penetrant examination is performed as an alternative to ultrasonic examination (UT) because the configuration of the core support block attachment welds is not suitable for performing meaningful UT. TVA further stated that, in this particular respect, the situation is similar to a partial penetration weld, and progressive liquid penetrant examination is considered the acceptable alternative examination. The NRC staff concludes that, since the core support block attachment welds are full-penetration welds and the NDE requirements meet ASME Code requirements for examination of structural attachment welds, all ASME code requirements are met and the design of the welds is, therefore, acceptable. TVA's response to RAI 5.3.1-6 is acceptable to the staff.

Based on its review of the information provided by TVA in FSAR Amendment 97, as supplemented by letter dated July 31, 2010, the NRC staff concludes that the changes to the welding and NDE of the core support block attachment welds meet ASME Code, Section III requirements; therefore, TVA's changes to the special processes meet the requirements of GDC 1, GDC 30, and 10 CFR 50.55a.

Conclusions

Pending resolution of Open Item 44 (Appendix HH), the NRC staff concludes that the changes to the FSAR pertaining to the RV materials surveillance program are acceptable because the surveillance program meets the provisions of ASTM E185-82 and, therefore, meets the requirements of 10 CFR Part 50, Appendix H.

The staff concludes that the USE and RT_{PTS} values projected at EOL for WBN Unit 2 are acceptable because the values meet the criteria of Appendix G to 10 CFR Part 50 and 10 CFR 50.61, respectively.

The staff concludes that the changes to the special processes meet the requirements of GDC 1 and 30 and 10 CFR 50.55a because the welding and NDE of the core support block attachment welds meet the requirements of ASME Code, Section III.

5.3.1.6 References for Section 5.3.1

1. WCAP-17035-NP, "Watts Bar Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation," Revision 2, December 2009, ADAMS Accession No. ML100550651.

2. Shell, Ralph H., letter to U.S. Nuclear Regulatory Commission, Subject: "Browns Ferry Nuclear Plant (BFN), Sequoyah Nuclear Plant (SQN), and Watts Bar Nuclear Plant (WBN)—Response to Generic Letter 92-01 (Reactor Vessel Structural Integrity)," July 7, 1992, ADAMS Accession No. ML082380433.

3. Medford, Mark O., letter to U.S. Nuclear Regulatory Commission, Subject: "Response to NRC Generic Letter (GL) 92-01, Revision 1, Supplement 1, Reactor Vessel Structural Integrity (TAC Nos. M92649, M92650, M92651, M92730, M92731, M83525, and M83526)," August 17, 1995, ADAMS Accession No. ML082420707.

4. Carier, Patrick P., letter to U.S. Nuclear Regulatory Commission, Subject: "Response to NRC Generic Letter (GL) 92-01, Revision 1, Supplement 1, Reactor Vessel Structural Integrity—Browns Ferry (BFN), Watts Bar (WBN), and Sequoyah (SQN) Nuclear Plants (TAC Nos. M92649, M92650, M92651, M92730, M92731, M83525, and M83526)," November 7, 1995, ADAMS Accession No. ML082420460.

5. Mebdon, Frederick J., U.S. Nuclear Regulatory Commission, letter to Oliver D. Kingsley, Subject: "Closeout for Tennessee Valley Authority Response to Generic Letter 92-01, Revision 1, Supplement 1, for the Watts Bar Nuclear Plant, Units 1 and 2, the Sequoyah Nuclear Plant, Units 1 and 2, and the Browns Ferry Nuclear Plant, Units 1, 2, and 3

(TAC Nos. M92748, M92749, M92730, M92731, M92749, M92750, and M92751)," July 26, 1996, ADAMS Accession No. ML082401248, non-publicly available.

6. Nunn, Dwight E., letter to U.S. Nuclear Regulatory Commission, Subject: "Watts Bar Nuclear Plant (WBN) Units 1 and 2—Generic Letter 92-01, Revision 1—Reactor Vessel Structural Integrity and Low Upper Shelf Energy Evaluation (TAC Nos. M83525, M83526, and M85712)," June 13, 1994, ADAMS Accession No. ML073230667.

5.3.2 Pressure-Temperature Limits

Radiation embrittlement causes a reduction in the ductility of the RV beltline materials. The presence of elements such as copper, nickel, and phosphorus is controlled to limit reductions in ductility and fracture toughness in the steel that forms the RV. P-T limits, derived using linear-elastic fracture mechanics principles, provide margins of safety to prevent nonductile fracture during normal operation, heatup, cooldown, anticipated operational occurrences (AOOs), and system hydrostatic, preservice, and inservice leakage tests.

In SSER 16, issued September 1995, the NRC staff reviewed the P-T limits for WBN Unit 1 only and stated that the Unit 2 P-T limits would be reported as part of the Unit 2 technical specifications (TS), with the finding to be included in future SSERs. Therefore, in SSER 21, issued February 2009, the staff's review related to SRP Section 5.3.2, "Pressure-Temperature Limits," was identified as open.

In WBN Unit 2 FSAR Amendment 97, Section 5.4.2.3, TVA provided new information on the limiting RT_{NDT} values used to determine the operating P-T limits. FSAR Section 5.4.2.3 states, "[B]ased on copper and nickel content and initial RT_{NDT}, the intermediate forging is determined to be limiting for Unit 1. The limiting material for Unit 2 is the intermediate forging through 7 EFPY." This information was changed to be consistent with the PTLR, WCAP-17035-NP, Revision 2 (Ref. 1), submitted to the NRC via letter dated February 2, 2010 (Ref. 2), which provides the basis for the P-T limits for WBN Unit 2.

The revised TS for WBN Unit 2 also reference WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," as the methodology that must be used to develop P-T limits. WBN Unit 2 TS Bases B3.4.3, "RCS Pressure and Temperature (P/T) Limits," and TS 5.9.6, "Reactor Coolant System (RCS) Pressure And Temperature Limits Report (PTLR)," reference WCAP-14040-A, Revision 4. TS 3.4.3, "RCS Pressure and Temperature (P/T) Limits," contains the limiting condition for operation that the P-T limits be consistent with the PTLR.

The NRC staff used SRP Section 5.3.2, Revision 1, dated July 31, 1981, as guidance for reviewing the information related to P-T limits. Section 5.3.1 of this report discusses the staff's evaluation of PTS, USE, and the RV materials surveillance program.

The NRC staff's review of TVA's P-T limits for WBN Unit 2 is based on the following regulatory requirements:

- 10 CFR 50.55a, as it relates to quality standards for the design, fabrication, erection, and testing of SSCs important to safety

- 10 CFR 50.60, as it relates to compliance with the requirements of Appendix G to 10 CFR Part 50

- GDC 1, as it relates to quality standards for design, fabrication, erection, and testing
- GDC 14, as it relates to ensuring an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture of the RCPB
- GDC 31, as it relates to ensuring that the RCPB will behave in a nonbrittle manner and that the probability of rapidly propagating fracture is minimized
- GDC 32, as it relates to the RV materials surveillance program
- Appendix G to 10 CFR Part 50, as it relates to material testing and fracture toughness

Additionally, GL 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," dated January 31, 1996 (Ref. 3), establishes the information that must be included in (1) an acceptable PTLR methodology (which will be used to develop the PTLR) and (2) the PTLR itself.

Verification of Material Properties—RT_{NDT} and Adjusted Reference Temperature

Section 5.3.1 of this SSER, under the subheading "Pressurized Thermal Shock," discusses the NRC staff's evaluation of the initial (unirradiated) RT_{NDT} values for the beltline materials.

The P-T limits for WBN Unit 2 are based on the irradiated RV material properties at 7 EFPY, which are provided in Reference 1. At the projected neutron fluence for 7 EFPY, Table 4-2 of Reference 1 provides the adjusted reference temperature (ART) at the 1/4t location as 61 degrees F, and Table 4-3 of Reference 1 provides the ART for the 3/4t location as 45 degrees F for intermediate shell forging 05. The ART values were determined using the methodology of RG 1.99, Revision 2.

Using TVA's projected neutron fluence at the vessel inner surface, and the copper and nickel content of the beltline materials, the NRC staff reviewed TVA's calculations of the ART for each beltline material using the methodology of RG 1.99, Revision 2. The staff's calculation of ART for each beltline material yielded the same results as TVA's calculation and confirmed that intermediate shell forging 05 is the limiting material. The staff concludes that TVA's projected values of ART for the beltline materials are acceptable because they were confirmed by the staff's independent calculation.

Pressure-Temperature Limits

The regulations in 10 CFR 50.60 and Appendix G to 10 CFR Part 50 describe the conditions that require P-T limits and provide the general basis for these limits. Appendix G to 10 CFR Part 50 specifically requires that P-T limits must be at least as conservative as limits obtained by following Appendix G to Section XI of the ASME Code during heatup, cooldown, and test conditions. Appendix G to 10 CFR Part 50 also requires additional safety margins when the reactor core is critical.

Reference 1 describes TVA's methodology for the development of the P-T curves for WBN Unit 2. In the PTLR, TVA stated that it used the NRC-approved methodology in Revision 4 of WCAP-14040-A to develop the P-T limits. Using the methodology of WCAP-14040-A, the allowable pressures for a given dynamic crack initiation or arrest toughness, K_{Ic}, and metal temperature are determined consistent with ASME Code, Section XI, Appendix G. However, the thermal stress intensity, K_{It}, is determined using the ORIGEN computer code, which results in less conservative thermal stress intensity values than the method provided in ASME Code, Section XI, Appendix G. However, the NRC staff concluded that the thermal stress intensity value used by TVA was acceptable, because the methodology for determining the K_{It} factors and metal temperatures is consistent with methods allowed by the ASME Code. The staff's evaluation is documented below in the portion of this section of the SSER entitled "Heatup and Cooldown Curves."

Minimum Temperature Requirements

The NRC staff reviewed the minimum temperature requirements provided in the PTLR against the criteria of paragraph IV.2.a of Appendix G to 10 CFR Part 50 for the conditions identified below. Table 5.3.2-1 provides the requirements, the minimum requirement, and the actual temperatures for WBN Unit 2.

1) Hydrostatic Pressure and Leak Tests (Core not Critical)

   a) For hydrostatic pressure and leak tests with fuel in the vessel (inservice), Appendix G to 10 CFR Part 50 requires that the minimum temperature at pressures less than or equal to 20 percent of the preservice system hydrostatic test (PSHT) pressure is the highest reference temperature of the material in the closure flange region of the vessel that is highly stressed by bolt preload.

      A minimum temperature of 60 degrees F is shown by a vertical line on the noncritical P-T curve on Figure 5-1 of the PTLR. For WBN Unit 2, the highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload is -22 degrees F for the WBN Unit 2 vessel. Therefore, the minimum temperature requirement of 60 degrees F is significantly greater than the minimum allowed by 10 CFR Part 50, Appendix G, and is acceptable.

   b) For hydrostatic pressure and leak tests with fuel in the vessel (inservice), Appendix G to 10 CFR Part 50 requires that the vessel pressure may not exceed 20 percent of the PSHT pressure until the vessel temperature exceeds by 90 degrees F the highest reference temperature of the material in the closure flange region of the vessel that is highly stressed by bolt preload.

      For WBN Unit 2, the highest reference temperature of the material in the closure flange region of the vessel is -22 degrees F, so the vessel pressure could exceed 20 percent of the PSHT pressure at 68 degrees F (-22 °F + 90 °F) during hydrostatic pressure and leak tests. PTLR Figure 5-1 shows that the leak test limit curve does not begin pressure ascension until 100 degrees F or greater is reached, so the Appendix G requirement is met.

   c) For hydrostatic pressure and leak tests, Appendix G to 10 CFR Part 50 also requires that the minimum temperature for the PSHT with no fuel in the vessel is the highest reference temperature of the vessel plus 60 degrees F.

      For WBN Unit 2, the highest reference temperature for the vessel is 61 degrees F, so the minimum PSHT temperature should be at least

121 degrees F. Section 5 of the PTLR gives the minimum hydrotest temperature as 122 degrees F, so the Appendix G requirement is met.

2) Normal Operation (Including Heatup and Cooldown) Including AOOs

a) For normal operation (including heatups and cooldowns) including AOOs, Appendix G to 10 CFR Part 50 requires that, when the core is not critical, the minimum temperature at pressures less than or equal to 20 percent of the PSHT pressure is the highest reference temperature of the material in the closure flange region of the vessel that is highly stressed by bolt preload.

A minimum temperature of 60 degrees F is shown by a vertical line on the core noncritical P-T curve on Figure 5-1 of the PTLR. For WBN 2, the highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload is -22 degrees F for the WBN 2 vessel. Therefore, the minimum temperature requirement of 60 degrees F is significantly greater than the minimum allowed by 10 CFR Part 50, Appendix G, and is therefore acceptable.

b) For normal operation (including heatups and cooldowns) including AOOs, Appendix G to 10 CFR Part 50 also requires that the vessel pressure may not exceed 20 percent of the PSHT until the vessel temperature exceeds by 120 degrees F the highest reference temperature of the material in the closure flange region of the vessel that is highly stressed by bolt preload.

In accordance with Section 3.3 of the PTLR, the PSHT pressure for the WBN 2 vessel is 3,107 pounds per square inch gauge (psig), so 20 percent of that value is 621 psig. Since the minimum RT_{NDT} of the WBN 2 vessel closure flange is -22 degrees F, the pressure must not exceed 621 psig until the metal temperature exceeds 98 degrees F. Since Tables 5-1 and 5-2 provide the P-T limits in increments of 5 degrees F, the pressure is shown as 621 psig at all temperatures from 60 degrees F to 100 degrees F. PTLR Figure 5-1 also shows this and indicates that pressure can exceed 621 psig at 100 degrees F and above. Therefore, this requirement is met.

c) For normal operation, Appendix G to 10 CFR Part 50, requires that, when the core is critical at pressures less than or equal to 20 percent of the PSHT pressure, the minimum temperature is the larger of the minimum permissible temperature for the inservice hydrotest or the highest reference temperature of the material in the closure flange region of the vessel that is highly stressed by bolt preload plus 40 degrees F.

For WBN Unit 2, the minimum permissible temperature for the inservice hydrotest is 122 degrees F and the highest reference temperature in the closure flange region plus 40 degrees F is 18 degrees F (-22 °F + 40 °F). Therefore, for normal operation when the core is critical with pressures less than or equal to 20 percent of the preservice system hydrostatic test pressure, the minimum temperature should be 122 degrees F. This limit is shown by a vertical line on the criticality limit curve on Figure 5-1 of the PTLR (which is labeled "Criticality

Limit based on inservice hydrostatic test temperature (122°F) for the service period up to 7 EFPY"). Therefore, this requirement of Appendix G is met.

d) For normal operation, Appendix G to 10 CFR Part 50, requires that, when the core is critical at temperatures greater than 20 percent of the preservice system hydrostatic test pressure, the minimum temperature is the larger of the minimum permissible temperature for the inservice hydrotest or the highest reference temperature of the material in the closure flange region of the vessel that is highly stressed by bolt preload plus 160 degrees F.

For WBN Unit 2, this requirement would be governed by the highest reference temperature of the material in the closure flange region of the vessel that is highly stressed by bolt preload plus 160 degrees F, which is 138 degrees F (-22 °F + 160 °F). This limit is shown by an unlabeled, vertical line on the criticality limit curve on Figure 5-1 of the PTLR. Therefore, this requirement of Appendix G is met.

Table 5.3.2-1
Summary of WBN 2 Compliance with Minimum Temperature Requirements*

Operating condition	Vessel pressure[1]	Requirements for pressure-temperature limits	Minimum temperature requirements	WBN 2 Min. Req.	WBN 2 Actual
1. Hydrostatic pressure and leak tests (core is not critical):					
1.a Fuel in the vessel	≤ 20%	ASME Code, Section XI, Appendix G Limits	([2])	-22 °F	60 °F
1.b Fuel in the vessel	> 20%	ASME Code, Section XI, Appendix G Limits	([2]) +90 °F([6])	68 °F	100 °F
1.c No fuel in the vessel (Pre-service Hydrotest Only)	ALL	(Not Applicable)	([3]) +60 °F	121 ° F	122 °F*
2. Normal operation (including heatup and cooldown), including AOOs:					
2.a Core not critical	≤ 20%	ASME Code, Section XI, Appendix G Limits	([2])	-22 °F	60 °F
2.b Core not critical	> 20%	ASME Code, Section XI, Appendix G Limits	([2]) + 120 °F	98 °F	100 °F
2.c Core critical	≤ 20%	ASME Code, Section XI, Appendix G Limits + 40 °F	Larger of [([4])] or [([2]) + 40 °F]	122 °F	122 °F
2.d Core critical	> 20%	ASME Code, Section XI, Appendix G Limits + 40 °F	Larger of [([4])] or [([2])+160 °F]	138 °F	138 °F

1 Percent of the preservice system hydrostatic test pressure.

2 The highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload.

3 The highest reference temperature of the vessel.

4 The minimum permissible temperature for the inservice system hydrostatic pressure test.

6 Lower temperatures are permissible if they can be justified by showing that the margins of safety of the controlling region are equivalent to those required for the beltline when it is controlling.

* Inservice and preservice hydrotest temperatures were not explicitly given.

Based on its review, the NRC staff concludes that the minimum temperature requirements associated with the P-T limits for WBN Unit 2 meet the criteria of 10 CFR Part 50, Appendix G, and are, therefore, acceptable.

Heatup and Cooldown Curves

During its review of TVA's P-T limits, the NRC staff performed confirmatory calculations of the P-T limits using the guidance of SRP Section 5.3.2. The staff calculated the allowable pressures for temperatures of 100 degrees F through 170 degrees F (at 10-degree intervals) for heatup and 100 degrees F through 140 degrees F for cooldown (at 10-degree intervals) using the methods of ASME Code, Section XI, Appendix G.

- The thermal stress intensities (K_{It}) were calculated using the simple equations from ASME Code, Section XI, Appendix G, G-2214.3, which provides a constant K_{It} through the wall thickness:

  $$K_{It} = 0.953\times10^{-3} \times CR \times t^{2.5}$$

  $$K_{It} = 0.753\times10^{-3} \times HU \times t^{2.5}$$

  Where:

  HU is the heatup rate in °F/h

  t = wall thickness in inches

- The staff used ASME Code, Section XI, Appendix G, Figure G-2214-2, to determine the temperature difference from the coolant at a specific wall depth.

- The metal temperatures for the 1/4t and 3/4t locations corresponding to a given coolant temperature were determined by adding or subtracting, as appropriate, the temperature difference for the specified wall depth from Figure G-2214-2.

The NRC staff performed its confirmatory calculations for heatup curves of 100 degrees F per hour and 60 degrees F per hour and cooldown curves of 100 degrees F per hour and 60 degrees F per hour. For the cooldown curves, the staff's calculated allowable pressures are higher for every temperature from 170 degrees F through 100 degrees F for a cooldown of 100° degrees F per hour and 140 degrees F through 100 degrees F for a cooldown of 60 degrees F per hour. TVA only provided values up to 140 degrees F because, at 140 degrees F, TVA's calculated allowable pressure is higher than the normal operating pressure of 2,250 degrees F. For the heatup curve of 100 degrees F per hour, the staff's calculated allowable pressures are significantly lower at 100 degrees F, but converge with the WCAP curve values by 170 degrees F. For the heatup curve of 60 degrees F, the value of the staff's calculated pressure is slightly lower at 100 degrees F, but slightly exceeds TVA's value at 150 degrees F.

The thermal stress intensities (K_{It}) calculated by TVA at the lower temperatures are lower, and the metal temperatures are higher, compared to the values calculated using the equations from Appendix G of the ASME Code, Section XI. TVA provided the K_{It} and metal temperature values at the 1/4t and 3/4t locations for a heatup of 100 degrees F in Appendix A to Reference 1. The ASME K_{It} equation calculates K_{It} only as a function of heatup rate, while TVA's K_{It} values also vary with the absolute temperature, increasing as temperature increases. At lower absolute temperatures, the higher metal temperatures used by TVA result in higher K_{Ic} values for a given coolant temperature, which, combined with the lower thermal stresses calculated by TVA, results in higher allowable pressures for a given coolant temperature.

When TVA's K_{It} and metal temperature values were used in the ASME Code, Section XI, Appendix G, equations for allowable pressure, the values obtained are identical to TVA's allowable pressures. However, because the method of determining the K_{It} factors and metal temperatures used by TVA was unclear, the NRC staff asked TVA (RAI 5.3.2-1) to provide additional information on how it determined the K_{It} and metal temperatures. The staff's RAI addressed which set of equations from WCAP-14040-A, Revision 4, were used to determine the K_{It} values, how the constants in the equations were determined, how the thermal stress distribution was determined, how the crack tip metal temperature was determined, and the boundary conditions.

Based on the information provided by TVA in its response letter dated July 31, 2010, TVA's methodology of determining K_{It} can be summarized as follows:

- The temperature distribution through the wall was calculated as a function of time and position for both heatup and cooldown. The one-dimensional transient heat conduction equation of Section 2.6.1 of WCAP-14040-A, Revision 4, was used to determine the through-wall temperature distribution as a function of time during the heatup or cooldown.

- With respect to boundary conditions, the vessel inner surface is assumed to have a very high convection coefficient (7,000 British thermal units per hour per square foot per degree F (BTU/(h*ft^{2}*°F). The very high convection coefficient essentially allows unimpeded transfer of heat from the coolant to the inside surface of the vessel. The outside surface is assumed to be adiabatic. Therefore, it is perfectly insulated so that heat does not transfer from the outside surface to the air.

- The initial position and time-dependent thermal hoop stress profile was determined using the equation given by Timoshenko (Reference 14 of WCAP-14040-A, Revision 4), which is Equation 2.6.1-4 in WCAP-14040-A, Revision 4.

- A polynomial fitting method was used to determine the values for C_0, C_1, C_2, and C_3, as defined in Equations 7 and 8 of WCAP-17035-NP, Revision 2:

- Equation 7 of the PTLR is applicable for an inside surface defect during cooldown:

  $K_{It} = (1.00359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3)^* \sqrt{\pi a}$

- Equation 8 of the PTLR is applicable for an outside surface defect during heatup:

  $K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3)^* \sqrt{\pi a}$

- If this method is used, the linearization technique presented in Appendix A to Section XI of the ASME Code does not need to be used to determine the components of the membrane and bending stress, which would be necessary if the equations of WCAP-14040-A, Revision 4, Sections 2.6.3 and 2.6.4, were used. Section 2.7 of WCAP-14040-A Revision 4, discusses PTLR Equations 7 and 8 as an alternative to the methods presented in Sections 2.6.2 through 2.6.4 of WCAP-14040-A, Revision 4. These equations were incorporated into the ASME Code, Section XI, Appendix G, starting with the 1996 Edition.

- The allowable pressure is then determined using the method of paragraph G-2215 of Appendix G to ASME Code, Section XI. The NRC staff verified the pressure by duplicating TVA's allowable pressure results when TVA's K_{It} values were input to the paragraph G-2215 equations:

    - For membrane tension, the corresponding K_I for the postulated defect is $K_{Im} = M_m \ (pR_i/t)$

      This is Equation 3 of the PTLR and is the same as the equation in ASME Code, Section XI, Appendix G, paragraph G-2214.1.

    - The governing equation for the heatup and cooldown analysis is $C^* K_{Im} + K_{It} < K_{Ic}$ where:

      K_{Im} = stress intensity factor caused by membrane (pressure) stress
      K_{It} = stress intensity factor caused by the thermal gradients
      K_{Ic} = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}

      C = 2.0 for Level A and Level B service limits

      C = 1.5 for hydrostatic and leak test conditions during which the reactor core is not critical

      This step is identical to ASME paragraph G-2215, Equation 1.

    - Equations 3, 7, and 8 of the PTLR were implemented in the OPERLIM computer code.

Since Equations 7 and 8 of the PTLR are included in Section XI, Appendix G, in the 2004 Edition of the ASME Code, which 10 CFR 50.55a incorporates by reference, the NRC staff finds the use of these equations to determine K_{It} acceptable. The simplified equations for K_{It} used by the staff in its confirmatory calculation will result in more conservative K_{It} values and thus more conservative allowable pressure values. The staff's calculated allowable pressures are lower than TVA's allowable pressures primarily in the lower temperature portion of the heatup curve. This is to be expected since the simple equations used by the staff for calculating K_{It} in its confirmatory calculations determine K_{It} based on a generic thermal stress distribution, while TVA modeled the specific thermal stress distribution for the WBN Unit 2 vessel, including the specific boundary conditions. Additionally, the simplified equations determine K_{It} only as a function of heatup rate and RV wall thickness and therefore do not take absolute temperature into account. Thermal stresses are likely to be higher at higher temperatures. Considering the above, the differences in the staff's and TVA's allowable pressures are reasonable. TVA's response to RAI 5.3.2-1, in combination with the staff's confirmatory calculations, confirmed that the methodology used to determine the WBN Unit 2 P-T curves was consistent with the staff-approved methodology of WCAP-14040-A, Revision 4, and ASME Code, Section XI, Appendix G, and is, therefore, acceptable. TVA's response to RAI 5.3.2-1 is acceptable.

The NRC staff concludes that TVA's P-T limits for heatup and cooldown curves are acceptable because (1) the limits have been determined in accordance with an NRC-approved methodology (WCAP-14040-A, Revision 4), and (2) the methodology for determining the K_{It} factors and metal temperatures is consistent with methods allowed by the ASME Code. Therefore, the resulting allowable pressures meet the requirements of ASME Code, Section XI, Appendix G.

Hydrostatic Test Pressure-Temperature Limits

The NRC staff performed a confirmatory calculation of the WBN Unit 2 hydrostatic test P-T limits in the PTLR. The PTLR provides a minimum temperature for the inservice hydrostatic test of 122 degrees F and an inservice hydrostatic test pressure of 2,485 psig. The criterion for the hydrostatic test pressure-temperature relationship is as follows:

$$1.5K_{Im} < K_{Ic}$$

$$K_{1c} = 33.2 + 20.734\exp[0.02(T-RT_{NDT})$$

Where:

K_{Im} = the membrane stress from pressure

The staff determined the K_{Im} corresponding to a pressure of 2,485 psig and then solved the K_{Ic} equation for the temperature at which $1.5K_{Im} = K_{Ic}$. The result was 121.6 degrees F; therefore, the minimum hydrostatic test temperature is acceptable.

The staff also calculated the allowable pressures corresponding to the allowable K_{Im} for the hydrostatic test in accordance with the above equations. The allowable pressures are equal to or higher than those calculated by TVA. TVA provided temperatures and pressures for two points on the hydrostatic test curve. The NRC staff concludes that TVA's hydrostatic test P-T limits are acceptable because they are more conservative than the ASME Code, Section XI, Appendix G, criteria. Therefore, these limits meet the requirements of 10 CFR Part 50, Appendix G.

Low-Temperature Overpressure Protection System Enable Temperature

The NRC staff reviewed the low-temperature overpressure protection (LTOP) system enable temperature to verify TVA's compliance with GDC 15, "Reactor Coolant System Design," and 10 CFR Part 50, Appendix G. In its review, the staff used the guidance of SRP BTP Reactor Systems Branch (RSB) 5-2, Revision 1, "Overpressurization Protection Of Pressurized-Water Reactors While Operating at Low Temperatures," issued November 1988. BTP RSB 5-2 recommends that the LTOP system be operable during startup and shutdown conditions below the enable temperature, defined as the water temperature corresponding to a metal temperature of at least RT_{NDT} + 90 degrees F at the beltline location (1/4t or 3/4t) that is controlling in the 10 CFR Part 50, Appendix G, limit calculations.

For WBN Unit 2, the 3/4t location is controlling during heatup, but the 1/4t location is controlling during cooldown. The 1/4t location has a higher RT_{NDT} at 61 degrees F (versus 45 degrees F at 3/4t). RT_{NDT} plus 90 degrees F would therefore be 151 degrees F. Using the ASME Code, Section XI, Appendix G, method of calculating the temperature differential from the coolant to the 1/4t location, the corresponding coolant temperature during cooldown is 131 degrees F. During heatup, when the 3/4t location is controlling, RT_{NDT} plus 90 degrees F is 135 degrees F. The corresponding coolant temperature would be 155.3 degrees F. Even if the higher RT_{NDT} of 61 degrees F is assumed for the 3/4t location during heatup, the corresponding coolant temperature would be 181.3 degrees F. In its letter dated July 31, 2010, in response to RAI 5.3.2-2, TVA provided the arming temperature for the cold overpressure mitigation system (COMS), which is the same as the LTOP enable temperature, as less than or equal to 225 degrees F. This temperature is conservative because it is higher than the minimum LTOP enable temperature calculated by the staff. Therefore, TVA's LTOP enable temperature is acceptable. TVA stated in its response to RAI 5.3.2-2 that it would revise the PTLR to incorporate the COMS arming temperature. This is Open Item 45 (Appendix HH).

Generic Letter 96-03 Criteria

GL 96-03 contains seven criteria that must be met for P-T limits to be relocated from the TS to a PTLR, as discussed below:

1) The methodology shall describe how the neutron fluence is calculated (reference new RG when issued).

   The PTLR references WCAP-14040-A, Revision 4, which has been approved by the staff (ADAMS Accession No. ML050120209). WCAP-14040-A, Revision 4, Section 2.2, describes the fluence methodology.

   The NRC safety evaluation, which is included for information in WCAP-14040-A, Revision 4, concluded that the fluence methodology is acceptable because it meets the requirements of RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," issued March 2001.

   Therefore, this criterion is met.

2) The RV material surveillance program shall comply with Appendix H to 10 CFR Part 50. The RV material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.

   The RV material surveillance program meets the requirements of Appendix H to 10 CFR Part 50, pending resolution of Open Item 44 (see Section 5.3.1 of this SSER).

3) LTOP system lift-setting limits for the PORVs developed using NRC-approved methodologies may be included in the PTLR.

   The LTOP lift settings were not included in the PTLR, but were provided in TVA's response to RAI 5.3.2-2 in its letter dated July 31, 2010. TVA stated in its RAI response that it would revise the PTLR to incorporate the LTOP lift settings into the PTLR. This is Open Item 46 (Appendix HH).

4) The ART for each reactor beltline material shall be calculated, accounting for radiation embrittlement, in accordance with RG 1.99, Revision 2.

   TVA calculated the ART values for each beltline material in accordance with RG 1.99, Revision 2, as described in Section 4 of the PTLR and verified by the NRC staff. Therefore, this criterion is met.

5) The limiting ART shall be incorporated into the calculation of the P-T limit curves in accordance with SRP Section 5.3.2.

   TVA used the limiting ART values to generate the P-T curves, as stated in PTLR Section 1.0 and given on Figures 5-1 and 5-2. The staff's confirmatory calculation verified the limiting ART values used.

6) The minimum temperature requirements of Appendix G to 10 CFR Part 50 shall be incorporated into the P-T limit curves.

   The NRC staff verified that the minimum temperature requirements of 10 CFR Part 50, Appendix G, are incorporated into the P-T curves, as discussed in the section above entitled "Minimum Temperature Requirements."

7) Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature (RT_{NDT}) to the predicted increase in RT_{NDT}, where the predicted increase in RT_{NDT} is based on the mean shift in RT_{NDT} plus the two standard deviation value (2σ) specified in RG 1.99, Revision 2. If the measured value exceeds the predicted value (increase $RT_{NDT} + 2\sigma$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.

   There are no surveillance results yet, so this criterion is not applicable.

Pending resolution of Open Item 44 (Appendix HH), TVA has met the criteria of GL 96-03 for relocation of the P-T limits from the TSs to a PTLR.

Conclusions

The NRC staff concludes, pending resolution of Open Items 44, 45, and 46, that the P-T limits imposed on the RCS for operating and testing conditions to ensure adequate safety margins against nonductile or rapidly propagating failure conform to the fracture toughness criteria of Appendix G to 10 CFR Part 50. The use of operating limits, as determined by the criteria defined in Section 5.3.2 of the SRP, provides reasonable assurance that nonductile or rapidly propagating failure will not occur. This is an acceptable basis for satisfying the requirements of 10 CFR 50.55a; Appendix G to 10 CFR Part 50; and GDC 1, 14, 31, and 32. Therefore, WBN Unit 2 FSAR Section 5.3 is acceptable.

References for Section 5.3.2

1. WCAP-17035-NP, Revision 2, "Watts Bar Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation," December 2009, ADAMS Accession No. ML100550651.

2. Bajestani, Masoud, letter to the U.S. Nuclear Regulatory Commission, Subject: "Watts Bar Nuclear Plant (WBN)—Unit 2—Developmental Revision B of the Technical Specifications (TS), TS Bases, Technical Requirements Manual (TRM), TRM Bases; and Pressure and Temperature Limits Report (PTLR)," February 2, 2010, ADAMS Accession No. ML100550326.

3. U.S. Nuclear Regulatory Commission, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," Generic Letter 96-03, January 31, 1996, ADAMS Accession No. ML031110004.

5.3.3 Reactor Vessel Integrity

The NRC staff reviewed the factors that affect the integrity of the RV in the areas of design, materials of construction, fabrication methods, and operating conditions. Although most areas are reviewed separately in accordance with other SRP sections, RV integrity is of such importance that the staff conducted a special summary review of all factors relating to RV integrity.

In Section 5.3.3 of the SER, the NRC staff concluded that TVA had met all applicable regulatory requirements except paragraphs III.B.4, IV.A.1, and IV.B of Appendix G to 10 CFR Part 50. Specific exemptions to these requirements were granted in Section 5.3.1 of the SER. As discussed in Section 5.3.1 of this SSER, the staff concluded in SSER 14 that all of the specific exemptions were unnecessary because of changes made to 10 CFR Part 50, Appendix G, subsequent to the SER.

In WBN Unit 2 FSAR Amendment 97, TVA made the following changes:

- TVA made changes to certain RV material specifications listed in FSAR Table 5.8-2.
- TVA added a description of the process for zinc addition to FSAR Section 5.2.3.4.
- TVA made changes to the information on the RV materials surveillance program, including the surveillance capsule withdrawal schedule, capsule lead factors, and description of the specimen orientation.
- TVA revised information on the PTS reference temperature (RT_{PTS}) and provided limiting RV beltline material (see FSAR Section 5.2.4.3). Additionally, TVA submitted a PTLR (ADAMS Accession No. ML100550326) in conjunction with FSAR Amendment 97 that provides the basis for the P-T operating limits for WBN Unit 2. The PTLR includes revised P-T operating limits, an EOL USE evaluation, and EOL RT_{PTS} evaluation.
- TVA modified the description of the liquid penetrant examinations of the core support block attachment welds in FSAR Section 5.4.4.2, "Penetrant Examinations."

The NRC staff reviewed the following factors affecting RV integrity: design, materials of construction, fabrication methods, and operating conditions.

TVA's changes to the FSAR in the area of RV design, fabrication methods for the RV, and shipment and installation were editorial in nature and were therefore acceptable to the NRC staff.

With respect to fracture toughness, SRP Section 5.3.3 cites Appendix G to 10 CFR Part 50 as the basis for acceptable fracture toughness and Appendix H to 10 CFR Part 50 as the basis for acceptance of the RV materials surveillance program. The SRP recommends that the predicted toughness parameters (ART and USE) be at least as conservative as those predicted by RG 1.99, Revision 2.

In FSAR Amendment 97, TVA made changes in the area of fracture toughness. Specifically, FSAR Amendment 97 included a revised evaluation of the USE and RT_{PTS} at EOL. Section 5.3.1 of this SSER includes the NRC staff's evaluation of the USE and RT_{PTS} values. The staff notes that TVA used RG 1.99, Revision 2, to determine the ART, RT_{PTS}, and USE values for the limiting RV beltline materials. The staff found that the fracture toughness of RV materials meets the requirements of 10 CFR Part 50, Appendix G. The staff also found, pending resolution of Open Item 44, that the RV materials surveillance program meets the requirements of 10 CFR Part 50, Appendix H, as detailed in Section 5.3.1 of this SSER.

The NRC staff found that, as a consequence of the relatively low copper and nickel content of the WBN 2 RV beltline materials, the projected fracture toughness of these materials at EOL, as measured by the RT_{PTS}, USE, and ART values, is well within the acceptable regulatory bounds for these parameters (i.e., far below the PTS screening criteria and well above the minimum required irradiated USE).

Section 5.2.3 of this SSER provides the NRC staff's detailed evaluation of the changes made by TVA to material specifications in FSAR Amendment 97. SRP Section 5.3.3 states that, "Although many materials are acceptable for reactor vessels according to Section III of the Code, the special considerations relating to fracture toughness and radiation effects effectively limit the basic materials that are currently acceptable for most parts of reactor vessels to SA 533 Gr. B C1 1, SA 508 C1 2, and SA 508 C1 3." This requirement specifically relates to the ferritic materials in the RV beltline. The allowable materials for the head plates were expanded to include SA 533 Gr B, Class 1, Gr A, B, or C, Class 1 or 2 (formerly only SA 533 Grade B, Class 1 was allowed), and SA 508 Class 3 was added for shell, flange, and nozzle forgings (formerly only SA 508, Class 2 was permitted). These changes are consistent with the guidance of SRP Section 5.3.3. The NRC staff concluded that TVA's material specification changes were acceptable because the new materials were either ASME Code approved or the materials met approved ASME Code cases.

In FSAR Amendment 97, TVA changed the description of the liquid penetrant examinations of the core support block attachment welds in FSAR Section 5.4.4.2. In Section 5.3.1 of this SSER, the NRC staff concluded that the examination requirements for the core support block attachment welds are consistent with ASME Code, Section III, and are, therefore, acceptable.

Section 5.3.2 of this SSER provides the NRC staff's detailed evaluation of the P-T limits for the WBN Unit 2 RV. TVA relocated the WBN Unit 2 P-T limits from the TS to the PTLR. The NRC staff had previously approved the methodology described in the PTLR in a topical report, as detailed in Section 5.3.2 of this SSER. Considering some refinements in the topical report methodology compared to the methodology of ASME Code, Section XI, Appendix G, the NRC staff concluded that the P-T limits developed for WBN Unit 2 are equally conservative to limits that would be developed using the methods of ASME Code, Section XI, Appendix G. Therefore, the staff concludes that the P-T limits proposed by TVA meet the requirements of 10 CFR Part 50, Appendix G.

The NRC staff concluded that, pending resolution of Open Items 45 and 46 (Appendix HH), TVA met the requirements for protection against low temperature overpressurization events for WBN Unit 2.

The NRC staff reviewed the adequacy of the RV materials surveillance program using the guidance of SRP Section 5.3.1. The staff concluded that the RV materials surveillance program meets the requirements of Appendix H to 10 CFR Part 50. Section 5.3.1 of this SSER includes details of the staff's evaluation.

The NRC staff reviewed the information provided by TVA in the FSAR regarding the inservice inspections to be performed on the RV and concluded that the acceptance criteria for accessibility and inspection plan details are consistent with ASME Code, Section XI. The staff also concluded that the inservice inspection program and its implementation conform to the requirements of 10 CFR 50.55a(g).

In summary, the NRC staff concludes that there are no special considerations that make it necessary to consider potential RV failure for WBN Unit 2 because the design, materials, fabrication, inspection, and quality assurance requirements for the plant will continue to conform to applicable NRC regulations and RG, as well as to the provisions of ASME Code, Section III. The stringent fracture toughness requirements of the regulations and ASME Code, Section III, will be met, including requirements for surveillance of vessel material properties throughout service life, in accordance with Appendix H to 10 CFR Part 50. TVA will also establish operating limitations on temperature and pressure for WBN Unit 2 in accordance with ASME Code, Section III, Appendix G, "Protection Against Nonductile Failure," and 10 CFR Part 50, Appendix G.

Subject to resolution of Open Items 44, 45, and 46 (Appendix HH), the NRC staff concludes that integrity of the WBN Unit 2 RV is assured for the following reasons:

(1) The vessel will be designed and fabricated to the high standards of quality required by the ASME Code and any pertinent Code cases.

(2) The vessel will be made from materials of controlled and demonstrated high quality.

(3) The vessel will be subjected to extensive preservice inspection and testing to provide assurance that the RV will not fail because of material or fabrication deficiencies.

(4) The vessel will be operated under conditions and procedures and with protective devices that provide assurance that the RV design conditions will not be exceeded during normal reactor operation, maintenance, testing, and anticipated transients.

(5) The vessel will be subjected to periodic inspection to demonstrate that the high initial quality of the RV has not deteriorated significantly under service conditions.

(6) The vessel may be annealed to restore the material toughness properties if this becomes necessary.

(7) The vessel will be subjected to surveillance to account for neutron irradiation damage so that the operating limitations may be adjusted.

5.4 Component and Subsystem Design

5.4.1 Reactor Coolant Pumps

5.4.1.1 Pump Flywheel Integrity

The NRC staff reviewed the information provided by TVA in WBN Unit 2 FSAR Amendment 97, Sections 5.2.6 and 5.5, and concluded that there are no substantive changes to the information on reactor coolant pump flywheel integrity. Therefore, based on the prior staff evaluation documented in the SER, the staff concludes that the measures taken by the TVA to ensure that the integrity of the reactor coolant pump flywheel meets the relevant requirements of GDC 4 are acceptable.

5.4.2 Steam Generators

5.4.2.1 Steam Generator Materials

The NRC staff reviewed the steam generator materials, including material specifications, and the compatibility of the steam generator materials with the primary and secondary coolant. The staff also reviewed the design of the steam generators, particularly with regard to minimizing corrosion and other inservice degradation mechanisms. In addition, the staff reviewed the provisions for secondary water cleanup.

In its review, the staff applied the following regulatory criteria to the review of steam generator materials:

- GDC 1 requires that "[SSCs] important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function...Appropriate records...shall be maintained...."

- GDC 14 requires that "The [RCPB] shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture."

- GDC 15 requires that, "The reactor coolant system and associated...systems shall be designed with sufficient margin to assure that design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences."

- GDC 31 requires that "The [RCPB] shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized."

- Appendix B to 10 CFR Part 50 requires that measures be established to control the cleaning of material and equipment in accordance with work and inspection procedures to prevent damage or deterioration.

The NRC staff reviewed WBN Unit 2 FSAR changes related to steam generator materials in accordance with SRP Section 5.4.2.1, Revision 2, "Steam Generator Materials," issued July 1981, and reviewed the compatibility of steam generator materials with the primary and secondary coolant and cleanliness control.

In Section 5.4.2 of the SER, the NRC staff concluded that TVA had met all regulatory requirements for the steam generators; however, a concern related to potential tube degradation resulting from flow-induced vibration in Westinghouse Model D steam generators was not resolved. In SSER 1, issued September 1982, the staff documented discussions of proposed modifications to the Model D steam generators. In SSER 4, issued March 1985, the staff documented that an NRC-approved modification of the steam generators for both WBN units had been completed and concluded that WBN could be operated at 100-percent power.

<u>Compatibility of Materials with the Primary and Secondary Coolant, and Cleanliness Control</u>

In WBN Unit 2 FSAR Amendment 95, TVA revised FSAR Section 5.5.2.3.2, "Natural Circulation Flow," to further describe the safety valves used for steam release to maintain the reactor at hot standby. TVA revised the description to read that "Steam release to maintain the reactor at hot standby is accomplished via the main steam power-operated atmospheric relief valves, or the safety valves, if needed (safety grade source) or the preferred flow path through the steam dump valves."

TVA also revised FSAR Section 5.5.2.3.5, "Compatibility of Steam Generator Tubing with Primary and Secondary Coolant," to remove the description of the specific secondary water treatment chemical from the description of the secondary water chemistry control program. Previously, the FSAR indicated that ethanolamine and ammonium hydroxide would be used for pH control and hydrazine would be used to create a reducing environment and for oxygen scavenging.

In its review, the NRC staff used the regulatory guidance in SRP Section 5.4.2.1, "Steam Generator Materials," which refers to BTP Materials Engineering Branch (MTEB) 5-3, "Monitoring of Secondary Side Water Chemistry in PWR Steam Generators," for detailed guidance on maintenance and monitoring of secondary water chemistry. BTP MTEB 5-3 does not specify the chemicals to be used for secondary water chemistry control. FSAR Section 10.3.5 addresses secondary water chemistry in more detail and indicates that the program is based on the latest EPRI and Westinghouse PWR secondary water chemistry guidelines. FSAR Section 10.3.5 also discusses the specific chemicals to be used for secondary water treatment.

The other relevant recommendation of BTP MTEB 5-3 is that the secondary water chemistry monitoring and control program should identify a sampling schedule for critical parameters during each mode of operation, as well as the acceptance control criteria for these parameters. Meeting this recommendation is assured through compliance with the EPRI guidelines, which specify sampling intervals and the parameters to be sampled. Section 10.3.4 of this SSER includes the NRC staff's evaluation of secondary water chemistry.

With respect to secondary water chemistry, SRP Section 5.4.2.1 states that the requirements of GDC 14, 15, and 31 are met if the secondary coolant purity is monitored as described in BTP MTEB 5-3. Because the removal of the reference to the specific water treatment chemicals does not impact the ability to meet the criteria of BTP MTEB 5-3, the NRC staff concludes that this change is acceptable. Deletion of a reference to a specific secondary water treatment chemical does not affect compliance with GDC 14 as it relates to preventing corrosion-related degradation of the RCPB.

In FSAR Amendment 95, TVA added information to FSAR Section 5.5.2.3.2 regarding the main steam safety valves related to operation under natural circulation flow. SER Section 5.4.3 contains the NRC staff's evaluation of WBN operation under natural circulation flow.

Based on the above, the NRC staff concludes that the steam generator materials will continue to meet the applicable regulatory criteria of GDC 1, 14, 15, and 31 and Appendix B to 10 CFR Part 50.

5.4.4 Pressurizer Relief Tank

The NRC staff used the requirements of GDC 2, "Design Bases for Protection against Natural Phenomena," and the guidance of RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," and Regulatory Position C.2 of RG 1.29, "Seismic Design Classification," in its review of FSAR Section 5.5.11, "Pressurizer Relief Tank." The staff reviewed WBN Unit 2 FSAR Amendment 100, Section 5.5.11, and concluded that TVA made no substantive changes to Section 5.5.11. The staff also reviewed the information provided by TVA in its letter dated July 31, 2010, which clarified the number of rupture discs on the pressurizer relief tank. Therefore, the requirements of GDC 2 and the guidance of RG 1.26 and Regulatory Position C.2 of RG 1.29 are met.

Based on its evaluation of the information provided by TVA and its previous evaluation, as documented in the SER and its supplements, the NRC staff concludes that the failure of the pressurizer relief tank does not affect the integrity of the RCPB or the capability to shut down the plant safely. WBN Unit 2 FSAR Section 5.5.11 is, therefore, acceptable.

6 ENGINEERED SAFETY FEATURES

6.1 Engineered Safety Features Materials

6.1.2 Organic Materials

In Section 6.1.2 of NUREG-0847, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2," issued June 1982 (the SER), the staff of the U.S. Nuclear Regulatory Commission (NRC) concluded that the protective coating systems and their applications are acceptable and meet the requirements of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities." The NRC staff concluded that the coating systems and their applications meet the positions in Regulatory Guide (RG) 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," issued June 1973, with an acceptable alternative to American National Standards Institute (ANSI) N101.4, "Quality Assurance for Protective Coatings Applied to Nuclear Facilities," issued 1972, and the testing requirements of ANSI N101.2, "Protective Coatings (Paints) for Light-Water Nuclear Reactor Containment Facilities," issued 1972. The coating systems chosen by the Tennessee Valley Authority (TVA) have been qualified under conditions that take into account the postulated design-basis accident (DBA) conditions.

The NRC staff reviewed Amendments 92 through 99 to the Watts Bar Nuclear Plant (WBN) Unit 2 final safety analysis report (FSAR). TVA made only minor changes to wording and format and maintained its commitment to meet the positions of RG 1.54, with the acceptable alternative to ANSI N101.4-1972 and the testing requirements of ANSI N101.2-1972.

Based on the NRC staff's review of the information provided by TVA in its amendments to the FSAR, the staff concludes that the changes are acceptable. The staff's conclusions in the SER remain valid.

6.1.3 Postaccident Emergency Cooling Water Chemistry

In the SER, the NRC staff concluded that the postaccident emergency cooling water chemistry meets the minimum pH acceptance criterion of Section 6.1.1 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (hereafter referred to as the SRP); the positions of the NRC's Branch Technical Position Materials Engineering Branch 6-1, "pH for Emergency Coolant Water for PWRs," issued July 1981; and the requirements of General Design Criterion (GDC) 14, "Reactor Coolant Pressure Boundary," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50.

In FSAR Amendments 92 through 99, TVA revised the final postaccident pH value from 8.1 to 7.5 and also made minor wording and format changes. TVA stated that the sump pH after a loss-of-cooling accident (LOCA) remains within the range of 7.5 to 10.0 for the duration of the event. Since the revised pH value remains within the acceptance criterion (greater than 7.0), the NRC staff concludes that the changes are acceptable.

6.2 Containment Systems

6.2.1 Containment Functional Design

The NRC staff reviewed FSAR Amendment 97 to evaluate the changes proposed by TVA after the NRC issued the SER. The staff reviewed the information provided by TVA in FSAR Section 6.2.1 to verify that the information continues to meet the following relevant regulatory requirements with respect to protection against natural phenomena, environmental effects, containment design, and monitoring radioactivity releases:

- GDC 4, "Environmental and Dynamic Effects Design Bases," requires that structures, systems, and components important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation and postulated accidents.

- GDC 2, "Design Bases for Protection Against Natural Phenomena," requires that structures, systems, and components important to safety be designed to withstand the effects of natural phenomena.

- GDC 16, "Containment Design," requires that the reactor containment and associated systems be provided to ensure that containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

  Since the primary reactor containment is the final barrier of the defense-in-depth concept to protect against the uncontrolled release of radioactivity to the environs, preserving containment integrity under the dynamic conditions imposed by postulated LOCAs is essential.

- GDC 50, "Containment Design Basis," requires that the reactor containment structure and containment heat removal system be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA.

- GDC 38, "Containment Heat Removal," requires that a system to remove heat from the reactor containment be provided and that it reduce rapidly the containment pressure and temperature following any LOCA and maintain them at acceptably low levels.

- GDC 39, "Inspection of Containment Heat Removal System," requires that the containment heat removal system be designed to permit periodic inspection of important components to ensure the integrity and capability of the system.

- GDC 40, "Testing of Containment Heat Removal System," requires that the containment heat removal system be designed to permit periodic pressure and functional testing to ensure operability.

- GDC 13, "Instrumentation and Control," requires that instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions, as appropriate, to ensure adequate safety.

- GDC 64, "Monitoring Radioactivity Releases," requires that means be provided for monitoring the reactor containment atmosphere for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

On the basis of its review of the information provided by TVA in FSAR Amendment 97, the NRC staff concluded that TVA proposed changes that affected the containment functional design basis previously evaluated by the staff in the SER, as well as some changes that required verification of acceptability by the staff and other changes that were administrative and editorial. The staff reviewed these changes as noted below.

As described in WBN Unit 2 FSAR Section 6.2.1.1.1, the containment is designed to ensure that an acceptable upper limit of leakage of radioactive material is not exceeded under DBA conditions. In FSAR Section 6.2.1.3.3, TVA proposed a revised long-term containment analysis for the design-basis LOCA in support of a proposed reduction for the minimum allowable weight of ice in the ice condenser. The significant changes in the revised analysis, when compared with the original analysis, include (1) a reduced initial ice inventory and (2) changes to the mass and energy release model and the containment heat sink model. The NRC staff reviewed the assumptions and results of the revised analysis described in the FSAR and found that the analysis is consistent with a similar reanalysis done by the staff in 2001 for WBN Unit 1, which the staff approved in WBN Unit 1 License Amendment No. 33, dated November 29, 2001. Based on the similarity between the analyses for WBN Units 1 and 2, the staff concludes that the method of analysis, modeling assumptions, and results of the Unit 2 reanalysis are acceptable. Therefore, the NRC staff concludes that the lower ice weight may be used as the basis for establishing the technical specification (TS) limit for the minimum allowable weight of ice in the ice condenser in WBN Unit 2.

Based on its review of the information provided by TVA in FSAR Amendment 97, and its previous evaluation as documented in the SER and WBN Unit 1 License Amendment No. 33, the NRC staff concludes that the Unit 2 containment functional design meets the relevant requirements of GDC 2, 4, 16, 50, 38, 39, 40, 13, and 64 of Appendix A to 10 CFR Part 50 with respect to protection against natural phenomena, environmental effects, containment design, and monitoring radioactivity releases and that the design is consistent with the acceptance criteria in SRP Section 6.2.1.

6.2.2 Containment Heat Removal Systems

The NRC staff reviewed Section 6.2.2 of FSAR Amendment 97, to evaluate the changes proposed by TVA after the NRC issued the SER. The staff reviewed the information in this section, as revised by the proposed changes, to verify that it continues to meet the relevant regulatory requirements with respect to the capability of the system to accomplish its safety functions.

The staff used the following regulatory requirements during its review:

- GDC 38 requires that a system to remove heat from the reactor containment be provided and that it reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintain them at acceptably low levels. Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to ensure that, for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available), the system safety function can be accomplished, assuming a single failure.

- GDC 39 requires that the containment heat removal system be designed to permit periodic inspection of important components to ensure the integrity and capability of the system.
- GDC 40 requires that the containment heat removal system be designed to permit periodic pressure and functional testing to ensure operability.
- GDC 4 requires that structures, systems, and components important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation and postulated accidents.
- 10 CFR 50.46(b)(5) requires that, after any calculated successful initial operation of the emergency core cooling system (ECCS), the calculated core temperature be maintained at an acceptably low value and that decay heat be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The staff also referred to the regulatory guidance of SRP Section 6.2.2 during its review.

On the basis of its review of the information provided by TVA in FSAR Amendment 97, the NRC staff concluded that TVA proposed changes that affected the containment heat removal system design previously evaluated by the staff in the SER, as well as some changes that required verification of acceptability by the staff and other changes that were administrative and editorial. The staff reviewed these changes as noted below.

As described in Section 6.2.2 of WBN Unit 2 FSAR Amendment 97, containment heat removal capability is provided by the ice condenser, the air return fan system, the containment spray system, and the residual heat removal (RHR) spray system. The containment heat removal systems are designed to provide means of removing containment heat, without loss of functional performance in the postaccident containment environment, and to operate without benefit of maintenance for the duration of time needed to restore and maintain containment conditions at atmospheric pressure.

In FSAR Amendment 97, TVA clarified its text concerning conflicting requirements for control rod drive mechanism coolers and lower compartment coolers (LCCs). In FSAR Section 6.2.2.1, TVA described LCC units that may be operated continuously throughout all accidents, except for a main steamline break (MSLB), which does not initiate a containment Phase B isolation signal. After an MSLB, all four coolers (two from Train A and two from Train B) are started within 1.5 to 4 hours, although only two are required, to recirculate air throughout the lower containment spaces and to prevent hot spots from developing. During or after a LOCA, the LCC units, including their fans, are not required to be operable. The NRC staff concludes that the change to the FSAR is acceptable because the heat removal system will maintain environmental conditions for optimum equipment operation during all modes of plant operation, including normal, transient, and accident conditions. Therefore, the system meets the requirement of GDC 4.

In FSAR Section 6.2.2.2, TVA revised the system description to state that the RHR spray ring headers contain 147 nozzles per header and deliver a design flow of 2,000 gallons per minute per header. TVA stated that each RHR spray header will perform its design function with 142 unobstructed spray nozzles.

The long-term containment pressure analysis (FSAR Section 6.2.1.3.3) assumes a containment spray pump flow rate of 4,000 gallons per minute and an essential raw cooling water (ERCW) temperature of 88 degrees Fahrenheit (F) for the spray heat exchanger and component cooling heat exchanger. TVA stated that, to provide additional conservatism, the containment analysis used an ERCW temperature of 88 degrees F, although the containment spray, component cooling, and RHR heat exchangers' heat capacity rate values are based on an ERCW temperature of 85 degrees F.

Based on its review of the information provided by TVA in FSAR Amendment 97 and its previous review, as documented in the SER, the NRC staff concludes that the design of the containment heat removal system meets the relevant requirements of GDC 38, 39, and 40 and is consistent with the acceptance criteria in SRP Section 6.2.2.

6.2.3 Secondary Containment Functional Design

The NRC staff reviewed Section 6.2.3 of WBN Unit 2 FSAR Amendment 97 to verify that TVA's proposed changes to the secondary containment functional design meet the relevant requirements of GDC 2; 4; 5, "Sharing of Structures, Systems, and Components," 16, "Containment Design," 60, "Control of Releases of Radioactive Materials to the Environment"; and 61, "Fuel Storage and Handling and Radioactivity Control," of Appendix A to 10 CFR Part 50 with respect to protection against natural phenomena, environmental effects, and control of releases of radioactive materials to the environment, and of Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," to 10 CFR Part 50 The staff also referred to the regulatory guidance of SRP Section 6.2.3 during its review. As stated in Section 6.2.3 of the SER, the secondary containment functional design goal is to preclude airborne fission products that may result from a LOCA from reaching the atmosphere in an uncontrolled manner.

On the basis of its review of the information provided by TVA in FSAR Amendment 97, the NRC staff concluded that TVA proposed changes that affected the secondary containment functional system design previously evaluated by the staff in the SER, as well as some changes that required verification of acceptability by the staff and other changes that were administrative and editorial. The staff reviewed these changes as noted below.

As described in FSAR Section 6.2.3, the structures that are included as part of the secondary containment system include the shield building of each reactor unit, the auxiliary building, the condensate demineralizer waste evaporator building, and the ERCW pipe tunnels adjacent to the auxiliary building. The design bases for the secondary containment structures were devised to ensure that an effective barrier exists for airborne fission products that may leak from the primary containment, or from the auxiliary building fuel-handling area, during a LOCA or a fuel-handling accident. This is consistent with GDC 16, which requires, in part, that the reactor containment and associated systems establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment, GDC 60, which requires, in part, that the nuclear power unit design include means to control suitably the release of radioactive materials produced during normal operation and GDC 61, which requires in part, that the fuel storage and handling system be designed with appropriate containment, confinement, and filtering systems, and GDC 4.

TVA revised FSAR Section 6.2.3.2.3 about the auxiliary building gas treatment system (ABGTS) to state that, during periods when the primary containment, the annulus, or both, of both units are open to the auxiliary building, the auxiliary building secondary containment enclosure (ABSCE) also includes these areas. TVA has designed the ABGTS to establish a negative pressure in these additional areas for this configuration. During fuel-handling operations in this configuration, a high-radiation signal from the spent fuel pool radiation monitors will result in containment ventilation isolation, in addition to auxiliary building isolation, and will start the ABGTS. Similarly, a containment ventilation isolation signal, including one generated by a high-radiation signal from the containment purge air exhaust radiation monitors, will initiate an auxiliary building isolation and start the ABGTS. These actions will ensure the proper operation of the ABSCE. This is consistent with GDC 5, which requires, in part, that SSCs important to safety not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, GDC 4, GDC 60, and GDC 61.

TVA revised FSAR Section 6.2.3.2.2 about the emergency gas treatment system (EGTS) to state that the annulus vacuum control subsystem aids in containment pressure relief by exhausting the containment vent air after it goes through the containment vent air cleanup units and is discharged into the annulus, then into the auxiliary building exhaust stack.

TVA revised FSAR Section 6.2.3.3.2 to state, in part, the following:

> Air leakage into the shield building annulus was assumed to be 250 cfm [cubic feet per minute] at the post accident annulus control setpoint for a postulated single failure of one EGTS train. A more conservative air inleakage value of 957 cfm was assumed for an alternate single failure scenario which results in one pressure control train in full exhaust to the shield building exhaust stack while the other train remains functional. This higher inleakage value is based upon the higher negative pressure that will exist within the annulus during this scenario.
>
> Only one train of the EGTS was assumed to operate, during the first single failure scenario, allowing for a possible single failure in the other. An alternate single failure scenario was analyzed which postulates one pressure control train failing which results in full exhaust to the shield building exhaust stack while the other train remains functional.

Based on its review of the information provided by TVA in FSAR Amendment 97 and its previous evaluation, as documented in the SER, the NRC staff concludes that the secondary containment functional design meets the relevant requirements of GDC 2, 4, 5, 16, 60, and 61, and Appendix J to 10 CFR Part 50 and is consistent with the acceptance criteria in SRP Section 6.2.3.

6.2.4 Containment Isolation Systems

The NRC staff reviewed Section 6.2.4 of WBN Unit 2 FSAR Amendment 97 and concluded that there were both substantive and editorial changes related to the containment isolation systems, as noted below. In conducting its review, the staff used the regulatory requirements of GDC 16;

GDC 54, "Systems Penetrating Containment"; GDC 55, "Reactor Coolant Pressure Boundary Penetrating Containment"; GDC 56, "Primary Containment Isolation"; and GDC 57, "Closed System Isolation Valves," and the guidance of SRP Section 6.2.4.

TVA revised FSAR Section 6.2.4.1, Item 5, of the design-bases description of alternative containment isolation provisions to state that relief valves used as containment isolation valves have a setpoint greater than 1.5 times internal containment design pressure. The NRC staff concluded that the change is consistent with Acceptance Criterion II.6.g of SRP Section 6.2.4, Revision 2, issued July 1981, and is, therefore, acceptable. TVA revised the entry for penetration X-30 to show the addition of a relief valve RFV-63-28 outside containment for overpressure protection of the penetration. By letter dated July 31, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML102290258), TVA stated that there are five relief valves used as containment isolation valves and that the valves do not directly discharge outside containment. The discharge of RFV-63-28 is to a header pipe that is routed back into containment through penetration X-24. The change reflects the current plant configuration and is acceptable to the staff.

TVA revised FSAR Section 6.2.4.1, Item (4)(b), of the design-bases description of design requirements for containment isolation barriers to state that an enclosed system inside containment will withstand external pressure and temperature equal to the containment structural integrity test pressure and containment design temperature. The change is consistent with Section 3.5, Criterion (5) (for closed systems inside containment) of ANSI N271-1976, "Containment Isolation Provisions for Fluid Systems", which is endorsed by RG 1.141, "Containment Isolation Provisions for Fluid Systems," issued April 1978, and is, therefore, acceptable.

TVA revised FSAR Section 6.2.4.2 to describe the provision for manual operator action to isolate ERCW, component cooling water, and control air supply lines, in the event the associated outboard isolation valves (FCV-32-110, FCV-67-107, FCV-70-92, or FCV-70-140) fail in the open position, concurrent with a high-energy line break inside containment that results in a breach of the potentially affected line in containment. The inboard isolations are check valves and would permit the continued discharge of water or air into the containment, potentially resulting in the dilution of the recirculation pool boron concentration or in exceeding the containment design pressure. A similar condition also exists for WBN Unit 1. The NRC staff approved a similar FSAR change for WBN Unit 1 in License Amendment No. 51, dated March 29, 2004 (ADAMS Accession No. ML040900172). Based on its acceptance of the similar change for Unit 1, the staff concludes that the change for Unit 2 is acceptable.

TVA revised FSAR Section 6.2.4.2.3 regarding penetration type XV—personnel access to explain that both personnel airlock doors may be allowed open during fuel-handling activities under special administrative controls. The NRC staff approved a similar change for WBN Unit 1 in License Amendment No. 26, dated August 24, 2000 (ADAMS Accession No. ML003744777). Based on the staff's acceptance of a similar change for Unit 1, the change for Unit 2 is acceptable.

TVA revised FSAR Section 6.2.4.2.3 regarding penetration type XVIII—ice blowing to explain that the ice blowing line penetration has a blind flange with double O-rings installed outside the containment. The prior configuration had a blind flange on the penetration inside containment with a single O-ring and a blind flange on the penetration outside containment with a single O-ring. The double O-ring configuration provides for local leakage rate testability of the penetration with a Type B test, as specified in Appendix J to 10 CFR Part 50, which satisfies GDC 16; GDC 53, "Provisions for Containment Testing and Inspection"; and GDC 54. The change is consistent with the similar configuration approved on Unit 1 and is, therefore, acceptable for Unit 2.

TVA revised FSAR Section 6.2.4.3 to add exception "e," which describes the temporary modification of the ice blowing and negative return penetrations during plant operating modes 5, 6, and defueled, to support the filling of ice condenser baskets. The blind flanges would be removed and temporary piping with manual valves would be installed through the penetration. A seal would be placed between the temporary piping and the penetration, which, along with administrative controls on the manual isolation valves, would ensure timely containment closure subsequent to a fuel-handling accident. The NRC staff approved a similar change for WBN Unit 1 in License Amendment No. 26, dated August 24, 2000 (ADAMS Accession No. ML003744777). Based on the staff's acceptance of a similar change for Unit 1, the change for Unit 2 is acceptable.

TVA revised FSAR Section 6.2.4.2.3 regarding penetration type XIX—electrical to explain that the design, construction, and installation of fiber optic feed-throughs in modular-type electrical penetrations conform to Institute of Electrical and Electronics Engineers (IEEE) Standard (Std.) 317-1983, "IEEE Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations." RG 1.63, Revision 3, "Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants," issued February 1987, endorses the use of IEEE Std. 317-1983; therefore, the change is acceptable.

TVA made other changes to Section 6.2.4 that the NRC staff concluded were editorial and, therefore, acceptable. The staff noted some inconsistencies in the information provided in Tables 6.2.4-1 and 6.2.6-2, regarding containment penetrations and barriers and containment isolation valves subjected to Type C testing, respectively. In its letter dated July 31, 2010, TVA provided acceptable explanations or commitments to make changes to correct the inconsistencies. TVA corrected the information in subsequent WBN Unit 2 FSAR Amendments 98, 99, and 100.

Based on its review of the information provided by TVA, as discussed above, and its previous review as documented in the SER, the NRC staff concludes that the containment isolation systems meet the relevant requirements of GDC 16, 54, 55, 56, and 57 and the acceptance criteria of SRP Section 6.2.4 and are, therefore, acceptable.

6.2.5 Combustible Gas Control Systems

TVA revised FSAR Section 6.2.5.1 by stating the following:

> In an accident more severe than the design-basis loss-of-coolant accident, combustible gas is predominantly generated within containment as a result of the following:
>
> (1) Fuel clad-coolant reaction between the fuel cladding and the reactor coolant.
>
> (2) Molten core-concrete interaction in a severe core melt sequence with a failed reactor vessel.

If a sufficient amount of combustible gas is generated, it may react with the oxygen present in the containment at a rate rapid enough to lead to a containment breach or a leakage rate in excess of Technical Specification limits. Additionally, damage to systems and components essential to continued control of the post-accident conditions could occur.

The systems provided for combustible gas control have the following functional and mechanical requirements:

(1) The air return fans enhance the ice condenser and containment spray heat removal operation by circulating air from the upper compartment to the lower compartment through the ice condenser, and then back to the upper compartment. Hydrogen concentration is limited in potentially stagnant regions by providing air flow in these regions.

(2) The HAS [hydrogen analyzer system] provides the capability for extracting a sample and obtaining the measurement necessary to determine the volume percent concentration for hydrogen present in the sample. The system provides indication and alarms of volume percent concentrations in the main control room. Indication is also provided at the remote control center.

(3) The HMS [hydrogen mitigation system] is designed to increase the containment capability to accommodate hydrogen that could be released during a degraded core accident. The system is based on the concept of controlled ignition using thermal igniters.

(4) The air return fans and HAS are designed to operate continuously during accident conditions. The HMS igniter assemblies are qualified for a 30-year life of operational cycles.

(5) The combustible gas control system is designed for periodic testing and inspection.

TVA listed the systems provided for combustible gas control in FSAR Section 6.2.5: (1) containment air return system, (2) HAS, and (3) HMS.

Containment Air Return System:

TVA provided a design description of the containment return air fans operation and stated that the fans automatically start to operate 10 minutes after a phase B containment isolation signal (described in WBN Unit 2 FSAR Section 7.3.1.1.1) is received. The system reduces containment pressure after a LOCA blowdown and provides continuous long-term mixing of the containment atmosphere during post-LOCA conditions by recirculating air in potentially stagnant regions and dead-ended compartments, to prevent excessive hydrogen buildup. A portion of the return air ductwork is embedded in concrete to prevent damage from the buildup of pressure during a LOCA, while the exposed ductwork is designed to withstand the beyond-DBA environment. Backdraft dampers are provided to prevent back flow from the lower compartment to the upper compartment under a differential pressure of 15 pounds per square inch gauge.

These dampers prevent steam from bypassing the ice condenser during the initial blowdown. The system has redundancy, is single-failure-proof, and will remain operable with a loss of onsite or offsite power. The fans also have a manual start capability.

In Request for Additional Information (RAI) 6.2.5-3, the NRC staff asked TVA to describe the analysis and the results that demonstrate the effectiveness of the proposed method for providing a mixed atmosphere during a beyond-DBA, such that the combustible gases will not accumulate within a compartment or cubicle to form a detonable mixture that could cause a loss of containment integrity. In addition, the NRC asked TVA to describe the approach taken to demonstrate that mixing was not overestimated by the computer code used for the analysis.

In its letter dated July 31, 2010, TVA referred to the Hanford Engineering Development Laboratory series of large-scale mixing tests and studies using the CLASIX computer code documented in Supplement 6 of the Sequoyah Nuclear Plant (SQN) SER (NUREG-0011, "Safety Evaluation Report Related to the Operation of Sequoyah Nuclear Plant, Units 1 and 2, Dockets Nos. 50-327 and 50-328, Tennessee Valley Authority," issued September 1980). These tests and studies were performed to show that large hydrogen gradients would not occur during a degraded core accident. The results of these tests and studies demonstrated that combustible gas will not accumulate within a compartment or cubicle to form a detonable mixture that could cause a loss of containment integrity. TVA stated that the WBN Unit 2 containment structure is nearly identical to the SQN containment structure, as are the air return fans and the hydrogen ignition system. TVA proposes that, since WBN Unit 2 and SQN are sister plants, the Hanford tests and CLASIX studies are directly applicable to WBN Unit 2. Based on the similarity of SQN and WBN Unit 2, the NRC staff concludes that the comparison is reasonable and is, therefore, acceptable.

In RAI 6.2.5-5, the NRC staff asked TVA to explain why FSAR Section 6.2.5.2 stated that the "Ductwork not protected by embedment is designed to withstand the LOCA environment," instead of "...beyond-design-basis accident environment." In its letter dated July 31, 2010, TVA stated that it had revised WBN Unit 2 FSAR Sections 6.2.5.2 and 6.8 by removing references to the LOCA and replacing them with references to the beyond-DBA. During its review of FSAR Amendment 101, the NRC staff verified that the change was acceptable.

Hydrogen Analyzer System (HAS):

In FSAR Section 6.2.5.2, TVA provided a design description of the HAS. The sampling system consists of a single, nontrained detection loop. The analyzer is fed by one process sample line and returns to containment on one process effluent line. This line is equipped with two manually controlled isolation valves on both the sample and return lines. The system is installed at two locations. The sample and return lines are each provided with two containment isolation valves manually operated from the main control room. During postaccident conditions, the system continuously extracts samples from the containment, measures hydrogen partial pressure, and provides a main control room alarm and indication of volume percent hydrogen concentration. The analyzer is calibrated to measure the volume concentration of hydrogen between 0 and 10 percent with an accuracy of ±0.2 percent hydrogen. The hydrogen analyzer components are seismically supported.

TVA stated that the hydrogen analyzer is a highly reliable commercial-grade Category 3 instrument, as defined in RG 1.97, Revision 4, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," issued June 2006, and permitted by RG 1.7, Revision 3, "Control of

Combustible Gas Concentrations in Containment," issued March 2007. The regulations at 10 CFR 50.44 require, in part, that equipment be provided for monitoring hydrogen in the containment. RG 1.7 provides one method that is acceptable to the NRC staff of implementing the regulation. In RAI 6.2.5-1, the NRC staff asked TVA to describe how the criteria in Section C.2.1 of RG 1.7, Revision 3, for a commercial-grade hydrogen analyzer are met. In its response letter dated July 31, 2010, TVA explained how the criteria are met. Based on its review of the information provided by TVA, the staff concludes that the HAS is acceptable, because the system meets the criteria of RG 1.7, Revision 3, Section C.2.1.

Hydrogen Mitigation System (HMS):

In FSAR Section 6.2.5.2, TVA provided a design description of the HMS. The HMS uses durable thermal igniters to burn any postaccident hydrogen release in a controlled manner as soon as the hydrogen concentration exceeds the lower flammability limit at any location in the containment. The system thereby prevents the accumulation of explosive concentrations. The igniters maintain surface temperatures in excess of the required minimum for extended periods, initiate combustion, and continuously operate in various combustion environments. In consideration of a single failure, the igniters are divided into two equal redundant groups, each with independent and separate controls, power supplies, and locations, which ensures adequate coverage. Each group is provided with manual control and status indication in the main control room. The power supply of each group of igniters is backed by power from the emergency diesel generators. A total of 68 igniters are distributed in the regions of containment in which hydrogen could be released or where it could accumulate in sufficient quantities. At least two igniters, controlled and powered redundantly, are located in each of these regions. The components of the HMS inside the containment are seismically supported.

In RAI 6.2.5-2, the NRC staff asked TVA to describe the approach for demonstrating equipment survivability in the beyond-DBA environment conditions inside the containment, compared to Section C.2.1, Item (1), of RG 1.7, Revision 3, which identifies acceptable approaches for demonstrating equipment survivability.

In its letter dated July 31, 2010, TVA referred to supplemental SER (SSER) 8, issued January 1992, for WBN Unit 1, in which the NRC staff found that WBN Unit 1 met the requirements of 10 CFR 50.44, "Combustible Gas Control for Nuclear Power Reactors," and concluded that the issue of equipment survivability during a postulated degraded-core accident was satisfactorily resolved. Based on the similarity of the igniters, the air return fans, and the hydrogen monitors, TVA proposed that a plant-specific analysis of degraded-core accidents was not necessary for WBN Unit 2. Based on the similarity of WBN Units 1 and 2 and the HMS at each unit, the NRC staff considers the WBN Unit 2 HMS acceptable.

In Section 6.2.5.2 of the FSAR, TVA stated that the combustible gas control system is subjected to periodic testing and inspection to demonstrate its availability.

Based on its review of the information provided by TVA, as discussed above, the NRC staff concludes that the design of the combustible gas control system meets the requirements of GDC 5; GDC 41, "Containment Atmosphere Cleanup"; GDC 42, "Inspection of Containment Atmosphere Cleanup Systems"; and GDC 43 and 10 CFR 50.44 and is, therefore, acceptable.

6.2.6 Containment Leakage Testing

The NRC staff reviewed Section 6.2.6 of WBN Unit 2 FSAR Amendment 97 and concluded that the section contained both substantive and editorial changes regarding containment leakage testing, as noted below.

TVA revised FSAR Section 6.2.6.1 to state that periodic leakage rate testing subsequent to the preoperational test will be conducted in accordance with Option B of Appendix J to 10 CFR Part 50, with approved exemptions. The NRC staff previously approved this change for WBN Unit 1 in License Amendment No. 5, dated May 27, 1997 (ADAMS Accession No. ML020790130). Because of the similarity between WBN Units 1 and 2 and the Commission's instructions in SRM-SECY-07-0096, "Possible Reactivation of Construction and Licensing Activities for the Watts Bar Nuclear Plant Unit 2," dated July 25, 2007, the change is acceptable for Unit 2.

TVA FSAR Section 6.2.6.2 states that local leakage rate testing (Type B and Type C tests) will be conducted in accordance with Option B of Appendix J to 10 CFR Part 50. The NRC staff previously approved this change for WBN Unit 1 in License Amendment No. 5. Because of the similarity between WBN Units 1 and 2 and the Commission's instructions in SRM-SECY-07-0096, the change is acceptable for Unit 2.

TVA made other changes to FSAR Section 6.2.6 that were editorial and nonsubstantive and that the NRC staff found acceptable. The staff noted several inconsistencies in the information provided in Tables 6.2.6-2 and 6.2.4-1. In its letter dated July 31, 2010, TVA provided acceptable explanations or commitments to correct the information in the two tables. TVA's FSAR Amendments 98, 99, and 100 subsequently corrected the inconsistencies.

The NRC staff noted that TVA's changes to Section 6.2.6 in FSAR Amendment 97, regarding the implementation of Option B of Appendix J, were incomplete, because several statements remained regarding performing water-sealed valve leakage tests "as specified in 10 CFR [Part] 50, Appendix J." With the adoption of Option B, the specified testing requirements are no longer applicable; Option A to Appendix J retains these requirements. The NRC discussed this discrepancy with TVA in a telephone conference on September 28, 2010. TVA stated that it would remove the inaccurate reference to Appendix J for specific water testing requirements in a future FSAR amendment. This is Open Item 47 (Appendix HH).

Based on its review of FSAR Amendment 97, as discussed above, and its previous evaluation as documented in the SER, the NRC staff concludes that containment leakage testing meets the relevant requirements of GCD 52, 53, and 54 and Option B of Appendix J to 10 CFR Part 50 and conforms to the guidance of SRP Section 6.2.6. Therefore, WBN Unit 2 FSAR Section 6.2.6 is acceptable.

6.4 Control Room Habitability

The staff reviewed Section 6.4 of WBN Unit 2 FSAR Amendment 97 on habitability systems against the regulatory requirements of NUREG-0737, "Clarification of TMI [Three Mile Island] Action Plan Requirements," issued November 1980; TMI Action Plan Item III.D.3.4, and GDC 2; 4; and 19, "Control Room." The staff also referred to the guidance of RG 1.52, Revision 2, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear

Power Plants," issued March 1978, and RG 1.78, Revision 1, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," issued December 2001.

The NRC staff previously approved TVA's request to adopt Technical Specification Task Force (TSTF) Change Traveler TSTF-448, Revision 3, "Control Room Envelope Habitability," for WBN Unit 1 in License Amendment No. 70, dated October 8, 2008 (ADAMS Accession No. ML082730261). Because the main control room envelope is shared for WBN Units 1 and 2, the NRC staff had previously reviewed the changes to the common systems in the main control room envelope habitability program for WBN Unit 1 License Amendment No. 70. Therefore, the systems are acceptable for WBN Unit 2.

On this basis of the NRC staff's safety evaluation for WBN Unit 1 License Amendment No. 70 and its previous evaluation as documented in the SER, the staff concludes that the control room habitability systems meet the relevant requirements of TMI Action Plan Item III.D.3.4 and GDC 2, 4, and 19 and the guidance of RGs 1.52 and 1.78 and are, therefore, acceptable for WBN Unit 2.

6.5 Engineered Safety Feature Filter Systems

6.5.1 Engineered Safety Feature Atmosphere Cleanup System

GDC 41, 42, and 43 of Appendix A to 10 CFR Part 50 require that containment atmosphere cleanup systems be provided as necessary to reduce the amount of radioactive material released to the environment following a postulated design basis accident. They also require that these systems be designed to permit appropriate periodic inspection and testing to ensure their integrity, capability, and operability.

GDC 61 requires that fuel storage and handling systems, radioactive waste systems, and other systems that may contain radioactivity be designed to ensure adequate safety under normal and postulated accident "conditions and that they be designed with appropriate containment, confinement, and filtering systems. GDC 19 requires that adequate radiation protection be provided to permit access to and occupancy of the control room under accident conditions and for the duration of the accident without personnel radiation exposures in excess of 5 rem to the whole body.

RG 1.52, Revision 2, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," provides an acceptable method to the NRC staff of meeting the GDC requirements. The NRC staff also used the guidance of SRP Section 6.5.1, Revision 2, in its review of WBN Unit 2 FSAR Section 6.5.1.

WBN Unit 2 FSAR Section 6.5.1 states the following:

> Four Engineered Safety Feature (ESF) air cleanup systems' units are provided for fission product removal in post-accident environments. These are:
>
> (1) The emergency gas treatment system (EGTS) air cleanup units.
> (2) The Auxiliary Building gas treatment system (ABGTS) air cleanup units.
> (3) The Reactor Building purge system air cleanup units.

(4) The Main Control Room emergency air cleanup units.

TVA analyses these subsystems in FSAR Sections 6.2.3.3.2, 6.2.3.3.3, 9.4.6.2, and 6.4.4, respectively.

FSAR Amendment 97 revised Section 6.5.1.2.3 to discuss the performance of the filters in the reactor building purge system air cleanup units. The section now states that the high-efficiency particulate air (HEPA) filters installed in the air cleanup units for the EGTS, the ABGTS, the reactor building purge system, and the main control room are 1,000 cubic feet per minute (cfm) units designed to remove at least 99.97 percent of the particulates greater than 0.3 microns in diameter and to meet the requirements of military specification MIL-F-51068. The carbon adsorbers installed in the air cleanup units are Type II unit trays, fabricated in accordance with the requirements of American Association for Contamination Control (AACC) Standard CS-8T, "Tentative Standard for High Efficiency Gas-Phase Adsorber Cells." AACC-CS-8T has been superseded, and ANSI/American Society of Mechanical Engineers (ASME)-N509-1989, "Nuclear Power Plant Air-Cleaning Units and Components," specifies that ASME AG-1-1988, "Code on Nuclear Air and Gas Treatment," be used. Therefore, all new charcoal Type II cells shall meet AG-1, Section FD, with the exception that the 1991 version of the code may be used. Existing Type II cells do not have to be replaced to meet the AG-1 code if being refilled. New replacement charcoal adsorbent (for use in new and refilled Type II cells) shall be procured to meet the ASME AG-1-1991 requirements, in lieu of the 1988 version (or later version, provided proper evaluation justifies adequacy), with the exception that laboratory testing of the adsorbent be in accordance with American Society for Testing and Materials D3803-1989, "Standard Test Methods for Nuclear-Grade Activated Carbon." FSAR Table 6.5-5 lists the total numbers of filters and adsorber unit trays provided in each air cleanup unit.

TVA revised FSAR Section 9.4.6.2, the system description of the reactor building purge ventilating system, to state that the containment is vented into the shield building annulus, during normal operation, continuously, through the containment vent air cleanup units, which contain HEPA and charcoal filters, to maintain the containment pressure within the TS limits. Exhaust air mixes with the annulus atmosphere before it is discharged into the auxiliary building exhaust stack by the annulus vacuum control fan(s).

The NRC staff has reviewed the information provided by TVA in FSAR Amendment 97 and concludes that the engineered safety feature atmosphere cleanup systems meet the guidance of SRP Section 6.5.1, Revision 2. The design conforms to the guidelines of RG 1.52, Revision 2, and is, therefore, acceptable.

6.5.3 Fission Product Control System and Structures

SER Sections 6.2.3, 6.5.1, and 15.4.1 include the NRC staff's evaluation of the WBN secondary containment as a fission product control system.

The NRC staff reviewed the changes made by TVA to WBN Unit 2 FSAR Sections 6.2.3 and 6.5.1 after the NRC issued the SER and concluded that they were nonsubstantive and were, therefore, acceptable. The staff also reviewed the changes made by TVA to WBN Unit 2 FSAR Section 6.5.3 since the licensing of Unit 1. All of the changes except one were editorial in nature and were acceptable. The change that was not editorial appeared in FSAR Section 6.2.3, regarding the use of a more conservative air inleakage value for a specific single failure scenario. The NRC staff reviewed this change, as documented in Section 6.2.3 of this

SSER, and found it acceptable.

The NRC staff should verify that its conclusions in the review of FSAR Section 15.4.1 do not affect the conclusions of the staff regarding the acceptability of Section 6.5.3. This is Open Item 48 (Appendix HH).

8 ELECTRIC POWER SYSTEMS

8.1 General

The electric power system is the source of power for station auxiliaries during normal operation and for the reactor protection system and engineered safety features (ESFs) during abnormal operational and accident conditions. The NRC staff evaluated the design, design criteria, and design bases for the Watts Bar electric power system using the criteria set forth in the Standard Review Plan (SRP), Section 8.1, Table 8-1, "Acceptance Criteria and Guidelines for Electric Power Systems." The acceptance criteria and guidelines include the applicable general design criterion (GDC) from Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities"; branch technical positions, regulatory guides (RGs), and reports published by the staff (NUREG reports). Conformance to the applicable GDC and guidelines provides sufficient basis for acceptance of the electric power systems. In this regard, the staff reviewed Section 8.1 of Amendments 95 and 100 to the final safety analysis report (FSAR) to ensure that the information contained therein satisfies the requirements of 10 CFR Part 50 and that any supplemental information provided by the Tennessee Valley Authority (TVA) has been addressed in its application.

FSAR Amendment 95, Section 8.1.5.3, "Compliance to Regulatory Guides and Institute of Electrical and Electronics Engineers (IEEE) Standards," states that Watts Bar Nuclear Plant (WBN) Unit 2 complies with the guidance in RG 1.32, Revision 0, "Criteria for Power Systems for Nuclear Power Plants"; RG 1.81, Revision 1, "Shared and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants"; and Institute of Electrical and Electronics Engineers (IEEE) Standard 308-1971, "Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations," in meeting the NRC regulations in 10 CFR Part 50, Appendix A, GDC 5, "Sharing of Structures, Systems, and Components," and GDC 17, "Electric Power Systems." FSAR Section 3.1.2 states that the preferred and emergency electric power systems are shared. Because the NRC staff has not previously reviewed the capability of the preferred and emergency electric power systems for dual-unit operation, the staff requested, by letter dated July 12, 2010 (Agencywide Documents Access and Management Systems (ADAMS) Accession No. ML101530354), that TVA provide an executive summary of the analysis to support the following design requirements:

- dual-unit trip as a result of an abnormal operational occurrence
- accident in one unit and concurrent shutdown of the second unit (with and without offsite power)
- accident in one unit and spurious ESF actuation in the other unit (with and without offsite power)

In its response dated July 31, 2010 (ADAMS Accession No. ML102290258), TVA stated that it performed an analysis using the Electrical Transient Analyzer Program, Version 5.5.6N. The evaluation was performed for various configurations and included steady-state and transient voltages, equipment short-circuit currents, dynamic motor-starting equipment capability, bus loading, and degraded voltage analysis. The detailed analysis was limited to safety-related equipment and components powered from the safety-related boards. The scope of this analysis included the common station service transformers (CSSTs) A, B, C and D; 6.9-kilovolt (kV) shutdown boards; 6.9-kV start buses; 6.9-kV common boards; 6.9-kV unit boards; downstream 6.9-kV-480-volt (V) transformers; 480-V distribution systems loads, and all interconnections.

For the scenario in which an accident occurs in one unit and a concurrent shutdown of the second unit occurs with offsite power available, TVA determined that the auxiliary power system (APS) could adequately support the scenario for two-unit operation. The voltage recovery times were within the time limits so that the 6.9-kV shutdown board degraded voltage relays (DVRs) reset and would not separate the 6.9-kV shutdown boards from the offsite power source. For the scenario in which an accident occurs in one unit and a concurrent shutdown of the second unit occurs without offsite power, TVA stated that preoperational testing for WBN Unit 2 will validate the diesel response to load sequencing on the Unit 2 emergency diesel generators (EDGs). The staff noted that TVA did not provide a summary of the worst-case EDG loading analysis under this scenario for staff's review. The NRC staff will evaluate the status of this issue and will update the status of the EDG loading and load response in a future SSER. This is Open Item 26 (Appendix HH).

For a scenario in which an accident occurs in one unit and spurious ESF actuation occurs in the other unit without offsite power, TVA stated that each EDG and battery is sized adequately, and all of the design parameters are consistent with the design basis specified in Sections 8.2 and 8.3 of the FSAR. However, for a scenario in which an accident occurs in one unit and spurious ESF actuation occurs in the other unit with offsite power, TVA stated that there is no design requirement, consistent with the FSAR, when supplied from offsite power. TVA also stated that the APS and supporting analysis comply with the requirements of Position C.2.b of RG 1.81. Therefore, TVA has not done an analysis in which an accident occurs in one unit while the spurious actuation of ESF loads occurs in the second unit. The design criteria provided in SRP Section 8.2, Part III.9, issued April 1978, requires the NRC staff to evaluate the capability of the preferred power system for spurious or false accident signals (i.e., should not overload the preferred power source circuits). In its December 6, 2010, supplemental letter, TVA provided additional information stating that TVA performed an analysis with one unit in accident and the spurious ESF actuation in the other unit with offsite power available. This analysis concluded that the CSSTs have adequate capacity to support all ESF loads for one unit in accident and spurious ESF actuation in the other unit. The transient voltage drop due to block starting of the ESF motors does not result in transfer of the loads from the offsite power source to onsite power (diesel generators). Based on this information, the NRC staff concludes that the design of the offsite power system acceptable for a scenario in which an accident occurs in one unit and spurious ESF actuation occurs in the other unit.

To verify compliance with GDC 17, the NRC staff also considered a scenario in which a dual-unit trip results from an abnormal operational occurrence. TVA did not provide any specific analysis to conclude that both offsite and onsite power systems have adequate capacity and capability in this scenario. In its December 6, 2010, supplemental letter, TVA stated that a separate analysis was not performed to verify loading under this scenario because the loading for a dual unit trip is enveloped by the analysis for a spurious accident signal on one unit with an accident on the other unit. Also, TVA stated that the diesel generator loading analysis is based on worst case accident loading on the diesel generators, which bounds a dual unit trip. Based on this information, the NRC staff concludes that the design of the onsite power system acceptable for a scenario in which a dual-unit trip results from an abnormal operational occurrence. The adequacy offsite power system for this scenario is discussed in Section 8.2.2.

In response to a staff question regarding how it had reviewed and incorporated industry and WBN Unit 1 operating experience (OE), as well as NRC generic communications, into the electrical design, maintenance, surveillance testing, and future operations for WBN Unit 2, TVA stated that the electrical power system is under the control of the WBN Unit 1 staff. The TVA operating unit staff reviews OE and NRC generic communications in accordance with the TVA Operating Experience Program (SPP-3.9). Under this program, external and internal OE, as well as generic communication, is reviewed and actions developed. The NRC staff concludes that this is acceptable because TVA considers both industry and plant-specific OE in WBN operations.

The NRC staff reviewed the FSAR for this section against the relevant NRC regulations, guidance in SRP Section 8.1, and applicable RGs and, except for the open item discussed above, concludes that TVA is in compliance with the relevant NRC regulations.

Before issuing an operating license, the NRC staff intends to conduct an onsite review of the installation and arrangement of electrical equipment and cables, confirmatory electric drawings, and verification of test results for the purpose of confirming the adequacy of the design and proper implementation of the design criteria. The NRC will address any issues identified during the onsite review in a supplement to the SER.

8.2 Offsite Electric Power System

The offsite electric power system consists of an alternating current (ac) power system supplied from the power grid and is the preferred power supply. The safety function of the offsite power system, assuming the onsite power system is not functioning, is to provide sufficient capacity and capability to ensure that the structures, systems, and components (SSCs) important to safety perform as intended. The NRC staff reviewed FSAR Amendment 95, Section 8.2, "Offsite Electric Power System," to determine whether the offsite power system satisfies the requirements of GDC 5, 17, and 18, "Inspection and Testing of Electric Power Systems," and will perform its design function during all plant operating and accident conditions.

8.2.1 Compliance with General Design Criterion 5

The NRC staff previously concluded in the SER that TVA met the requirements of GDC 5 with respect to the sharing of circuits in the preferred power system. The two offsite preferred power circuits at WBN are shared between Units 1 and 2. In accordance with GDC 5, it must be demonstrated that this sharing of circuits will not significantly impair the ability of Class 1E power systems to perform their safety functions. In accordance with Section 8.1.1 of IEEE Standard 308-1974, which was endorsed in RG 1.32, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants," at a minimum, the capacity of each preferred circuit must be sufficient to operate the ESFs for a design-basis accident (DBA) on one unit, as well as those systems required for the concurrent safe shutdown of the remaining unit.

TVA documented in FSAR Section 8.2.2 that each station service transformer has sufficient capacity to supply the essential safety auxiliaries of one unit under loss-of-coolant accident (LOCA) conditions, in addition to the power required for shutdown of the nonaccident unit. This capacity meets the above-stated position and, therefore, is acceptable. By letter dated October 9, 1981 (ADAMS Accession No. ML073521461), TVA subsequently proposed a system design change. The design change added two service transformers to improve the capability of the offsite preferred circuits. TVA stated that each of these service transformers can supply the Class 1E power system for both units, with one unit responding to a DBA and the other unit in a concurrent full-load rejection, without exceeding the transformers' self-cooling power ratings. The station service transformer primary windings are rated at 33/44/55 megavolt-ampere (MVA) OA/FOA/FOA (two-stage, self-cooled, and assisted by forced air and forced oil) and the secondary winding of each transformer is rated at 24/32/40 MVA OA/FOA/FOA. The FSAR also states that the calculated loads for CSSTs A, B, C, and D are well below winding ratings for all acceptable conditions. The proposed change exceeds the capacity requirements of the above-stated position.

In Section 8.2.1.2 of FSAR Amendments 95 and 100, TVA documented that FSAR Section 8.2.1.8 states that CSST A or B (but not both simultaneously) may be used as an immediate or delayed source replacement for CSST D or C, respectively, through the 6.9-kV shutdown board maintenance supply breakers. For an acceptable range of 161-kV grid conditions, either offsite power circuit can start and supply all electrical equipment that would be supplied from the Class 1E distribution systems for both a DBA in one unit and the simultaneous orderly shutdown of the other unit, as well as a simultaneous single worst-case transmission system contingency. For this event, transformers C or D would be operating within its OA rating, transformers A or B would be operating within its forced air (FA) rating, and adequate voltage would be supplied to the safety-related buses. However, FSAR Section 8.2.2, "Analysis," states that each 161-kV circuit and CSSTs C and D have sufficient capacity and adequate voltage to supply the essential safety auxiliaries of a unit under LOCA conditions concurrent with a simultaneous worst-case single transmission system contingency. TVA should clarify that the analysis considered the second unit. In its letter dated December 6, 2010, TVA stated that amendment 103 to the Unit 2 FSAR will revise the first paragraph of Section 8.2.2 to replace "Each 161kV circuit and CSSTs C and D have sufficient capacity and adequate voltage to supply the essential safety auxiliaries of a unit under loss of coolant accident conditions concurrent with a simultaneous worst-case single transmission system contingency," with "Each 161-kV circuit and CSSTs C and D have sufficient capacity and adequate voltage to supply the essential safety auxiliaries of a unit under loss-of coolant accident (LOCA) conditions and the other unit in concurrent orderly shutdown with a simultaneous worst-case single transmission system contingency." The staff finds the response acceptable since TVA clarified that the analysis considered the second unit's loading requirements to determine the capacity of offsite source.

TVA submitted selective excerpts from its calculation WBN-EEB-EDQ000-999-2007-0002, "AC Power Systems Analyses," which evaluated plant loading conditions. In this calculation, TVA assumed that the existing rating of CSSTs A and B will be upgraded from 57/76 MVA OA/FA to a design rating of 95 MVA (primary winding) and 60 MVA (secondary and tertiary windings) by retrofitting with an additional cooling system. TVA should clarify whether the proposed change is essential to provide adequate capacity for the shared station service transformers to support plant loads for all postulated conditions and should provide to the NRC staff a schedule for completing this modification. Specifically, TVA needs to verify that each service transformer can supply the Class 1E power system for both units under all postulated design conditions, including (1) a DBA with single failure on one unit and a spurious accident signal with full-load rejection on the other unit and (2) a dual-unit shutdown resulting from an abnormal operating occurrence. In its letter dated December 6, 2010, TVA stated "Based on the auxiliary power system analysis, TVA concluded that the upgrade of CSSTs A and B to 95 MVA (primary winding) and 60 MVA (secondary and tertiary winding) is not required for two unit operation. Additionally, CSSTs A and B will not simultaneously be credited as an independent source of offsite power for the Class 1E system. Therefore, only one train shall be transferred to CSSTs A and B at any given time. Additionally, CSSTs A or B cannot be credited as an offsite source if all Balance of Plant station loads are supplied by one CSST (i.e., B or A CSST out of service). Analysis has been performed that verified the adequacy of CSSTs A and B to support shutdown of both units (one unit in accident and other unit in orderly shutdown or with spurious ESF loads actuation) with one train of Class 1E system transferred." TVA has not evaluated the capability of the CSSTs for a dual-unit shutdown resulting from an abnormal operating occurrence. This is discussed in section 8.2.2 as Open Item 27 (Appendix HH) discussed in section 8.2.2. Pending resolution of the open item, the staff concludes that design of WBN Unit 2 meets intent of GDC 5.

8.2.2 Compliance with General Design Criteria 17 and 18

During its previous review in support of the operation of WBN Unit 1, the NRC staff concluded that TVA met the requirements of GDC 17 with respect to the offsite power systems having the (1) capacity and capability to permit functioning of SSCs important to safety; (2) provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit or loss of power from the onsite electric power supplies; (3) physical independence of circuits; and (4) availability of circuits.

The WBN facility is interconnected to the electric grid system through six 161-kV transmission lines that terminate in an existing 161-kV switchyard (i.e., the Watts Bar Hydro Plant switchyard), located about 1.5 miles from the plant. The 161-kV lines enter the switchyard by way of a number of physically separate and independent rights of way. In addition, five hydro-electric generators and four steam-driven generators terminate at the switchyard.

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

Two independent, immediate-access circuits provide offsite power from the switchyard to the onsite Class 1E distribution system. Each of the two circuits is routed from the switchyard through a 161-kV transmission line and a 161-kV-to-6.9-kV transformer (CSST) to the onsite Class 1E distribution system.

The onsite Class 1E distribution system consists, in part, of two redundant and independent 6.9-kV buses, each capable of being fed from either of the above-described offsite circuits. One of the offsite circuits is the normal supply for one of the 6.9-kV buses, while the other offsite circuit is the normal supply for the other 6.9-kV bus. Consequently, each 6.9-kV bus can also be fed by the other (or alternate) offsite circuit. Offsite power is normally supplied to the onsite distribution system from the plant's main generator through a 22.5-kV-to-6.9-kV transformer (unit station service transformer) and 6.9-kV switchgear (unit startup board) to the onsite Class 1E distribution system. For any unit generator trip, offsite power is automatically transferred from the normal supply to the two preferred offsite circuits.

The results of TVA's grid stability analysis indicate that loss of the largest capacity generator supplying power to the grid, loss of the largest load from the grid, loss of the most critical transmission line, or loss of both WBN units themselves will not cause grid instability.

The staff has previously concluded that TVA met the requirements of GDC 18 with respect to the design of the offsite power system to permit periodic inspection and testing to show (1) the operability and functional performance of the components of the circuits, (2) the operability of the offsite power system as a whole system, and (3) the full operation sequence that brings the system into operation under conditions as close to design as practical.

Transfers between the normal and alternate offsite circuits at the 6.9-kV buses are executed manually or automatically. To avoid possible plant transients that could trip the unit, testing of these transfers is done only when the unit is shut down. In addition, provisions in the design of the balance-of-plant load-shedding circuits allow for testing of the transfer from the main generator supply to the offsite supplies while maintaining the load-shedding capability of the circuits.

The NRC staff reviewed Section 8.2, "Offsite AC Power System," in FSAR Amendment 95. Based on its review, the staff requested additional information in a letter dated July 7, 2010 (ADAMS Accession No. ML101620047), pertaining to FSAR Section 8.2. TVA provided responses in its letter dated July 31, 2010.

As a result of its review of FSAR Section 8.2, the NRC staff focused on how TVA met the requirements of GDC 17 and 18 with respect to the design of the offsite circuits for the startup of WBN Unit 2. Specifically, the staff wanted to ensure the capacity and capability of the offsite circuits to permit functioning of the station safety systems given dual-unit operation. In addition, the staff wanted to ensure that a loss of any one ac supply (offsite or onsite) would not result in the loss of the remaining sources, given dual-unit operation.

FSAR Section 8.2.1.2 describes the CCSTs and states that the calculated loading of the CSSTs is well below their winding ratings for all conditions. FSAR Position C2, on page 8.1-13, states, "The shared safety systems are designed so that one load group (Train 1A & 2A or Train 1B & 2B) can mitigate a design basis accident in one unit and accomplish an orderly shutdown of the other unit." These CSSTs are shared between WBN Units 1 and 2. In view of the Unit 2 and the Unit 1 loads being applied to the CSSTs, the NRC staff requested that TVA provide a summary of the calculations and analyses that detailed the loading for both units, or that added loads from WBN Unit 2 to the existing loads of Unit 1, including the design margin in the CSSTs assuming a DBA in one unit with a concurrent safe shutdown of the other unit.

The NRC staff reviewed TVA's response to confirm that the calculations assumed conditions consistent with an accident in one unit concurrent with a safe shutdown of the other unit, while supplied by offsite power. TVA's response confirmed that all operational configurations for offsite power supply to the units yield loadings well within the rating of the transformer, with design margins from 10 percent to as high as 48 percent. Specifically, CSSTs C and D (the normal preferred offsite circuits) have design margins of 48 percent and 32 percent, respectively. The staff finds this design acceptable. However, TVA should provide a summary of similar margin studies based on scenarios described in Section 8.1 above for CSSTs A, B, C, and D. In its December 6, 2010, letter, TVA provided additional information regarding transformer loadings. TVA stated that the loading for a dual-unit trip is slightly less than the loading for one unit in an accident and a spurious accident signal in the other unit. However, TVA did not provide a summary of the analysis for staff's review. TVA should provide a summary of similar margin studies based on a dual-unit trip as a result of an abnormal operational occurrence and an accident in one unit concurrent with a spurious ESF actuation. These should be based on the completed analysis for uprating CSSTs A and B. This is Open Item 27 (Appendix HH).

In FSAR Section 8.2.1, TVA states that, to provide a stable voltage, CSSTs C and D have automatic high-speed load tap changers (LTCs) on each secondary winding, which adjust voltage based on the normally connected shutdown boards. The NRC staff requested that TVA provide a summary of the calculations and analyses that detail the plant load flow and voltage studies and operations of the LTC units, including a detailed discussion of the control voltage setting, the voltage control band, and the time delays for LTC operation. The staff reviewed TVA's response to confirm that the calculations assumed conditions consistent with an accident in one unit concurrent with a safe shutdown of the other unit while supplied by offsite power. In its response, TVA demonstrated that the LTC controls were set to regulate voltage at the Class 1E 6.9-kV buses between 101.6 percent and 103.4 percent, with an initial LTC response delay of 2 seconds followed by a step delay of 1 second per step (1.25 percent voltage per step). Based on the analysis with bounding loading conditions, safety-related systems can start and run without actuating the DVRs. (Section 8.3.1.2 of this report evaluates the bases for relay settings). TVA should provide to the staff a detailed discussion showing that the LTC is able to maintain the 6.9-kV bus voltage control band given the normal and postcontingency transmission operating voltage band, bounding voltage drop on the grid, and plant conditions. This is Open Item 28 (Appendix HH).

FSAR Section 8.2.1 describes the 161-kV preferred offsite power supply from the Watts Bar Hydro Plant switchyard for dual-unit operation. The FSAR states that completed transmission system studies demonstrate that one 161-kV line and one CSST are adequate and capable of starting and running all required safety-related loads in the event of a DBA in WBN Unit 1 with no fuel loaded into WBN Unit 2. Since WBN Unit 2 loads will be supplied from the same 161-kV preferred power supply, the NRC staff requested information with regard to the adequacy and capability of the 161-kV preferred offsite power supply for dual-unit operation (considering a DBA in one unit with a concurrent safe shutdown of the other unit).

The NRC staff requested that TVA provide a detailed description of all such transmission system grid conditions and the operating characteristics of the offsite power supply at the Watts Bar Hydro Plant (for dual-unit operation), including the operating voltage range, postcontingency voltage drops (including bounding values and post-unit trip values), and operating frequency range. In addition, the staff requested that TVA provide the design operating voltage range of the shutdown boards (minimum and maximum voltage) and information regarding how low the Watts Bar Hydro Station voltage could drop (assuming operation of the LTCs) while still supplying the worst-case shutdown board loads at the minimum design voltage of the shutdown boards. The staff requested that the summary of the grid studies address dual-unit operation, the transmission network interface available fault current changes, and the impact on the switchyard and plant switchgear and cabling.

In its December 6, 2010, letter, TVA stated that the grid stability analyses addressed the loss of the largest electric supply to the grid, loss of the largest load from the grid, loss of the most critical transmission line, loss of both units, all of which did not result in grid instability. NRC staff considers the stability analysis portion of the grid studies acceptable. However, TVA did not provide information about the operating characteristics of the offsite power supply and other information as discussed above. This is Open Item 29 (Appendix HH).

Based on a review of the current information, TVA continues to meet the requirements of GDC 18 with respect to the offsite ac power system. The offsite power system is designed to be testable during station operation, as well as during those intervals when the station is shut down.

8.2.4 Evaluation Findings

The NRC staff reviewed the offsite power system for WBN Unit 2 as described in FSAR Section 8.2, including the single-line diagrams, station layout drawings, schematic diagrams, and descriptive information. The staff concluded that the offsite power system conforms to the requirements of GDC 17 and 18 and is, therefore, acceptable, pending resolution of the open items noted above.

8.3 Onsite Power Systems

The safety function of the onsite power system, assuming that the offsite power system is not functioning, is to provide sufficient capacity and capability to ensure that the SSCs important to safety can perform their safety function. The objective of the NRC staff's review is to determine that the onsite power system satisfies the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," GDC 4, "Environmental and Dynamic Effects Design Bases," GDC 5, GDC 17, GDC 18, and GDC 50, "Containment Design Basis," and will perform its intended safety function during all plant operating and accident design conditions.

The onsite power system consists of an ac power system and a direct current (dc) power system. Compliance with GDC 2, 4, 5, 18, and 50 relates to both the ac and dc systems. Section 8.3.3.2 of this evaluation documents the NRC staff's evaluation of the ac and dc power system for compliance with GDC 5. Compliance with GDC 2, 4, 18, and 50 also relates to both the ac and dc systems and is documented in Section 8.3.3 of this evaluation. Section 8.3.1 evaluates compliance with GDC 17 for the ac system, and Section 8.3.2 documents compliance with GDC 17 for the dc system.

8.3.1 Onsite Alternating Current Power System Compliance with General Design Criterion 17

In its SER for the operating license for WBN Unit 1, the NRC staff concluded that TVA met the requirements of GDC 17 with respect to the onsite ac system having the (1) capacity and capability to permit functioning of SSCs important to safety; (2) independence, redundancy, and testability of the SSCs to perform their safety functions, while assuming a single failure; and (3) provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit or the loss of power from the transmission network. The NRC staff evaluated the WBN Unit 2 onsite ac system to ensure compliance with GDC 17.

The ac power system for each WBN unit comprises two redundant and independent distribution system divisions, each powered by one of two redundant standby EDGs or by either of two offsite circuits. Each distribution system division includes a 6.9-kV shutdown board, 480-V shutdown boards, 480-V motor control centers, and distribution system cables to accommodate the voltage requirements of the safety loads. The safety loads for each unit are distributed among these two electrical divisions in such a manner that the operation of any one of the two is all that is required to meet minimum safety requirements. No automatic transfers of electrical loads occur between these two redundant electrical divisions.

Each of the EDGs and selected associated auxiliaries is located in a separate compartment within a seismic Category 1 structure. Each EDG has a continuous rating, 2,000-hour rating, and 2-hour rating of 4,400 kilowatt (kW), 4,750 kW, and 5,225 kW, respectively. Each EDG will be automatically started by an undervoltage signal from its respective 6.9-kV emergency bus or by a safety injection signal. Upon loss of offsite power (LOOP), the 6.9-kV emergency buses will be automatically isolated from all supply sources, and all associated 6.9-kV motor loads will be tripped off. Further, all EDGs will be connected automatically to their respective emergency bus, and, under accident conditions or LOOP, the safety loads will be automatically connected in a predetermined sequence to their respective EDG.

The staff reviewed Section 8.3.1, "Onsite AC Power System," in FSAR Amendment 95. Based on its review, the staff requested additional information in a letter dated July 12, 2010. TVA provided responses in its letter dated July 31, 2010.

As guidance in its review, the NRC staff used the acceptance criteria described in SRP Section 8.3.1, "AC Power Systems (Onsite)," issued in July 1981. According to the acceptance criteria, the onsite ac power system is acceptable when the integrated design complies with GDC 2, 4, 5, 17, 18, and 50, as described in Appendix A to

10 CFR Part 50. While reviewing FSAR Section 8.3.1, the staff focused on how TVA met the requirements of GDC 17 for the startup of WBN Unit 2.

8.3.1.2 Low and Degraded Voltage Conditions

In response to the NRC staff's request for additional information regarding the DVRs, TVA stated that the degraded voltage analysis evaluates the capability to individually start and run Class 1E motors at steady-state conditions. This analysis ensures that all motors have adequate starting voltage at the upper boundary of the DVR setpoint (6,672 V) and adequate running voltage at the lower boundary of the DVR setpoint (6,555 V). The upper boundary represents the lowest voltage that guarantees offsite power supply recovery following a transient from a design-basis event. TVA revised the upper boundary setpoint to 6,681 V; therefore, the analysis performed with the value of 6,672 V is conservative.

TVA should confirm that all safety-related equipment (in addition to the Class 1E motors) will have adequate starting and running voltage at the most limiting safety-related components (such as motor-operated valves (MOVs), contactors, solenoid valves or relays) at the DVR setpoint dropout setting. TVA should also confirm that (1) the motor-starting transient studies are based on the dropout voltage value of DVR and time delay, (2) the steady-state voltage drop studies are carried out by maximizing running loads on the Class 1E distribution system (bounding combination of safety systems loads), with the voltage at 6.9-kV Class 1E buses (monitored by the DVRs) at or just above the DVR dropout setting, and (3) the DVR settings do not credit any equipment operation (such as LTC transformers) upstream of the 6.9-kV Class 1E buses. TVA should also confirm that the final technical specifications (TSs) are properly derived from these analytical values for the degraded voltage settings. This is Open Item 30 (Appendix HH).

Regarding the loss of voltage (LOV) relays, TVA stated in its application that the LOV relay voltage setpoint lower limit is selected by evaluating the operation of the APS under steady-state (running) conditions, with the 6.9-kV shutdown board voltage as low as possible and all connected safety-related motor loads above their stall voltage (greater than 70.7 percent of rated motor voltage for National Electrical Manufacturers Association Design B motors). The lower limit must also be greater than the 6.9-kV shutdown board voltage that is equivalent to having the lowest switchyard voltage that could be sustained without instability or collapse. The lowest boundary of the LOV voltage relay time delay should be long enough to ride through short circuits and other short-time system transients (e.g., lightning strikes, switching transients), taking into account the total sensing and clearing times for the above type of events. The time-delay upper boundary should be less than the safety analysis time allowed for LOV detection. Based on this criterion, the LOV relay lower and upper limits were calculated to be 5,968 V and 6,060 V, respectively. The associated time-delay relay lower and upper limits are at 0.4 seconds and 1.14 seconds, respectively.

The NRC staff notes that the LOV relay lower limit time delay of 0.4 seconds may be too short for the transmission line protection system to clear a short-circuit fault. The staff asked TVA to confirm that the longest time setting for the 161-kV transmission line protection to clear a short-circuit fault, such as a time setting in a second or third zone of distance protection, will not cause the actuation of the LOV relay. In its letter dated December 6, 2010, TVA stated that for the offsite study, it evaluates clearing time based on Zone 1 and 2 protection. For the 161-kV system, this protection has a nominal clearing time of 5 cycles or less. A post-fault 161-kV voltage recovery analysis was performed as part of the Grid Voltage Study. This analysis was performed with one line removed from service pre-event and then by applying a 3-phase fault on another line (there are six 161kV lines which terminate at WBN plant). After T = 0.0833 seconds (5 cycles), the fault is cleared. Based on the worst-case (slowest) voltage recovery, the voltage recovers to 0.9 per unit (90 percent) or 145 kV after about 0.3 seconds from the start of the fault. Based on this worst-case voltage recovery time, it is concluded that 0.4 second time delay provides ample margin for the loss of voltage relay setting. Based on the clarification provided by TVA, the staff finds the LOV relay lower limit time delay of 0.4 seconds is acceptable.

8.3.1.10 Fast Transfer Scheme at Shutdown Boards

In response to NRC staff questions regarding the fast transfer scheme, TVA stated in its letter dated July 31, 2010, that the 6.9-kV shutdown boards have been provided with an automatic fast bus transfer scheme. Because all WBN Unit 1 and Unit 2 shutdown boards are currently required to support Unit 1 operation, the existing bus transfer scheme is operational and is not affected by Unit 2. Therefore, the NRC staff considers the fast transfer scheme to be acceptable.

8.3.1.11 Automatic Sequencing of Loads

In response to staff questions regarding the automatic sequencing of loads, TVA stated in its letter dated July 31, 2010, that the load sequencer consists of discrete relays, which are part of the individual load's breaker control circuit. TVA stated that the loads are shed upon LOV and reconnected upon restoration of an EDG through timing of discrete relays. The NRC staff notes that the FSAR also states that the automatic sequencing logic is designed for a scenario in which the accident signal occurs before the loss of preferred (offsite) power. The staff notes that this scenario is equivalent to a LOCA followed by a LOOP, which would require resequencing of the loads.

The staff asked TVA to evaluate the resequencing of loads, with time delays involved, in the scenario of a LOCA followed by a delayed LOOP and to ensure that all loads will be sequenced within the time assumed in the accident analysis. In its December 6, 2010, letter, TVA stated that a LOCA followed by a delayed LOOP is not a design-basis event for WBN. However, the load sequencing circuitry has features that minimize the impact of this event on the onsite power system. The design-basis event as described in Section 8.3.1.1 of the WBN Unit 2 FSAR is "A loss of offsite power coincident with a safety injection signal." A safety injection signal received during the course of non-accident shutdown loading sequence will cause actions described below:

- Loads already sequentially connected which are not required for an accident will be disconnected.
- Loads already sequentially connected which are required for an accident will remain connected.
- Loads awaiting sequential loading that are not required for an accident will not be connected.
- Loads awaiting sequential loading that are required for an accident will either be sequentially loaded as a result of the non-accident loading sequence or will have

their sequential timers reset to time zero. They will then be sequentially loaded in accordance with the accident sequence.

The above loading sequence as explained in the TVA letter is not clear to the staff. Specifically, statements such as (1) the load sequencing circuitry has features which minimize the impact of this event on the onsite power system, and (2) a safety injection signal received during the course of non-accident shutdown loading sequence will cause actions. TVA should clarify whether the existing statements in FSAR regarding automatic sequencing logic are correct. If the FSAR description is correct, TVA should explain how the EDG and logic sequencing circuitry will respond to a LOCA followed by a LOOP scenario. This is Open Item 31 (Appendix HH).

8.3.1.12 Bus Ratings and Connected Loads

In response to staff questions regarding the bus ratings and connected loads, TVA stated in its letter dated July 31, 2010, that FSAR Tables 8.3-4 through 8.3-7 describe the board/bus rating in kilovolt-amperes (kVA) and do not represent the connected load or the maximum demand. This kVA rating is calculated by the equation kVA = $\sqrt{3}$ VI, where V and I are the rated voltage and the rated current, respectively. The 6.9-kV and 480-V boards on WBN Unit 2 have the same ratings as the Unit 1 boards. The APS analysis verified that the loading on all safety-related boards was within their rating, and no overloading has been identified. The loading on all boards, when powered from the standby onsite power system (EDGs), is enveloped by the loading in the APS calculation.

The NRC staff concludes that TVA's response adequately addresses the board/bus ratings. However, TVA should correct the wording in FSAR Section 8.3.1.1, which states that the connected load and the maximum demand are shown in FSAR Tables 8.3-4 through 8.3.7. This position is contrary to the statement in TVA's letter dated July 31, 2010, that FSAR Tables 8.3-4 through 8.3-7 describe the board/bus rating in kVA and do not represent the connected load or the maximum demand. In its letter dated December 6, 2010, TVA stated that Amendment 103 to the Unit 2 FSAR will revise the Equipment Capacities portion of Section 8.3.1.1 to match the information in Tables 8.3-4 through 8.3-.7. The staff finds the TVA response acceptable.

8.3.1.13 Standby Diesel Generator Operation

In response to an NRC staff question regarding the explanation of "appropriate alignment" of standby diesel generator operation in the FSAR, TVA explained in its letter dated July 31, 2010, the concept of "appropriate alignment" as follows:

> During testing of the [EDG], should a LOOP and accident occur, an accident signal will trip the [EDG] feeder breaker. Tripping the [EDG] breaker will automatically place the [EDG] in asynchronous mode of operation. As soon as the offsite power supply breaker to the 6.9kV shutdown board is tripped and the undervoltage load stripping relays operate, the [EDG] feeder breaker to the board will close and load sequencing logic will be initiated to load the accident loads.

The NRC staff notes that, in the above explanation of "appropriate alignment," TVA considered the scenario a LOOP and an accident, while in the FSAR, the term "appropriate alignment" refers to a LOOP only. The staff asked TVA to resolve this discrepancy and revise the FSAR to describe the appropriate alignment based on the actual design of the standby diesel generator operation. In its December 6, 2010, letter TVA stated the explanation provided in the FSAR is correct. TVA also stated that the words "and an accident" were erroneously added and should be considered as deleted from the previous response. The staff finds the TVA response acceptable.

8.3.1.14 Adequacy of Diesel Generator Capacity

In response to NRC staff questions regarding the EDG capacity, in its letter dated July 31, 2010, TVA provided maximum EDG loadings under steady-state and transient conditions, as well as margins with respect to various EDG ratings. TVA stated that the maximum steady-state ratings of the EDG are 6,050 kW (0 to 2 hours), 5,500 kW (2 hours to end), and 4,400 kW (2 hours to end, excluding the load of the spare 125 V DC charger and with all pumps operating). These ratings provided minimum margins of 19.9 percent, 11.8 percent, and 4.97 percent, respectively, based on the worst-case accident scenario. The NRC staff concludes that the EDG steady-state ratings are adequate, because there is sufficient margin for the ratings.

TVA stated that the maximum transient ratings of the EDG are 4,785 kW (0 to 180 seconds) and 5,073 kW (180 seconds to end), and the maximum step load increase rating is 8,000 kVA (0 second to end). These ratings provided minimum margins of 17.7 percent, 6.2 percent, and 45 percent, respectively, based on the worst-case accident scenario. The staff requested TVA to provide the basis for the maximum transient ratings of 4,785 kW (0 to 180 seconds) and 5,073 kW (180 seconds to end) and the maximum step load increase rating of 8,000 kVA (0 second to end). In its letter dated December 6, 2010, TVA provided the basis for the above transient ratings, as follows:

> The maximum transient rating of the diesel generator (DG) set at 900 rpm, 900C intake air, elevation less than 10,000 ft and a guaranteed efficiency of 96.6% is calculated as follows:
>
> 2000 hr/yr: 6640 (BHP-tandem) x 0.746 KW/HP) x 0.966 = 4785 KW
> 30 min/yr: 7040 (BHP-tandem) x 0.746 KW/HP) x 0.966 = 5073 KW
>
> This analysis was performed by TVA and concurred by the DG supplier, MKW Power Systems, Inc.
>
> Similarly, the maximum step load increase was calculated based on maximum voltage dip as per the guaranteed contract data. The maximum kVA load step increase without exceeding the minimum voltage limit prescribed by RG 1.9 (75% nominal) was calculated as 8700 kVA. However, to be conservative, a load step increase value of 8000 kVA is used. The maximum load step increase as per the DG Loading Analysis is only approx 4400 KVa; therefore, significant margin is available between the actual load step increase and the allowable value of 8000 kVA.

Regarding the impact of off-normal frequency and voltage on the EDG loading, TVA stated the following in its letter dated July 31, 2010:

> TVA has added administrative limits to the plant operating procedures for both [EDG] voltage and speed range. These administrative limits are so tight that there would be negligible impact on the [EDG] loading due to off-normal frequency and voltage.

TVA should provide to the NRC staff the details of the administrative limits of EDG voltage and speed range, along with the basis for its conclusion that the impact is negligible. TVA should also describe how it accounts for the administrative limits in the TS surveillance requirements for EDG voltage and frequency. This is Open Item 32 (Appendix HH).

8.3.1.15 Underground Cables

In response to NRC staff questions, TVA stated the following in its letter dated July 31, 2010:

> Unit 2 underground cables are designed for submerged or flooded condition. All Unit 2 safety related underground cables that were routed in duct banks to the remote facilities like intake pumping station and diesel generator buildings were turned over to Unit 1 when Unit 1 was licensed in 1996 and have since been in service supporting Unit 1.

The NRC staff requested that TVA provide documentation that the underground cables are designed for submergence or that measures are provided in the raceways system that are adequate to prevent submergence. In a letter dated December 6, 2010, TVA stated that the discussion in Section 8.3.1.2.3 of the WBN Unit 2 FSAR pertains to submergence of the cables for the design-basis flood. It is not intended to address the issues associated with long-term submergence of energized cables such as "treeing." At WBN, underground cables are installed in seismically qualified concrete enclosed ductbanks. The safety-related manholes are equipped with sump pumps to remove any ground water and have alarms to monitor water level or improper operation of the pumps. The medium voltage cables are evaluated by a testing program which is based on Very Low Frequency (VLF) testing. The staff concludes that the TVA response is acceptable, because the underground cables are designed for submergence or measures are provided in the raceways system that are adequate to prevent submergence.

8.3.2 *Onsite Direct Current System Compliance with General Design Criterion 17*

In its SER for the WBN Unit 1 operating license, the NRC staff concluded that TVA met the requirements of GDC 17 with respect to the onsite dc system having the (1) capacity and capability to permit functioning of SSCs important to safety; (2) independence, redundancy, and testability of the SSCs to perform their safety functions, while assuming a single failure; and (3) provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit or the loss of power from the transmission network.

The NRC staff evaluated the WBN Unit 2 onsite dc system to ensure compliance with GDC 17.

The dc power system for each unit comprises two redundant and independent distribution systems, each powered by a 125-V battery or a battery charger. Each distribution system division includes a 125-V dc distribution board and system cables to accommodate the dc control power requirements of the safety loads. Each division supplies control power to its associated ac distribution system and loads. Each division also supplies power to two 120-V ac vital instrumentation buses and control inverters. One of the two inverters supplies vital loads of the opposite unit so that there is sharing of dc and ac supplies between units for vital instrumentation and control power. The dc power system also comprises a separate diesel generator battery system associated with each of the EDGs.

The 125-V dc vital power system is composed of four redundant channels (designated as channels I, II, III, and IV) and consists of four lead-acid-calcium batteries, six battery chargers (including two spare chargers), four distribution boards, battery racks, and the required cabling, instrumentation, and protective features. Each channel is electrically and physically independent from the equipment of all other channels, so that a single failure in one channel will not cause a failure in another channel. Each channel consists of a battery charger that supplies normal dc power; a battery for emergency dc power; and a battery board, which facilitates load grouping and provides circuit protection. These four channels provide emergency power to the 120-V ac vital power system, which furnishes control power to the reactor protection system. No automatic connections are used between the four redundant channels.

Battery boards I, II, III, and IV have a charger normally connected to them and have manual access to a spare (backup) charger for use upon loss of the normal charger. Additionally, battery boards I, II, III, and IV have manual access to the fifth vital battery system. The fifth 125-V dc vital battery is intended to serve as a replacement for any one of the four 125-V dc vital batteries during testing, maintenance, and outages with no loss of system reliability under any mode of operation.

8.3.2.3 Availability of the Battery Supplies to Vital Instrument Buses

On the basis of its review of FSAR Amendments 95 and 100, the staff requested that TVA provide a summary of the results of calculations used for determining the size of the inverters, battery chargers, batteries, and fuses.

In its letter dated July 31, 2010, TVA provided summaries for the above components. The NRC staff reviewed the summaries and concludes that TVA has demonstrated that it has adequately sized the inverters, battery chargers, and fuses associated with the dc power system at WBN Unit 2.

In its letter dated July 31, 2010, TVA stated the following:

> load shedding on vital inverters 1-I, 2-I, 1-II and 2-II is required to be completed within 30 minutes in accordance with Unit 2 FSAR Section 8.3.2.1.1, Load Time of Application, to conclude that the selected battery is adequate for the four (4) hour SBO [station blackout] coping duration

In its December 6, 2010, letter, TVA provided additional clarifying information on why load shedding is necessary to cope with an SBO event. Based on its review of this additional information, the NRC staff finds that the shedding of nonessential loads to cope with an SBO event is in accordance with Section 3.2.6 of RG 1.155, "Station Blackout," and is also consistent with considerations recognized in IEEE Standard 946, "IEEE Recommended Practice for the Design of DC Auxiliary Power Systems for Generating Stations," Section 5.2. Thus, the NRC staff finds that with the dual-unit load shedding, the batteries are sized adequately to respond to a 4-hour coping duration for an SBO event.

In Attachment 9, "125 V DC Vital Battery System Analysis," of its letter dated July 31, 2010, TVA applied the following assumptions in Section 5.13 of the dc system analysis:

- DCN (later)—Removal/abandonment of Reciprocating Charging Pump 2MTR-62-101, supplied from 480V SHDN BD2BI-B, Compt. 3B
- DCN (later)—Cable modifications for U2 AFWP Turbine Trip and Throttle Valve and Turbine Controls

TVA stated that the design change notices (DCNs) are required or anticipated for completion of WBN Unit 2, and that these were unverified assumptions used in its analysis of the 125-V dc vital battery system. Verification of the completion of these DCNs must be provided to the NRC staff before issuance of the operating license. This is Open Item 33 (Appendix HH).

The NRC staff requested that TVA clarify whether the battery service tests verify the design duty cycles for DBAs, station blackout (SBO), and scenarios related to Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," to 10 CFR Part 50.

In its letter dated July 31, 2010, TVA stated that the service test duty cycle was determined in a WBN Unit 1/Unit 2 vital battery sizing calculation from a review of battery load duty cycles associated with battery I, II, III, IV, and V for the SBO, LOCA/LOOP, and Appendix R cases. The duty cycle chosen by TVA represents the worst-case duty cycle for an SBO event; in this case, the term "worst-case" means that the battery load case controls the battery size (more positive plates) and voltage (minimum voltage). TVA considered the LOCA/LOOP cases for first-minute loading only because the total duty cycle is much smaller than the SBO duty cycle. The Appendix R cases do not have an impact because the duty cycles are bounded by the SBO. Since the load study case results represent the worst-case loading for the first minute and complete duty cycle for SBO (i.e., 1–240 minutes), the NRC staff considers TVA's battery-load duty-cycle evaluation to be acceptable.

The NRC staff requested that TVA provide the duty-cycle load profile for both the service and modified performance tests to show that the modified performance test completely envelops the service test for each of the vital and EDG system battery's design duty cycles (i.e., DBAs, SBO, and Appendix R). In its letter dated July 31, 2010, TVA stated that, "Due to duty cycle limitation imposed by IEEE Standard 450-1995, Section 5.4, Watts Bar has not developed a modified performance test duty cycle or an implementing procedure." In its December 6, 2010, letter, TVA provided detailed information on the discharge rates for both the service discharge test and the modified performance discharge test. The NRC staff finds that TVA has demonstrated that the modified performance discharge test bounds the service discharge test profile. Based on this information, the NRC staff concludes that the modified performance discharge test is acceptable for use at WBN Unit 2.

During the staff's review of Section 5.4 of Attachment 9 to its letter dated July 31, 2010, TVA addressed the analytical assumption that, "6900 and 480 volt breaker operation for each condition, if automatically actuated by control logic signals, were assumed to operate commencing at 0 second for the unit under accident and after 30 seconds for the non-accident unit for safe shutdown." In its technical justification for the above scenario, TVA stated the following:

> The accident logic is independent and not shared by both units; therefore an accident in one unit will not cause the other unit generator to trip concurrently with the accident unit. Hence assumed starting of safe shutdown of the non-accident unit at 30 Seconds is conservative. In this analysis Unit 1 has been assumed under DBA and Unit 2 as a non-accident unit. Both units' load values are very similar per Appendices A, B, F & G [of TVA's letter dated July 31, 2010], therefore, either Unit 1 or Unit 2 can be under DBA and other non-accident unit can be safely shutdown as discussed in Design Criteria WB-DC-30-27, Section 6.2.1.A (ref. 4.1.2.c) and UFSAR [updated final safety analysis report] Section 8.1.4 (ref. 4.1.2.a).

In its December 6, 2010, letter, TVA provided additional information to clarify the conservatism of the assumptions made in its 125-V dc battery system analysis. TVA stated that the normal alignments for the switchgear dc control feeders ensure that only the ESF loads for one unit are applied to one vital battery. TVA further stated that only one train would be aligned to alternate feeders such that the minimum required ESF loads of the other train would remain independent and would be available in the event of a single failure including the effects of a simultaneous spurious accident signal in the nonaccident unit. The NRC staff finds that TVA's analysis conservatively demonstrates adequate voltage and capacity considering the effects of either normal or alternate switchgear control feeder alignments in service wherein LOOP loads are applied simultaneously for the non-accident unit.

8.3.2.4 Diesel Generator Battery System

Each diesel generator battery system comprises a battery, battery charger, distribution center, and cabling. The diesel generator battery provides control and field flash power when the charger is unavailable. The charger supplies the normal dc loads, maintains the battery in a fully charged condition, and recharges the battery while supplying the required loads, regardless of the status of the plant. The diesel generator dc control power is ungrounded with ground detection instrumentation. The diesel generator battery comprises 58 cells. TVA stated that the diesel generator battery has adequate capacity to support its design function. The diesel generators' 125-V dc control and field flash circuits are supplied from their respective dc distribution panels located in each diesel generator room. Each diesel generator battery charger maintains a floating voltage of approximately 130 V and is capable of maintaining a 135-V equalizing charge. Each diesel generator battery charger has access to normal and alternate ac supply.

FSAR Section 8.3 states that the diesel generator battery chargers have the capacity to continuously supply all steady-state loads and maintain the batteries in the design maximum-charged state or to fully recharge the batteries from the design minimum-discharged state within an acceptable time interval, irrespective of the status of the plant during which these demands occur. In its letter dated July 31, 2010, TVA defined the term "acceptable time interval" as it relates to the diesel generator 125-V dc battery system's battery chargers.

TVA stated that the diesel generator battery chargers are sized to supply all steady-state loads and to fully recharge the batteries from the design minimum-discharged state to the fully charged state in less than 8 hours. This diesel generator battery recharge time compares favorably to the recharge time of 36 hours for 125-V dc vital battery following an SBO and 12 hours following a LOCA. Therefore, TVA judged the period of 8 hours to be an "acceptable time interval." TVA's response is reasonable, and so the NRC staff finds the 8-hour period to be an acceptable time interval.

8.3.2.9 Evaluation Findings

The NRC staff reviewed FSAR Amendments 95 and 100; the supplemental information provided by TVA in its letter dated July 31, 2010; and the staff's original SER for WBN Unit 1 and concludes that the WBN Unit 2 onsite dc system complies with GDC 17, pending resolution of the above open items. This conclusion is based on the following WBN Unit 2 design characteristics:

1. Separate 125-V dc channels supply control power for each of the two ac load groups.

2. The ac power for the battery chargers in each of these dc channels is supplied from the same ac load group for which the dc channel supplies the control power.

3. The Class 1E dc channels, including batteries, chargers, dc distribution boards, and equipment, are physically separate and independent in each unit.

4. Sufficient capacity, capability, independence, redundancy, and testability are provided in the dc channels, ensuring the performance of safety functions assuming a single failure.

In addition, the staff reviewed the original system design requirements for the dc system, including dc system redundancy, separation, capacity, battery room ventilation, loading, test and inspections, and seismic qualification, and concludes that TVA has adequately demonstrated that WBN Unit 2 will meet these original system design requirements.

8.3.3 Common Electrical Features and Requirements

The NRC staff evaluated the common electrical features and requirements of the onsite ac and dc power system, which pertain to distinct aspects of the systems.

8.3.3.1 Compliance with General Design Criteria 2 and 4

In its SER for the operating license for WBN Unit 1, the NRC staff concluded that TVA met the requirements of GDC 2 and 4 with respect to the ability of the SSCs of the onsite ac and dc power system to withstand the effects of natural phenomena (such as earthquakes, tornadoes, hurricanes, and floods), missiles, and environmental conditions associated with normal operation and postulated accidents. The onsite power system and components (1) are located in seismic Category I structures, which provide protection from the effects of tornadoes, tornado missiles, turbine missiles, and external floods; (2) have been given a quality assurance designation of "Class 1E"; (3) have been designated to be seismically and environmentally qualified; and (4) are designed to accommodate or to be protected from the effects of missiles and environmental conditions associated with normal operation and postulated accidents.

The following sections address issues identified during the NRC staff's review of FSAR Amendments 95 and 100.

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

In SER Section 8.3.3.1.1, the NRC staff noted that TVA indicated that Class 1E equipment required to operate after a LOCA will be relocated above the flood level. Class 1E and associated non-Class 1E equipment not required to operate after a LOCA will be either (1) automatically deenergized in the event of a LOCA, (2) disconnected from its power source during normal operation (power lockout), or (3) analyzed to show (given proper operation of overcurrent protection) that the short circuit resulting from equipment submergence will not degrade the Class 1E power systems.

In SER Section 8.3.3.1.1, the NRC staff stated that surveillance requirements for components with power lockout and disconnect provisions will be pursued as part of the TS review. The staff concluded that there is reasonable assurance that Class 1E power systems will not be lost or degraded as a result of the submergence of Class 1E and associated non-Class 1E equipment not required after a LOCA. With regard to the functional aspect of power systems, the staff accepted the resolution concerning submergence of equipment pending incorporation of surveillance requirements for the protective devices in the TS.

SSER 13, issued April 1984, further discussed the above issue. TVA included the protective device testing provisions in the surveillance requirements in the technical requirements manual (TRM) for isolation devices and containment overcurrent protection devices. The NRC staff reviewed those requirements and found them acceptable.

In Table 8.3-28 of FSAR Amendment 95, TVA listed the major electrical equipment that could become submerged following a LOCA. The listed equipment is either automatically deenergized or is not required to function after a LOCA. In Sections 8A and 8B, TVA summarized the analysis of submerged (post-LOCA) electrical equipment powered from the APS and from the instrumentation and control power system. The analysis concluded that submerged electrical equipment will not degrade the 6.9-kV or 480-V Class 1E APSs or Class 1E instrumentation and control power systems.

Therefore, the NRC staff considers the issue of submerged electrical equipment as a result of a LOCA to be resolved.

8.3.3.1.2 Thermal Overload Protective Bypass

In SER Section 8.3.3.1.2, the NRC staff stated that MOVs with thermal overload protection devices for the valve motors are used in safety systems and their auxiliary supporting systems. Operating experience has shown that indiscriminate application of thermal overload protection devices to the motors associated with these valves could result in needless hindrance to successful completion of safety functions. RG 1.106, "Thermal Overload Protection for Electric Motors on Motor-Operated Valves," issued November 1975, recommends bypassing during accident conditions or properly selecting the setpoints for the thermal overloads, supplemented with periodic testing of these devices, as acceptable methods to be implemented in the design of MOVs.

TVA stated in its application that the WBN design complies with the guidance in RG 1.106 and that the design has been changed to include bypass of thermal overload protective devices during an accident. This design change is consistent with RG 1.106 and is, therefore, acceptable.

In FSAR Amendment 95, Section 8.1.5.3, TVA discussed compliance with RGs and IEEE standards. TVA clarified that the WBN Unit 2 design complies with Position C.1(b) of RG 1.106, Revision 1, except for Position C.1(b), which requires bypass of the thermal overload contacts of all safety-related MOVs during accident conditions. TVA stated that it will bypass the thermal overload contacts of all active valves (i.e., valves required to perform a mechanical function after a safety injection signal). Since active valves are the only ones required to change position to shut down the reactor or to mitigate the effects of a design-basis event, they are the only MOVs that require this assurance of position change. TVA calculation WBN-OSG4-095, "Selection Criteria for MOVs Requiring Thermal Overload Bypass," identifies the list of the active MOVs that require thermal overlaod contacts bypass.

The NRC staff concludes that the above clarification by TVA is acceptable, and the issue of thermal overload protective bypass is resolved.

8.3.3.2 Compliance with General Design Criterion 5

In SER Section 8.3.3.2, the NRC staff concluded that TVA met (with some noted exceptions) the requirements of GDC 5 with respect to SSCs of the ac and dc onsite power systems.

The non-Class 1E control power circuits from the vital battery boards to the 6.9-kV common switchgear C and D have redundant protection (breaker and fuse) in the event of a failure. Selective coordination exists between the non-Class 1E and Class 1E circuits that are fed from each of the vital battery boards. Thus, failure of all of the non-Class 1E control power circuits on the vital battery boards will not have any effect on the Class 1E circuits or battery boards. TVA has credited selective coordination between protective devices to ensure adequate protection of safety-related dc systems from failures in non-Class 1E circuits between common circuits or safety and nonsafety-related circuits. For circuits that have short-circuit current above the instantaneous setting of successive devices, selective coordination may not be achievable. TVA should provide to the NRC staff additional clarification regarding coordination in the instantaneous region of the protective devices for circuits that have short-circuit current above the instantaneous setting of successive devices. In its December 6, 2010, letter, TVA stated that the adequacy of selective tripping has been verified to assure protection of safety-related dc systems from failure in the non-Class 1E circuits and common or safety/nonsafety-related circuits. All cascaded fuses were tested for selective coordination with the upstream protective devices.

8.3.3.2.1 Sharing of Direct Current Distribution Systems and Power Supplies between Watts Bar Nuclear Units 1 and 2

The WBN Unit 1 Class 1E dc system supplies Unit 1 buses I and II, and the WBN Unit 2 Class 1E dc system supplies Unit 2 buses III and IV. The Class 1E dc systems are common to both units, and the dc systems are shared in all modes of plant operation.

Section 8.1.5.2 of the FSAR describes compliance with RG 1.81, Revision 1. TVA stated that the design of the 125-V vital dc system, as a minimum, meets the requirements of Position 3 of RG 1.81. In particular, the system is capable of supplying minimum ESF loads and the loads required for attaining a safe and orderly concurrent shutdown of the unit, assuming a single failure and LOOP.

The staff requested TVA clarify the potential consequences of a scenario featuring a spurious accident signal in the nonaccident unit concurrent with an accident in the other unit and a single failure and the capability of the dc system to handle a dual-unit trip as a result of an abnormal operational occurrence. In its letter dated December 6, 2010, TVA stated that "In accordance with Regulatory Guide 1.81, Section C.2.b, a spurious accident is considered to be a single failure; therefore, no additional single failure is assumed. The DC analysis is performed considering the worst case loading for each channel considering both offsite and onsite power is available. This results in the same loading that would be seen with an accident on one unit and a spurious accident signal on the other." TVA has also concluded that the dc system loading due to a dual-unit trip is enveloped by the current battery system analysis. Since the scenario is bounded by the current battery system analysis, the staff finds the response acceptable.

The 120-V ac vital instrument power, powered by the dc system, has four uninterruptible power supply (UPS) units per unit. The UPS units supply power for the four-channel reactor protection system input relays. The relays fail in a safe position (i.e., they actuate a reactor protection system signal upon loss of power); thus, a single failure or a LOOP, or both, does not prevent the safe and orderly shutdown of either unit. Some plant common loads are supplied from WBN Unit 1 channels I and II, and other plant common loads are supplied from WBN Unit 2. TVA indicated that, for all postulated cases, the sharing of common loads does not inhibit the safe shutdown of one unit while the other unit is experiencing an accident. All shared systems are sized to carry all credible combinations of normal and accident loads.

In response to NRC staff questions regarding the capabilities of protective devices to isolate faults in common circuits or nonsafety circuits, or both, which could potentially disable power to safety-related circuits in both units, TVA stated that it reviewed the schematics and wiring diagrams of control circuits supplied by the 120-V vital ac power system to verify selective coordination with upstream protective devices. Also, the vital ac UPS units go into a current-limiting mode during fault conditions. All load output circuit breakers used on the distribution boards are a high-speed hydraulic-magnetic type, which has the unique characteristic of high-speed tripping at low-fault currents. Hence, coordination of protective devices is achievable. The FSAR states that the Unit 1 and Unit 2 vital instrument power boards for each channel are independent and supplied from a unitized UPS or regulated transformer bypass source. Based on the information provided by TVA, the NRC staff concludes that the design of the 120-V vital ac system meets the requirements of GDC 5.

Each EDG is supplied by a diesel generator battery system. Each battery system comprises a battery, battery charger, distribution center, and cabling. The diesel generator battery system is dedicated to each generator and is not shared with other units. The batteries are physically and electrically independent. The battery systems are each located in a different room, and the support systems, such as heating, cooling, and ventilation, are independent of each other. Since the diesel generator battery systems are separate and independent, the arrangement meets the requirements of GDC 5, "Sharing of [SSCs]."

The NRC staff requested that TVA explain how the dc system design meets the guidance provided in RG 1.32 and IEEE Standard 308-1971 with regard to the sharing of dc power sources at a multiunit nuclear power plant site. In its letter dated July 31, 2010, TVA stated the following:

> The vital batteries and battery chargers are demonstrated through analysis to have the capability to supply shared loads including those that would result from an accident in one unit and safe shutdown of the second unit...The safety loads are assigned to the vital 125-V batteries so that sharing will not significantly impair their ability to perform their safety functions including in the event of an accident in one unit and orderly shutdown and cool-down of the remaining unit while considering the effects of a single failure. ESF loads are assigned to batteries I and II for Unit 1 and to III and IV for Unit 2 such that an accident in one unit will not adversely impact battery sources for ESF loads of the other unit. The TDAFW [turbine-driven auxiliary feedwater] controls are supplied by battery III (normal) or battery IV (alternate) for Unit 1 and battery I (normal) or battery II (alternate) for Unit 2.

Based on the information provided by TVA, the NRC staff concludes that TVA has demonstrated that the sharing of the dc system will not significantly impair the ability of the system to perform its intended safety functions, including the scenario encompassing an accident in one unit and the orderly shutdown and cooldown of the remaining unit while considering the effects of a single failure. Therefore, the staff considers this issue resolved.

8.3.3.2.2 Sharing of Alternating Current Distribution Systems and Standby Power Supplies between Watts Bar Nuclear Units 1 and 2

The onsite ac power system is a Class 1E system consisting of the standby ac power system and the 120-V vital ac system. The standby power system serving each unit is divided into two redundant load groups (power trains). These power trains (train A and train B for each unit) supply power to all safety-related equipment. The equipment capacities identified in FSAR Tables 8.3-4 through 8.3-7 present the bus rating, connected load, and maximum demand load for each electrical distribution board in the standby (onsite) power system. Table 8.3-8 identifies the connected load and maximum demand load for each major transformer in the standby (onsite) power system. The EDG rating is 4,400 kW continuous or 4,840 kW for 2 hours out of 24 hours at a power factor of 0.8. The equipment capacities of the shared power systems have been evaluated in the original design of the plant and found to be acceptable. WBN Unit 2 FSAR Section 3.1.2 regarding compliance to GDC 5 states, in part, that "All shared systems are sized for all credible initial combinations of normal and accident states for the two units, with appropriate isolation to prevent an accident condition in one unit from carrying into the other." Therefore, the shared systems, for power operation, meet the requirement of GDC 5 that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions.

TVA stated that electric motors powered by the onsite distribution system of one unit drive safety-related machinery (e.g., essential raw cooling water (ERCW) pumps and condensate cleanup system pumps) required for safe shutdown of the other unit. For example, the ERCW system is arranged in two headers (trains), each of which serve certain components in each unit. The two independent ERCW trains are designed with duplicate safety-related mechanical and electrical components which are physically separated. Eight ERCW pumps are arranged electrically so that two pumps are fed from each shutdown board (1A-A, 1B-B, 2A-A, 2B-B). Only one pump per board can be automatically loaded onto an EDG at one time. The pumps supplied from the "A" boards pump into the "A" train header, and the "B" boards pump into the "B" train header. The control power to train A (both Units 1 and 2) is supplied from 125-V vital batteries I and III. Similarly, control power to train B (both Units 1 and 2) is supplied from 125-V vital batteries II and IV. TVA stated that a loss of control power to one plant train will not prevent the remaining plant train from performing its intended function.

The NRC staff requested information regarding the minimum number of power supplies (EDGs) and corresponding ERCW pump and isolation valve combination required to be operable at all times to ensure that safe shutdown can be achieved on the unit experiencing an accident and on the nonaccident unit assuming a simultaneous, worst-case single failure (including false or spurious accident signal) on each unit. In its letter dated August 30, 2010 (ADAMS Accession No. ML102510580), TVA stated the following:

> As stated [in] Unit 2 FSAR Section 9.2.1.3, the minimum combined safety requirements for one accident unit and one non-accident unit or two non-accident units are met by two pumps on the same plant train.... Watts Bar is capable of mitigating an accident in one unit and safely shutting down the other unit with or without offsite power and a total loss of either header. A total loss of one plant shutdown power train will not prevent safe shutdown of either unit under any credible plant condition.... In accident scenarios, either the "A" or the "B" Train of electrical power is assumed to be lost.

The NRC staff considers that, since there are four independent safety-related trains for the two units, TVA should address the combination of a single failure (or false actuation of accident signal) on cross trains (i.e., a single failure of an "A" power source on one unit combined with a single failure on a "B" power source of the second unit during a DBA and LOOP. Cross-train single failures, one per each unit, can potentially disable header valves such that ERCW flow may be restricted and result in inadequate cooling for operating equipment. A similar evaluation is needed for the condensate cleanup system and other shared systems. In its letter dated December 6, 2010, TVA has stated that this scenario is beyond the design basis for WBN. In accordance with Section C.2.b of RG 1.81, one single failure is considered in the design of WBN. Since WBN Unit 2 meets the guidance of Section C.2.b of RG 1.81, "Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants," Revision 1, the staff has concluded that this response is acceptable, in accordance with the guidance of SRM-SECY-07-0096, that the current licensing basis for WBN Unit 1 be used as the reference basis for the review and licensing of WBN Unit 2.

The electrical ac and dc systems have common buses and nonsafety loads supplied from train A or train B power supplies. In its letter dated August 30, 2010, TVA stated that separation is provided by selective coordination of protective devices for all ac (including 480 V) and dc circuits with molded case circuit breaker (MCCB) combinations or MCCB and fuse combinations or fuse/fuse combinations. Since selective coordination exists between the non-Class 1E and Class 1E circuits, the NRC staff concludes that this is acceptable.

8.3.3.2.3 Sharing of Raceway Systems between Units

In SER Section 8.3.3.2.3, the NRC staff stated the following:

> In Section 8.3.2.4.3 of the FSAR, [TVA] stated that WBN Unit 1 Train A cables may be routed in the same raceways as Unit 2 Train A cables, and Unit 1 train B cables may be routed in the same raceway as Unit 2 train B cables. Thus, raceway systems may be shared between units. At a September 17 and 18, 1981 meeting, [TVA] was requested to identify each shared raceway and provide the results of an analysis in accordance with GDC 5. The analysis must demonstrate that failure of all circuits located in each shared raceway will not significantly impair the capability of safety systems to mitigate an accident in one unit and an orderly and safe shutdown and cooldown of the remaining unit.

TVA, by letter dated October 9, 1981, provided the results of an analysis for one example in which the cables associated with the WBN Units 1 and 2 ERCW systems share a common raceway. Based on the results of this analysis, the NRC staff concluded that the design meets the guidelines of RG 1.81 and the requirements of GDC 5 and is acceptable for this example. For the remaining raceways that are shared between units, the staff requested a similar analysis be provided for staff audit verification that identifies the shared raceways with analyses results that demonstrate compliance with GDC 5. By letter dated January 7, 1982, TVA stated that cables for Units 1 and 2 of the same division may share a common raceway, but are physically separated from cables associated with the redundant division so that, given a failure of common division A cables, division B cables will not fail. Division B cables are sufficient to shut down both reactors given an accident in one and a safe shutdown in the other. The NRC staff concluded in the SER that, based on the information presented to date, the design meets the requirements of GDC 5 and is, therefore, acceptable.

Verification of the shared raceway design's conformance with GDC 5 through reviews of plant drawings and installation inspections is subject to the NRC construction inspection program.

8.3.3.2.4 Possible Sharing of Direct Current Control Power to Alternating Current Switchgear

Based on its review of FSAR Amendments 95 and 100, Section 8.3, "Onsite Power System," the NRC staff concludes that the WBN Unit 2 design is similar to the design of WBN Unit 1. Specifically, no automatic transfers of board supplies occur between redundant power sources. All 480-V shutdown boards and all motor control centers have alternate feeders to their respective board buses. Transfers between the normal and alternate feeders are manual. Some manual transfers of loads between power trains are used. FSAR Table 8.3-10 tabulates these transfers. The NRC staff asked TVA to confirm that it meets the NRC staff's requirement that FSAR Table 8.3-10 identify all possible interconnections between redundant divisions through normal and alternate power sources to various loads, regardless of the source of power, and that it will meet the staff's positions identified in Section 8.3.1.7 of the SER. The staff's positions in the SER were that (1) the circuit breaker at the 480-V shutdown board for the alternate feed to the battery charger be kept open and be alarmed in the control room when it is closed, and (2) the manual transfer switch must be alarmed in the control room if it is in the alternate supply position. The staff also asked TVA to confirm that redundant divisions are not cross-tied when both units are at power.

In its response letter dated December 6, 2010, TVA stated that Section 8.3.2.1.1, "Physical Arrangements of Components," in the WBN Unit 2 FSAR discusses that the interconnection between redundant divisions of normal and alternate power sources for the components listed in FSAR Table 8.3-10 is arranged to provide adequate physical isolation and electrical separation to prevent a common mode failure. The listed components in FSAR Table 8.3-10 also meet the staff's positions identified in Section 8.3.1.7 of the staff SER. TVA has reviewed the components listed in WBN Unit 2 FSAR Table 8.3-10 and verified that their normal and alternate power supplies are physically and electrically separated. TVA has indicated that the Integrated Safeguards Test conducted in accordance with RG 1.41, "Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments," will demonstrate the independence of the divisions and furthermore, these components are energized to support Unit 1 operation and no design change is required for their normal and alternate power supplies in support of two unit operation. Since the arrangement meets the staff's positions in the SER, the staff finds this response acceptable.

8.3.3.3 Physical Independence (Compliance with General Design Criterion 17)

(1) Physical Identification of Electrical Cables

The NRC staff's safety evaluation of the TVA Cable Corrective Action Program (CAP), dated August 31, 2009 (ADAMS Accession No. ML092151154), discusses the issue of physical cable separation and installation for WBN Unit 2. The staff's safety evaluation concluded the following:

> The NRC staff finds that TVA's CAP plan for the WBN Unit 2 cable issues acceptable because TVA's proposed approach for resolution of the issues establishes adequate program guidelines for identification, resolution, implementation, and inspection for each of the specific subissues as required by Appendix B to 10 CFR Part 50.
>
> The NRC staff may perform inspections to assure adequate implementation of the program. The staff will document the completion of the inspections in future inspection reports to verify that the items have been properly implemented. The NRC may also conduct inspections or audits to verify that TVA is adequately addressing its commitments with the Commitment Management Program.

(2) Associated Circuits

The WBN cable routing criteria allowed non-Class 1E cables associated with redundant Class 1E cables to share a common non-Class 1E raceway. Thus, the sharing of a common raceway by the non-Class 1E cables may compromise the independence of redundant cable systems. It is the NRC staff's position that non-Class 1E cables should meet the guidelines of Positions 4, 6, and 7 of RG 1.75, "Physical Independence of Electric Systems," and Sections 4.5 and 4.6.2 of IEEE Standard 384-1974. By letter dated March 3, 1982, TVA submitted an analysis for WBN of associated circuits in accordance with Section 4.6.2(1) of IEEE Standard 384-1974. Associated circuits, not separated from other non-Class 1E cables or protected from design-basis events, were identified and analyzed. The analysis demonstrated that electrical faults, caused by failure of the associated cables, will not compromise the independence of redundant Class 1E cable systems. The analysis verified that the cable's associated protective device will clear the imposed fault condition (in an acceptable time period) without exceeding the I^2t rating for the cable.

Based on the results of this analysis, the NRC staff concluded in the SER that, given proper operation of protection devices, there was reasonable assurance that Class 1E power systems and their distribution circuits will not be lost or degraded as a result of failure of non-Class 1E cables routed with Class 1E cables.

To ensure proper operation of protective devices, the NRC staff required that they be of a high quality commensurate with their importance to safety and that they be periodically tested. With the imposition of this requirement as a condition to the WBN license, the staff considers this item to be acceptable with respect to the functional aspects of electric power systems. The staff stated that those surveillance requirements associated with the protective devices will be reviewed with the TS. SSER 3 further reviewed this item and item 8.3.3(3).

Based on its review of the information provided by TVA, the NRC staff concludes that the above evaluation for WBN remains applicable to WBN Unit 2. TVA described the surveillance requirements for the protective devices in Section 8.3.1.4.2 of FSAR Amendment 95. The following section discusses the staff's review.

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

In SER Section 8.3.3.3(3), the NRC staff stated the following:

> Section 8.3.1.4.3 of the FSAR indicates that non-Class 1E cable trays are separated from Class 1E trays by 8 in. of air space vertically and 6 in. of air space horizontally. In addition, nonsafety circuits may be routed in the same conduit as Class 1E circuits at terminal equipment. Separation between an open top non-Class 1E cable tray and a Class 1E conduit has not been described in the FSAR. This separation of non-Class 1E from Class 1E circuits does not meet the guidelines of section 4.6.1 of IEEE Standard 384-1974. By letter dated March 3, 1982, the applicant submitted additional information concerning separation between Class 1E and non-Class 1E circuits at Watts Bar. Separation between 1E and non-Class 1E trays is 12 in. vertical (tray bottom to tray bottom) and 6 in. horizontal (tray side to tray side). Separation may be reduced to 9 in. vertical when solid bottom trays are used. Separation between conduits and open top cable trays was not described.
>
> In justification of the above described separation, the applicant has documented and substantiated by reference to tests or analysis (1) that a cable fault will be cleared by a protection device well before ignition temperature of cable insulation is reached and (2) that an electrically initiated fire may occur in cables routed in non-Class 1E cable trays, but that it is only minutely probable for the fire to propagate enough to challenge cables routed in adjacent Class 1E trays. Based on circuit protection, results of fire tests, and fire protection modifications, the applicant judges that the spacing of Class 1E and non-Class 1E cable systems provides adequate separation.
>
> Given proper operation of protective devices, the staff agrees with the applicant's judgment and concludes that there is reasonable assurance that Class 1E power systems and their distribution circuits will not be lost or degraded as a result of failure of non-Class 1E cables that are routed closer to Class 1E circuits than allowed by Regulatory Guide 1.75.
>
> To ensure proper operation of protective devices, the staff requires that non-Class 1E cables, routed closer to Class 1E cables than allowed by Regulatory Guide 1.75, be identified with their associated protective devices. The protective devices shall be of a high quality commensurate with their importance to safety and be periodically tested. With the imposition of this requirement as a condition to the Watts Bar license, the staff considers this item to be acceptable. Surveillance requirements for protective devices will be reviewed with the Technical Specifications.

In SSER 3, issued January 1985, the staff required that, for both items 8.3.3(2) and 8.3.3(3), protective devices used to isolate non-Class 1E from Class 1E circuits be of high quality commensurate with their importance to safety and that they be periodically tested. In Amendment 48 to the FSAR, TVA provided the results of a reliability study to demonstrate that a single circuit breaker which is periodically tested has reliability equivalent to each of the following protective device configurations:

(1) two series connected circuits breakers that are not tested
(2) a circuit breaker in series with a fuse that is not tested
(3) a single fuse that is not tested

On the basis of the results of this reliability study, the NRC staff concluded that each of the above configurations provides equivalent protection and is acceptable. In its letter dated January 17, 1984, TVA stated that (1) protective devices of the 160 associated circuits either have or are being modified to have one of the above protective device configurations and (2) protective devices of non-Class 1E circuits that are routed closer to Class 1E circuits than allowed by RG 1.75 have been identified and will be periodically tested, except those that have one of the above protective device configurations.

In FSAR Amendment 95, Section 8.3.1.4.2, TVA stated that electrical protection provides additional assurance that cables in conduits carrying cables of one division will not be damaged by an electrical fault on cables in open top trays or cables in free air of a redundant division. The reliability of circuit breakers that provide the electrical protection for some of these cables is enhanced by periodic testing. Appendix 8E discusses the results of a protection device reliability analysis. This analysis, based on data taken from IEEE Standard 500-1977, demonstrates that either of the following protective schemes has a reliability which is essentially equivalent to that of a single circuit breaker periodically tested:

(1) *a circuit breaker and fuse in series*
(2) two circuit breakers in series.

In addition to these protective schemes, IEEE Standard 500-1977 data verify that, for this application, a single fuse with no periodic testing has a failure rate approximately equal to the failure rate of two circuit breakers in series (see the Part B analysis of Appendix 8E). Therefore, a single fuse when used as an interrupting device for cables does not require periodic testing because of its stability, high reliability, and lack of drift. TVA concluded that any one of the following protective schemes provides a reliable means of protecting cables in conduits from electrical faults on cables in open top trays and cables in free air of the opposite train thus meeting the intent of RG 1.75 to not degrade redundant Class 1E cables:

(1) a circuit breaker and fuse in series
(2) two circuit breakers in series
(3) a single fuse
(4) a single circuit breaker periodically tested

The only exceptions to testing single Class 1E circuit breakers that protect cables in conduits from electrical faults on cables in open top trays and cables in free air of the opposite train will be those cases in which physical separation of specific circuits is shown to meet the requirements identified in IEEE Standard 384-1992. TVA is not committed to IEEE Standard 384-1992, but will use it as criteria for exempting individual circuits from circuit breaker testing. Some Class 1E MOV circuits were designed with two circuit breakers in series to maintain their safe operating position by administratively controlling one of the circuit breakers in the open position.

Since the motor operators are electrically isolated from their power supply during normal operation, periodic testing will not be performed on the circuit breakers. The MCCBs actuated by fault currents, which protect cables in conduits from electrical faults on cables in open top trays and cables in free air of the opposite train, will have at least 10 percent of each type breaker tested every 18 months and the recommended preventative maintenance performed on 100 percent of the Class 1E MCCBs within the past 72 months and non-Class 1E MCCBs within the past 96 months. For any breaker failure or breaker found inoperable, an additional 10 percent of that type will be tested until no more failures are found or all circuit breakers of that type have been tested. The test will ensure operability by simulating a fault current with an approved test set.

The NRC staff finds the information provided by TVA regarding isolation of non-Class 1E from Class 1E circuits to be acceptable. The NRC staff requested TVA confirm that, for those circuit breakers that are required to be tested periodically as discussed above, the TRM includes the surveillance requirements for both items 8.3.3.2 and 8.3.3.3. In a letter dated December 6, 2010, TVA stated that the breaker testing requirements are provided in Technical Requirement (TR) 3.8.1 of the WBN Unit 2 TRM. This section of the TRM was originally provided in accordance with a TVA to NRC letter dated March 4, 2009. It was updated in a TVA letter dated February 2, 2010. The NRC staff's review confirmed that necessary circuit breaker testing requirements have been included in Section TR 3.8.1 of the TRM submitted by TVA for Unit 2.

8.3.3.4 Compliance with NUREG-0737 Items

NUREG-0737, "Clarification of TMI Action Plan Requirements," issued November 1980, discusses two Three Mile Island action items relating to GDC 17: item II.E.3.1, "Emergency Power Supply for Pressurizer Heaters," and item II.G.1, "Emergency Power for Pressurizer Equipment." In letters to the NRC dated September 14, 1981, and March 11, 1982, TVA provided the background, the NUREG position, clarification of the positions, and descriptions of compliance for items II.E.3.1 and II.G.1.

(1) Emergency Power for Pressurizer Equipment (Item II.G.1)

The NRC staff reviewed the proposed resolution provided by TVA regarding the power supply requirements for the power-operated relief valves (PORVs) and block valves for WBN Unit 2 during its evaluation for WBN Unit 1 and considers it to be acceptable. The NRC staff requested TVA confirm for WBN Unit 2 that the PORVs and block valves will be powered from different emergency power sources (i.e., the PORV on dc power and its associated block valve on ac power), with both power sources emanating from the same division but from different buses. In a letter dated December 6, 2010, TVA stated that the Unit 2 PORVs (dc power supply) and block valves (ac power supply) are powered from the same division and from different buses. The Unit 2 Division A PORV is powered from Battery Board III, and the associated AC block valve is powered from 480V Reactor MOV Board 2A1-A. The Division B PORV is powered from Battery Board IV, and the associated AC block valve is powered from 480V Reactor MOV Board 2B1-B. The staff finds the TVA response as acceptable.

(2) Emergency Power Supply for Pressurizer Heaters (Item II.E.3.1)

In its SER, the NRC staff concluded that the WBN design meets the guidelines of item II.E.3.1 of NUREG-0737 and, therefore, is acceptable. The NRC staff requested TVA confirm that the WBN Unit 2 design will continue to meet the guidelines of item II.E.3.1 of NUREG-0737. In its letter dated December 6, 2010, TVA stated that the WBN Unit 2 power supply for the pressurizer heaters meets the requirements of NUREG-0737, Section II.E.3.1. The design for Unit 2 heaters is identical to that for the Unit 1 heaters. The Unit 2 heaters are fed from offsite power through safety-related 6.9kV Shutdown Boards (2A-A and 2B-B), and 6900/480V transformers. In case of a LOOP, the backup heaters are automatically loaded on to the EDGs after 90 seconds or they can be manually loaded by the operator in accordance with procedures (Abnormal Operating Instruction for Loss of Offsite Power, AOI-35 and System Operating Instructions, SOI-68-series). The staff finds the TVA response acceptable.

8.3.3.5 Compliance with General Design Criterion 18

TVA described its compliance with the requirements of GDC 18 in FSAR Amendment 95. The onsite power system is designed to be testable during operation of the nuclear power generating station, as well as during those intervals when the station is shut down. Based on its review of the information provided by TVA, the NRC staff concludes that TVA has met the requirements of GDC 18 with respect to the onsite ac and dc power system.

8.3.3.5.1 Compliance with Regulatory Guides 1.108 and 1.118

In SER Section 8.3.3.5.1, the NRC staff concluded that TVA met the intent of RG 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," and RG 1.118, "Periodic Testing of Electric Power and Protection Systems."

In FSAR Amendment 95, Section 8.1.5.3, TVA described its method for complying with the RGs. TVA has taken a minor exception to Position C.6(a) of RG 1.118, Revision 2. TVA stated that, "Where feasible test switches or other necessary equipment will be installed permanently to minimize the use of temporary jumpers in testing in accordance with the requirements in IEEE Standard 338-1977." The NRC staff reviewed the exception and concludes that it is not significant to safety and is, therefore, acceptable.

The NRC withdrew RG 1.108 in August 1993. In lieu of RG 1.108, TVA chose to comply with RG 1.9, Revision 3, "Selection, Design, Qualification, and Testing of Emergency Diesel Generators Units Used as Class 1E Onsite Electrical Power Systems at Nuclear Plants," issued July 1993. TVA has taken a number of exceptions (i.e., provided clarifications) to RG 1.9, Revision 3, as described in the notes to FSAR Amendment 95, Section 8.1.5.3. The NRC staff reviewed the exceptions to RG 1.9, Revision 3, and concludes that they are not significant to safety and are, therefore, acceptable.

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

In Section 8.3.1.1 of FSAR Amendment 95, TVA stated that tests will be performed on only one of the four power trains at any one time (see section entitled "System Testing"). Since TVA has updated the FSAR to reflect that tests will be performed on only one of the four power trains at any one time, the SER item is resolved for WBN Unit 2.

8.3.3.6 Compliance with General Design Criterion 50

GDC 50 requires, in part, that the reactor containment structures, including penetrations, be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA. The NRC staff reviewed Section 3.8.1 of FSAR Amendment 95 and concluded that it was acceptable. Therefore, the staff considers this item to be resolved for WBN Unit 2. The NRC staff concludes that TVA continues to meet the requirements of GDC 50 with respect to electrical penetrations containing circuits of the safety and nonsafety onsite power system.

8.3.4 Evaluation Findings

The NRC staff reviewed the onsite ac and dc power system for the WBN facility, including the single-line diagrams, station layout drawings, schematic diagrams, and descriptive information. The staff based its acceptance of the onsite power systems on conformance of the design criteria and bases to the Commission's regulations, as set forth in Appendix A to 10 CFR Part 50. The NRC staff concludes that the plant design meets the requirements of GDC 2, 4, 5, 17, 18, and 50 and conforms to the guidance of applicable RGs and NUREG reports, and is, therefore, acceptable, pending resolution of the open items noted in Section 8.3 above.

8.4 <u>Station Blackout</u>

On July 21, 1988, the NRC amended 10 CFR Part 50 to include 10 CFR 50.63, "Loss of All Alternating Current Power" (known as the SBO rule). The SBO rule requires that each light-water-cooled nuclear power plant be able to withstand and recover from an SBO of a specified duration. The SBO rule also requires licensees to submit information as defined in 10 CFR 50.63 and to provide a plan and schedule for conformance to the SBO rule. The SBO rule further requires that the baseline assumptions, analysis, and related information be available for NRC review. RG 1.155, "Station Blackout," and Nuclear Management and Resources Council, Inc. (NUMARC) 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," issued November 1987, provide guidance for conforming to the SBO rule. By letter dated August 31, 1992, as supplemented by letter dated January 27, 1993, TVA provided a response to the SBO rule for WBN Units 1 and 2.

The NRC staff issued a safety evaluation (SE), dated March 18, 1993, and a supplemental SE, dated September 9, 1993 (ADAMS Accession Nos. ML073200313 and ML073200358, respectively), which concluded that TVA's response to the SBO rule was acceptable for both WBN Unit 1 and Unit 2. Although the staff issued an SE for both units, TVA did not seek an operating license for Unit 2 at that time.

The SRP provides guidance to the NRC staff and to licensees on the technical information necessary to meet the regulatory requirements. SRP Section 8.4 provides guidance for addressing an SBO event. The NRC staff did not find relevant information addressing an SBO event during its review of FSAR Amendments 95 and 97, which provided information in support of the review of FSAR Section 8, "Electric Power Systems." The NRC staff performed its original review of WBN Unit 2 compliance with the SBO rule in 1992 and 1993. The NRC staff requested that TVA update or validate the original information or provide a new response addressing how WBN Unit 2 meets the SBO rule. The staff also requested that TVA update FSAR Section 8.4 to include the relevant information on SBO. TVA provided its response addressing SBO in its letter dated July 31, 2010.

8.4.1 Station Blackout Duration

Based on its review of TVA's original submittals, the NRC staff made findings and recommendations, which were documented in the staff SEs dated March 18 and September 9, 1993, as summarized below.

TVA calculated a minimum acceptable SBO duration of 4 hours based on a plant ac power design characteristic Group "PI," an emergency ac (EAC) power configuration Group "D," and a target EDG reliability of 0.975. The target EDG reliability was based on NUMARC 87-00, Section 3.2.4, which addressed minimum target reliability for plants in EAC Group D. The PI grouping is based on an independence of offsite power classification of Group "12," a severe weather (SW) classification of Group "2," and an extremely severe weather (ESW) classification of Group "1."

If the expected frequency of grid-related LOOP events, based on prior experience, exceeds once per 20 years, the ac power design characteristic group (P) should be classified as "P3." TVA stated that the frequency of such events at WBN, based on NUREG-1032, does not exceed once per 20 years. TVA further stated that site-specific records, as well as the annual report by the Electric Power Research Institute on LOOP events at U.S. nuclear plants, support the conclusion that the frequency of expected loss of grid-related LOOP events does not exceed once per 20 years for WBN. TVA stated that two 161-kV incoming lines were installed at WBN in 1977, and both lines were never out-of-service concurrently for the time periods covered up to the time when the SBO was implemented.

Other factors that affect the "P" classification are the "SW," "ESW," and "I" classifications for the site. The "SW" classification is based on the frequency of a LOOP resulting from severe weather. For sites not vulnerable to the effects of salt spray, this frequency depends on the following four factors:

(1) h_1—annual expectation of snowfall for the site in inches
(2) h_2—events per square mile of tornadoes of severity f2 or greater
(3) h_3—storms per year with wind velocities between 75 and 124 miles per hour (mph)
(4) b—factor based on the separation of the transmission lines from the plant

TVA used values for the above factors obtained from paragraph 3.2.1 of NUMARC 87-00 and calculated the frequency of a LOOP resulting from severe weather as 4.2775×10^{-3} per year, using the equation of paragraph 3.2.1, part IC. This results in an SW classification of 2. The NRC staff agreed with TVA's determination of these factors.

The ESW classification is based on the expectancy of storms with winds greater than 125 mph. TVA used the value of 1×10^{-4}, which was obtained directly from Table 3-2 of NUMARC 87-00. This results in an ESW classification of 1. The staff agreed with TVA's determination. The "I" classification depends on the capability of the plant to automatically and manually switch the power supply to the safety buses upon loss of the normal power source and the backup source or sources. For WBN, the normal source of power is from one of the two CSSTs (C and D). Upon the loss of power from one of the CSSTs, the capability exists for the manual transfer of the safety buses from the affected CSST (C or D) to the other CSST. Based on the guidance of NUMARC 87-00 or RG 1.155, TVA determined that the "I" classification for the plant is 1/2 (either 1 or 2). Based on the above described transfer scheme, the staff agreed with TVA's determination.

TVA states that four EDGs are available at the plant for the two units. TVA has determined, using NUMARC 87-00, Table 3-7, an EAC classification of D based on the fact that two specific EDGs (1A and 2A or 1B and 2B) are necessary (out of the four available) to operate safe-shutdown equipment (on a unit or plant basis) following a LOOP. TVA noted, and the staff agreed, that any three of the four EDGs will satisfy the requirement of two specific EDGs. Based on this, the staff agreed with TVA's EAC classification of D.

Plants with an EAC power configuration of D must select an EDG target reliability of 0.975 for the plant. TVA selected an EDG target reliability of 0.975, consistent with the given criteria. The required coping duration for an SBO event is based on the "P" classification (see NUMARC 87-00, Table 3-8), the EAC classification, and the EDG target reliability. The staff agreed with TVA's determination of a 4-hour coping duration based on a "P" classification of P1, an EAC classification of D, and an EDG target reliability of 0.975.

As noted above, the NRC staff did not find relevant information addressing an SBO event during its review of FSAR Amendments 95 and 97, which provided information in support of the review of FSAR Section 8. The NRC staff performed its original review of WBN Unit 2 compliance with the SBO rule in 1992 and 1993. Because so much time has passed since the original review, the NRC staff requested that TVA update or validate the original information or provide a new response addressing how WBN Unit 2 meets the SBO rule. The staff also requested that TVA update FSAR Section 8.4 to include the relevant information on SBO. The staff also requested that TVA provide information on the determination of specified coping duration to withstand and recover from an SBO in WBN Unit 2. Specifically, the staff requested that TVA provide its specified coping duration to withstand and recover from an SBO based on the factors listed in 10 CFR 50.63 and the expected frequency of a grid-related LOOP in the last 20 years. TVA addressed SBO in its letter dated July 31, 2010. In its response, TVA stated that WBN Design Criteria WB-DC-40-64, "Design Basis Events Design Criteria," Section

4.41, "Loss of All AC Power (Station Blackout (SBO))," defines the WBN SBO design basis. TVA documented its compliance with the SBO design basis in WBN calculation EPMMA041592, "Station Blackout Coping Evaluation," and in references to its response letter. The SBO coping time, determined in accordance with NUMARC 87-00 guidelines, is 4 hours.

TVA did not submit its design documents, WBN Design Criteria WB-DC-40-64, and WBN calculation EPMMA041592 to the NRC staff for review. The staff requested that TVA either submit the referenced documents to the NRC staff for review or provide a summary of the results that pertain to the information requested by the staff. In particular, TVA should validate and provide relevant information regarding the factors listed in 10 CFR 50.63 for determining specified coping duration to withstand and recover from an SBO and should demonstrate that the expected frequency of grid-related LOOP in the last 20 years did not exceed once per 20 years at the WBN Unit 2 site.

In letter dated December 6, 2010, TVA confirmed its validation for the effect of grid related LOOP events on SBO coping duration. TVA stated that determination of the duration of an SBO event was based on the following factors identified in10 CFR 50.63:

i. the redundancy of the onsite emergency AC power sources;
ii. the reliability of the onsite emergency AC power sources;
iii. the expected frequency of loss of offsite power; and
iv. the probable time needed to restore offsite power.

Items (i) and (iv) are features of the facility that have not changed since the original SBO analysis was performed and therefore, the information provided in the August 31, 1992, response remains valid.

The reliability of the onsite AC power sources (item ii) is monitored under TVA's fleet procedure NETP-100 (Emergency Diesel Generator Reliability Program) applies to the EDGs at TVA's three nuclear sites (including WBN Unit 2) and currently meets the goal of 0.975.

For item (iii), TVA had indicated in the August 31, 1992, response for WBN that grid-related LOOP classification (P-Group) was determined to be P1. That determination was based on the assumption that the site would not have more than one (1) LOOP event in a 20 year interval. TVA has noted that in the last 20 years, WBN has had one (1) LOOP event (i.e., on September 27, 2002, when the Watts Bar Hydro Station had an internal fire) but other aspects of item (iii) are features of the facility remain unchanged since the August 31, 1992 response. TVA has concluded that the P-Group classification remains at P1. The staff finds the response acceptable and concludes that Unit 2 coping duration assumptions remain valid.

8.4.2 Station Blackout Coping Capability

As required by 10 CFR 50.63(a)(2), licensees must demonstrate the capability for coping with an SBO of specified duration by an appropriate coping analysis. The NRC staff requested that TVA provide a summary of the strategies and analysis for coping with an SBO in WBN Unit 2 for the specified duration. The staff also requested that TVA provide sufficient information, including baseline assumptions, on the systems and equipment required for coping with an SBO for the specified duration without ac power for the issues discussed in the following sections.

8.4.2.1 Condensate Inventory for Decay Heat Removal

Based on its review of TVA's original submittals, the SEs dated March 18 and September 9, 1993, summarize the NRC staff's original findings and recommendations for WBN Unit 2. The staff reviewed the characteristics of the condensate inventory for decay heat removal to ensure that the systems have the availability, adequacy, and capability to achieve and maintain a safe shutdown and recover from an SBO with a 4-hour coping duration. The staff findings from its SE dated March 18, 1993, are as follows:

> TVA indicated that based on the guidance described in NUMARC 87-00, 75,451 gallons of water per unit are required for decay heat removal during an SBO with a 4-hour coping duration. The minimum permissible condensate storage tank (CST) level per the plant draft technical specifications (TS) will provide 210,000 gallons of water per unit, which exceeds the required quantity for coping with a 4-hour SBO event. Based on its review of similar Westinghouse plants (e.g., Sequoyah, Units 1 and 2) and provided that the final TS for the amount of water stored in the CST will not deviate significantly from the above draft TS, the staff concludes that TVA will have sufficient condensate inventory to cope with a 4-hour SBO event at the Watts Bar plant.

As part of its current review, the NRC staff requested that TVA provide information on the capacity of the condensate storage tank to ensure that a sufficient water inventory will exist to remove decay heat during the specified SBO duration. In its letter dated July 31, 2010, TVA stated the following:

> Per Technical Specification 3.7.6, the CST shall have at least 200,000 gallons reserved for the Auxiliary Feedwater System. It will take approximately 75,500 gallons of CST inventory to remove decay heat (without cooldown) during the SBO coping period. In the unlikely event that plant cooldown is required, then the required inventory from the CST is 197,200 gallons.

Although the information provided by TVA in its letter dated July 31, 2010, differs slightly from the original information provided on the condensate storage tank, the NRC staff concludes that TVA will have sufficient condensate inventory to cope with a 4-hour SBO event at the WBN plant. TVA should resolve the differences in the information provided to the NRC staff in its submittal dated August 31, 1992, and the information provided in its response dated July 31, 2010, regarding the condensate storage tank and the amount of condensate required for a 4-hour coping duration. In a letter dated December 6, 2010, TVA has provided a tabulation of condensate inventory requirements for coping with a 4 hour SBO event at WBN Unit 2. TVA concluded that the total CST inventory required when the reactor started at 102 percent of rated thermal power without cooldown is 76,960 gallons, which is less than the 200,000 gallons required in the Unit 2 TSs. The staff finds the response acceptable and concludes that the condensate storage tank has adequate capacity to ensure that a sufficient water inventory will exist to remove decay heat during the specified SBO duration.

8.4.2.2 Class 1E Battery Capacity

The SEs dated March 18 and September 9, 1993 summarize the NRC staff's original findings and recommendations on Class 1E battery capacity for WBN Unit 2. The staff reviewed the battery capacity, including load shedding to conserve battery capacity, to ensure that the battery systems have the availability, adequacy, and capability to achieve and maintain a safe shutdown and recover from an SBO with a 4-hour coping duration. The staff findings, as documented in its SE dated March 18, 1993, were as follows:

> For the Class IE 125V vital batteries, TVA takes credit for non-essential load shedding at 30 minutes into the SBO event. TVA's submittal included a list of the loads to be shed and the loads that will remain. TVA stated that no loads will be shed that are necessary for the SBO safe shutdown path, and at least three separate circuits (equivalent to one train) of dc-powered emergency lighting is assured available in the control room. In response to staff questions, TVA stated that the load shedding would begin immediately once the SBO event is identified and would involve opening approximately 121 individual breakers to shed miscellaneous loads. These loads are grouped in various panels located in the battery board rooms. In addition, there are four lighting panels located in the shutdown board rooms that require breaker manipulations for load shedding. These rooms are easily accessible and in close proximity to the main control room. Operations personnel estimate that the load shedding can be completed in approximately 20 minutes. TVA stated that the operating procedures will be structured such that if one of the breakers fails to open, the load shedding of the remaining breakers would continue. TVA stated that with load stripping, the final battery voltage at the end of the 4-hour SBO event is greater than 105 Volts and that this battery voltage is above the minimum required for successful operation of the equipment. The IEEE Standard 485 methodology was used in evaluating the battery adequacy. An aging factor of 1.25 and a temperature correction factor of 1.11 based, on a worst-case minimum battery room temperature of 60 °F were used.
>
> In response to a staff question regarding the margin remaining on the battery at the end of the 4-hour SBO event, TVA stated that additional margin above that provided by the temperature correction and aging factor was not specifically addressed in the battery calculation. However, a conservative load profile was used in the calculation. In particular, a three-step load profile was used for the calculation, and the ampere loads for the third step were based on a minimum battery voltage of 105 volts. In order to provide assurance that additional margin was available, TVA made a calculation with the third step of the load profile separated into three steps using the expected voltages for these load segments. The calculation indicated that a margin of approximately 6.9 percent would be available. TVA also noted that any future load growth could be accommodated by revisiting the load shedding list. For EDG field

flashing, WBN has separate 125 Vdc batteries from those discussed above. TVA proposes to attempt two EDG starts (field flashings) at the beginning of the SBO event (to identify that the event has occurred) and to reserve one start attempt (field flashing) for the end of the 4-hour period. TVA stated that the adequacy of these batteries have been analyzed using the IEEE Standard 485 methodology and that the results demonstrate sufficient battery voltage after the 4-hour event to flash the generator field.

In its August 31, 1992 submittal, TVA stated that no credit was taken for the 250 V non-Class 1E batteries (e.g., for closure or control of the breakers in the switchyard) for restoration of ac power from the offsite power sources at the end of the SBO event. Instead, the restoration of ac power was to be accomplished by the EDGs. The staff, in its RAI [request for additional information], stated that this assumption is not acceptable, and that the battery capability must be sufficient for restoration of power by each power source (i.e., one cannot choose which power source will become available). In its response to the staff's RAI, TVA stated that the ac power restoration procedures will allow the operators to attempt to restore power from either the EDGs or the switchyard/offsite power sources.

Further, TVA is currently in the process of replacing the 250 V batteries and has made an analysis which confirms that the replacement batteries have sufficient capacity, with implementation of load shedding procedures, to perform any needed control functions for restoration of offsite power after 4 hours.

In staff SEs dated March 18 and September 9, 1993, the staff found that there was reasonable assurance that the 125-V vital batteries, the 125-V EDG batteries, and the 250-V switchyard batteries will have sufficient capacity to cope with and recover from an SBO of 4 hours based on the information provided to the staff.

In its current review in support of the WBN Unit 2 application, the NRC staff requested that TVA provide information on the adequacy of the battery capacity to support loads required for decay heat removal for the specified SBO duration and EDG field flashing for recovering onsite power sources. In its letter dated July 31, 2010, TVA stated the following:

(1) 125 VDC Vital Power System:

As stated in Unit 2 FSAR 8.3.2.1.1, the vital 125V DC control power system batteries are designed to support an SBO event for four (4) hours. The coping duration for Watts Bar cannot be met with all the normal loads. Loads that are not required to mitigate an SBO will be removed within 30 minutes into the event to increase the discharge time of the battery. The battery capacity analysis demonstrates the capacity to provide the remaining loads for up to four (4) hours. The evaluation includes allowances for aging, design margin and temperature derating.

(2) 250 VDC Battery System:

As stated in Unit 2 FSAR 8.2.1.4, the 250 VDC batteries are designed to support an SBO event for four (4) hours. The coping duration for Watts Bar cannot be met with all the normal loads. Loads that are not required to mitigate an SBO will be removed between one (1) and three (3) hours into the event to increase the discharge time of the battery. The battery capacity analysis and [sic] demonstrates the capacity to provide the remaining loads for up to four (4) hours. The evaluation includes allowances for aging, design margin and temperature derating.

(3) 125 VDC Emergency Diesel Generator (EDG) Power System:

As stated in Unit 2 FSAR 8.3.1.1, the EDG batteries are designed to support an SBO event for four (4) hours. The batteries are required to provide power and indication to allow starting of the EDG to recover from the event. The coping duration can be achieved with the battery and all loads connected, provided that a maximum of three (3) start sequences are attempted. The batteries will have sufficient capacity remaining to "flash the generator field" with the third and final start occurring at the end of the coping period. The battery capacity analysis demonstrates the capacity to provide the remaining loads for up to four (4) hours. The evaluation includes allowances for aging, design margin and temperature derating.

Based on the above responses to the NRC staff questions, TVA maintains that the 125-V dc vital power system battery, 250-V dc battery system, and 125-V dc EDG power system have sufficient capacity with load shedding to cope with and recover from an SBO.

The NRC staff asked TVA to clarify what analysis was used for the batteries in order to reach a conclusion that batteries have adequate capacity to achieve and maintain a safe shutdown and recover from an SBO for a 4-hour coping duration. In its response letter dated December 6, 2010, TVA stated that, as discussed in WBN Unit 2 FSAR Section 8.3.2.1.1, the capability of a 125 V dc battery system was verified by analysis for each battery using normal system alignment with loss of all ac power to both the units. This 125 V dc system analysis demonstrates that each vital battery has adequate capacity to supply the required loads to achieve and maintain a safe shutdown of both the units and to recover from the SBO event. Based on the response, the staff concludes that the batteries have adequate capacity to supply the necessary loads following an SBO.

8.4.2.3 Compressed Air

In its SE dated March 18, 1993, the NRC staff stated the following:

> TVA indicated that the following modifications and procedure changes are necessary to ensure that air-operated valves required for decay heat

removal during a 4-hour SBO event have sufficient backup sources for operation or can be manually operated.

> The air-operated valves that will require modifications and associated procedure changes for the SBO event are the air-operated turbine-driven auxiliary feedwater pump (TDAFWP) level control valves. The amount of air available to operate these valves will be supplemented by the addition of nitrogen bottles. These modifications together with the associated procedure changes will provide sufficient compressed gas for the control of these valves during the 4-hour duration of an SBO event.
>
> Based on its review, and provided that the modifications and procedures discussed above will be implemented, the staff concludes that TVA will have sufficient compressed air to cope with a 4-hour SBO event at the Watts Bar plant.

In its current review in support of WBN Unit 2, the NRC staff requested that TVA provide information on the compressed air capacity to ensure that air-operated valves required for decay heat removal have sufficient reserve air and that appropriate containment integrity will be maintained for the specified duration. In its letter dated July 31, 2010, TVA stated the following:

> The only Air Operated Valves (AOVs) required during the SBO coping period are the Turbine Driven Auxiliary Feedwater Pump (TDAFWP) Level Control Valves (LCVs). These valves are normally supplied by the Auxiliary Control Air System (ACAS). This system will not be available during an SBO event. Modifications to install bottled nitrogen to supply these valves will be made to the plant. The bottled nitrogen has sufficient capacity to supply the TDAFWP LCVs during the SBO coping period. The nitrogen bottles are normally installed. They are sized for five (5) LCV cycles. The Appendix R event is more limiting than the SBO, so the nitrogen bottle sizing is based on Appendix R.

Based on its review of the information provided by TVA, and provided that the associated procedure changes are implemented, the NRC staff concludes that TVA will have sufficient compressed air to cope with a 4-hour SBO event at WBN Unit 2. TVA should confirm to the NRC staff that the modifications to install bottled nitrogen to supply the TDAFW pump level control valves (LCVs) included WBN Unit 2, as well as Unit 1. In a letter dated December 6, 2010, TVA stated that design change packages have been issued to install the nitrogen bottles for the Unit 2 TDAFWP LCVs. The staff finds the response acceptable, because TVA is installing bottled nitrogen to supply the TDAFW pump LCVs included installed in WBN Unit 2.

8.4.2.4 Effects of Loss of Ventilation

Based on its review of TVA's original submittals, the NRC staff documented its original findings and recommendations for WBN Unit 2 in the SEs dated March 18 and September 9, 1993. The staff reviewed the characteristics of the effects of loss of ventilation to ensure that the systems have the availability, adequacy, and capability to achieve and maintain a safe shutdown and recover from an SBO with a 4-hour coping duration. In its SE dated March 18, 1993, the staff stated the following:

> TVA performed a steady-state heat-up analyses in accordance with NUMARC 87-00 guidelines to determine the effects of loss of ventilation in main control room complex without heating, ventilation, and air conditioning (HVAC), TDAFWP room, north and south main steam valve rooms, 125V vital battery rooms, 125V vital battery board rooms, cable spreading room, pipe chase, 480V board rooms, and 6.9kV and 480V shutdown board room. The calculated peak temperatures in each of these areas are as follows:
>
> | Rooms | Peak Temp. (°F) |
> |---|---|
> | Main Control Room Complex (without HVAC) | 104 |
> | TDAFW Pump Room | 122 |
> | North Main Steam Valve Room | 162 |
> | South Main Steam Valve Room | 177 |
> | 125V Vital Battery Rooms | 95 |
> | 125V Vital Battery Board Rooms | 103 |
> | Cable Spreading Room | 104 |
> | Pipe Chase | 122 |
> | 480 V Board Rooms | 104 |
> | 6.9k V and 480 V Shutdown Board Room | 103 |
>
> The calculated steady-state temperatures for the above rooms with the exception of the north and south main steam valve rooms are well below the temperature limits described in NUMARC 87-00, Section 2.7. TVA indicated that there is no SBO equipment located in the north main steam valve room; therefore, the south main steam valve room which contains the TDAFWP level control valves is the only dominant area of concern (DAC). TVA further indicated that reasonable assurance of the operability of SBO response equipment in the above DAC had been assessed and that no system modifications or associated procedures would be required. TVA has also committed to revise the procedure for opening the control room cabinet doors within 30 minutes following an SBO event in accordance with the guidance described in NUMARC 87-00.

In its current review of Watts Bar Unit 2, the NRC staff requested that TVA discuss the integrity of the electrical cabinets and provide information on the effects of the loss of ventilation to other equipment, such as the TDAFW pump, valves, battery room, and other equipment credited for mitigating an SBO event. The staff also requested that TVA provide information on the effects of a loss of ventilation in all dominant areas of concern and on the equipment credited during an SBO event. In its letter dated July 31, 2010, TVA stated the following:

> Detailed room temperature evaluations consistent with RG 1.55 and NUMARC 87-00, Revision 1, guidelines have been performed in areas containing equipment required to cope with an SBO event. The following areas were evaluated:

1) 250 V Battery and Board Rooms
2) Control Room Complex
3) Cable Spreading Room
4) 125 V Battery and Board Rooms
5) 480 V Board Rooms
6) Pipe Chase Area
7) North and South Main Steam Valve Rooms
8) Turbine Driven Auxiliary Feedwater Pump room
9) 6.9 kV and 480 V Shutdown Board Room A

> In general, and consistent with Sequoyah and Watts Bar Unit 1, electrical heat loads in these areas are assumed to be reduced by 50 percent of their normal values. An assumption of 50 percent reduction is considered to be very conservative as all ac power has been lost with only battery power remaining. In the rooms where actual heat loads have been calculated, the results have been less than 50 percent of normal. If the corresponding evaluation predicted excessive temperatures, more detailed evaluations of actual electrical heat loads were performed. Most HVAC systems are AC powered and are therefore lost during an SBO event. The TDAFW pump room is serviced by a DC powered ventilation system. Resulting room temperatures are within equipment qualification limits and support human habitability requirements (if any).

TVA did not provide calculated peak temperatures for the control room complex, the TDAFW pump room, the north and south main steam valve rooms, the 125-V vital battery and board rooms, the 250-V battery and board rooms, the cable spreading room, the pipe chase, the 480-V board rooms, and the 6.9-kV and 480-V shutdown board room. However, TVA stated that the heatup calculations show the resulting temperatures to be less than 50 percent of normal. The TDAFW pump room is serviced by a dc-powered ventilation system and, therefore, is not affected by the loss of all ac power. TVA also stated that it had performed more detailed evaluations of actual electrical heat loads where necessary, and the resulting room temperatures are within equipment qualification limits and support human habitability requirements. Based on the information provided by TVA in its letter dated July 31, 2010, and in its original submittal to the NRC in 1992 regarding the effects of a loss of ventilation, the NRC staff concludes that TVA has adequately addressed the effects of a loss of ventilation during an SBO event at WBN Unit 2. Therefore, the staff finds TVA's evaluation of the effects of a loss of ventilation to be acceptable, provided the peak temperatures in the dominant areas of concern remain below the temperature limits described in NUMARC 87-00, Section 2.7, and pending TVA's revision of the procedure for opening the control room cabinet doors within 30 minutes following an SBO event, in accordance with the guidance in NUMARC 87-00.

8.4.2.5 Containment Isolation

In its SE dated March 18, 1993, the NRC staff stated the following:

> TVA reviewed the plant list of containment isolation valves (CIVs) to ensure that valves which must be capable of being closed or operated

(cycled) during an SBO event can be positioned (with indication) independent of the blacked-out unit's power supplies. TVA stated that no plant modifications are necessary to insure containment integrity during an SBO event. Based on its review, the staff concludes that CIV design and operation at Watts Bar plant have met the intent of the guidance described in RG 1.155 and are, therefore, acceptable.

Based on the information provided by TVA, the NRC staff concludes that containment isolation valve design and operation at WBN Unit 2 meets the intent of the guidance described in RG 1.155. However, TVA should either confirm that the original information it submitted to the NRC staff is applicable to WBN Unit 2 or provide additional information to ensure that valves at WBN Unit 2 which must be capable of being closed or operated (cycled) during an SBO event can be positioned (with indication) independent of the blacked-out unit's power supplies. In a letter dated December 6, 2010, TVA stated that the information previously provided for the containment isolation valves remains valid. TVA also identified exceptions to valve numbers FCV-63-72, FCV-72-44, FCV-63-73, FCV-72-45, FCV-62-61, FCV-62-63, and FCV-74-2, which are either associated with a 'closed system' or can be manually closed during a SBO. Valve FCV-74-2 is a normally closed 'fails-as-is' valve. The staff finds TVA's response acceptable because it confirmed that the original information TVA submitted to the NRC staff is applicable to WBN Unit 2

8.4.2.6 Reactor Coolant Inventory

In its SE dated March 18, 1993, the NRC staff stated that, for its inventory analysis in support of the original WBN review, TVA used a reactor coolant pump (RCP) seal leakage of 25 gallons per minute (gpm) for each of four pumps, and an allowable identified reactor coolant system (RCS) leakage of 10 gpm, resulting in a total leakage of 26,400 gallons during the 4-hour coping duration. The value of 26,400 gallons is equivalent to about 3,530 cubic feet (ft^3). TVA stated that the letdown isolation valve closes when the air supply is lost subsequent to the loss of ac power. In the unlikely event that cooldown is required, TVA estimated a shrinkage of 3,653 ft^3. The total volume of the RCS is about 12,145 ft^3. TVA concluded that, with the leakage plus the shrinkage, there would be sufficient RCS inventory remaining to keep the core covered. TVA stated that it understands that, if the final resolution of Generic Issue (GI) 23 defines higher RCP seal leakage rates than the 25 gpm assumed, the resolution could potentially affect its analysis and actions addressing conformance to the SBO rule. Based on the above, the staff concluded that there is reasonable assurance that sufficient RCS inventory exists to keep the core covered during a 4-hour SBO event.

During its current review of WBN Unit 2, the NRC staff requested that TVA discuss the core and reactor system conditions and the ability to maintain adequate RCS inventory to ensure that the core is covered and cooled. The staff also requested that TVA discuss and provide information on RCS inventory, taking into consideration shrinkage, leakage from pump seals, and inventory loss from letdown or other normally open lines.

In its letter dated July 31, 2010, TVA stated that the success criterion for RCS inventory is that the core remains covered throughout the SBO coping period (4 hours), accounting for (1) shrinkage, (2) letdown, (3) normal system leakage, and (4)RCP seal leakage. In particular, TVA stated that the following:

1) WBN is a "Hot Standby" plant, so shrinkage is not significant. Even if a cool-down to 350 °F occurred within the SBO coping period, the core would remain covered.
2) The letdown containment isolation valve closes on loss-of-AC power.
3) The normal system leakage is limited to 10 gpm (identified leakage) by TS 3.4.13.
4) RCP seal leakage is assumed to be 25 gpm per RCP. This is conservative with respect to the 21 gpm provided in [Topical Report] WCAP-10541, "Westinghouse Owners Group Report, Reactor Coolant Pump Seal Performance Following a Loss of All AC Power". This value has also been assessed with respect to seal leak-off line failure as reported by Revision 2 of WCAP-10541. The seal leak-off line will not fail at WBN. The total RCS inventory at the end of the SBO coping period is approximately 8,600 ft^3 (assuming no shrinkage) versus a reactor vessel volume of approximately 5,000 ft^3. Therefore, the core remains covered.

The information provided by TVA in its letter dated July 31, 2010, regarding its RCS evaluation differs slightly from the original information provided to the NRC staff; however, the staff concludes that TVA will have sufficient RCS inventory to cope with a 4-hour SBO event at WBN Unit 2 and the reactor core will remain covered. In its original submission on SBO in 1992, TVA stated that the total RCS inventory is 12,145 ft^3 and, based on a 35-gpm leakage (25 gpm for each of four pumps and an allowable identified RCS leakage of 10 gpm), the total volume used is equivalent to about 3,530 ft^3. Should cooldown be required, TVA estimated shrinkage of 3,653 ft^3. Therefore, the total RCS volume used would amount to 7,183 ft^3 when compared against the total RCS inventory of 12,145 ft^3, thereby assuring that the core will remain covered.

In its letter dated July 31, 2010, TVA stated that the total RCS inventory at the end of the SBO 4-hour coping period is approximately 8,600 ft^3 (assuming no shrinkage). TVA did not provide information on shrinkage in the unlikely event that cooldown should be required. Assuming shrinkage of 3,653 ft^3 because of cooldown, as stated in the original submittal, the total RCS inventory at the end of the SBO 4-hour coping period would be approximately 4,947 ft^3 versus a reactor vessel volume of approximately 5,000 ft^3.

The staff requested that TVA include the shrinkage from cooldown in its assessment of the RCS inventory for an SBO of 4-hour coping duration. TVA should also resolve the differences in the information provided to the staff in its submittal dated August 31, 1992, and the information provided in its letter dated July 31, 2010.

In letter dated December 6, 2010, TVA provided the following clarifications:

- RCS Volume is 12,145 ft^3
- Calculated Leakage is 3,530 ft^3
- Remaining volume, not including shrinkage due to system cooldown, is 8,615 ft^3
- Shrinkage due to system cooldown is 3,653 ft^3
- Remaining volume following shrinkage due to system cooldown is 4,962 ft^3
- Volume of the reactor vessel, including plenum above active fuel, is 4,945 ft^3

The staff finds the response acceptable because it provides clarification to assure that the core will not be uncovered during a SBO event.

8.4.3 Procedures and Training

The requirements in 10 CFR 50.63(a)(2) state that, "The capability for coping with a station blackout of specified duration shall be determined by an appropriate coping analysis." The assumptions and results of the analysis are incorporated into site-specific procedures and training. In its submittal dated August 31, 1992, TVA stated that, in accordance with NUMARC Initiative 2, TVA will implement site-specific procedures for the following:

- coping with a 4-hour SBO event
- restoration of ac power following a 4-hour SBO event
- preparing the plant for severe weather conditions to reduce the likelihood and consequences of a LOOP and to reduce the overall risk of an SBO event

In its submittal dated August 31, 1992, TVA briefly described its plans to modify the operating procedures or instructions applying to loss of ac power, ac power restoration, response to severe weather, and response to a tornado watch or warning.

TVA stated that the schedule for the procedure-related actions encompassing reviewing, revising, and implementing the procedures, including verification of adequate staffing and training, was 2 years following the NRC issuance of a final SE, in accordance with 10 CFR 50.63(c)(4).

The NRC staff did not review the procedures or proposed procedure modifications or associated training proposed in TVA's letter dated August 31, 1992. However, during its evaluation of the current information for WBN Unit 2, the NRC staff requested that TVA provide a summary of information on site-specific procedures and training on the following:

- coping with an SBO for the specified duration
- restoration of ac power following an SBO event of specified duration
- preparation for severe weather conditions to reduce the likelihood and consequences of a LOOP and to reduce the overall risk of an SBO event

In its letter dated July 31, 2010, TVA stated the following:

> The Unit 1 procedure for loss of shutdown board power is ECA-0.0 (Loss of Shutdown Power, currently Rev. 20). This procedure provides actions for responding to a loss of shutdown power. It also directs the restoration of shutdown board power based on the cause for the loss of shutdown board power. Step 6.a, Response Not Obtained (RNO), directs the operator to utilize AOI-35 (Loss of Offsite Power) or AOI-40 (Station

Blackout), or AOI-43 (Loss of Shutdown Boards). If the loss of shutdown power is due to a Station Blackout, the operator will refer to AOI-40. ECA-0.0 (Loss of Shutdown Power) is being drafted for Unit 2 and will be issued to support Unit 2 Startup.

In addition, TVA stated that it has drafted Procedure AOI-40 for WBN Unit 2, which will be issued to support Unit 2 startup. TVA stated that Procedure AOI-40 provides guidance for restoration of shutdown ac power via the EDGs or backfeed from the 500-kV system. It also provides for operator actions to reduce load on the 125-V vital and 250-V station batteries for complete loss of all ac power to extend the useful life of the dc backup power system or systems. Section 3.2 of Procedure AOI-40 provides restoration steps to return plant components and station service to normal configuration.

TVA also stated that it has drafted Procedure AOI-8, "Tornado Watch or Warning" (currently Revision 051), for WBN Unit 2, which will be issued to support Unit 2 startup. Procedure AOI-8 provides actions to be taken in the event that a tornado watch or tornado warning is issued for WBN. Procedure AOI-8 also provides guidance to prevent damage to the facility from the potential effects of tornadoes at or near the site. The procedure includes actions such as anchoring cranes, verifying or establishing required damper/door/hatch positions, stabilization of fuel movements, and securing loose items outside of buildings.

The NRC staff determined from its response that TVA is developing or has drafted the required SBO procedures, as discussed above, and will implement these procedures to support the startup of WBN Unit 2 for coping and recovering from an SBO of specified duration. The staff finds this acceptable. TVA should maintain and implement the required SBO procedures, including any others that may be required to ensure an appropriate response to an SBO event to support the startup of WBN Unit 2. Also, TVA should provide the appropriate training to its operating staff on the required SBO procedures to support startup of WBN Unit 2. Based on the NRC staff's review of the information provided by TVA, the procedures and training issue is resolved.

The preparation and implementation of site-specific procedures and applicable WBN staff training on (1) coping with an SBO for the specified duration, (2) restoration of ac power following an SBO event, and (3) preparation for severe weather conditions to reduce the likelihood and consequences of a LOOP and to reduce the overall risk of an SBO event are subject to future NRC inspection.

8.4.4 Proposed Modifications

In its original submittal dated August 31, 1992, TVA stated that the only equipment modification required was to provide nitrogen bottles which contain sufficient capacity for the duty cycle of the LCVs during the 4-hour SBO event. TVA stated that the schedule for the equipment-related modifications was 2 years following the NRC issuance of a final SE, in accordance with 10 CFR 50.63(c)(4).

In response to an NRC question regarding the information presented for WBN Unit 2, TVA provided information on the compressed air capacity in its letter dated July 31, 2010. The staff wanted to ensure that air-operated valves required for decay heat removal have sufficient reserve air and that appropriate containment integrity will be maintained for the specified duration. TVA stated that it had modified the plant to install bottled nitrogen to supply these valves. The bottled nitrogen has sufficient capacity to supply the TDAFW pump LCVs during the SBO coping period. The nitrogen bottles are normally installed. They are sized for five LCV cycles. TVA stated that the Appendix R event is more limiting than the SBO, so the nitrogen bottle sizing is based on Appendix R.

The staff requested TVA specify whether or not the completed modifications to install bottled nitrogen to supply TDAFW pump LCVs were for both WBN Unit 1 and Unit 2, or for only WBN Unit 1, since WBN Unit 2 did not have an operating license before issuance of the NRC staff's final SE on SBO on September 9, 1993. The staff's review of TVA's response was included in Section 8.4.2.5 of this report.

8.4.5 Quality Assurance and Technical Specifications

Section 2.5 of the NRC staff's SE dated March 18, 1993, states the following:

> With respect to quality assurance (QA), TVA stated that for any non-safety related equipment used to meet the requirements of 10 CFR 50.63 that are not already covered by QA requirements of 10 CFR Part 50, Appendix B or 10 CFR Part 50, Appendix R, Watts Bar will apply the guidance and specifications of RG 1.155, Appendices A and B, respectively. The staff finds this commitment to be acceptable.

Section 2.5 of the NRC staff's SE dated March 18, 1993, also states that, "With respect to Technical Specifications (TS), TVA stated that the equipment used for an SBO event is already part of the plant's draft proposed TS, or the equipment (e.g., the condensate components) is in continuous use during normal plant operation." This is acceptable to the staff, since the staff expects that plant procedures will reflect appropriate testing and surveillance requirements to ensure the operability of the necessary SBO equipment, even though there are no specific TS requirements for the equipment credited for the SBO.

The NRC staff found TVA's responses to be acceptable regarding (1) nonsafety-related equipment used to meet the requirements of 10 CFR 50.63 that is not already covered by the quality assurance requirements of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 or Appendix R to 10 CFR Part 50 and (2) technical specification considerations for SBO equipment, as documented in the staff's SE dated March 18, 1993. However, since WBN Unit 2 was not licensed at the time, TVA should confirm its responses, as provided in Sections 7 and 8 of Enclosure 2 to TVA's letter dated August 31, 1992, with regard to WBN Unit 2.

In a letter dated December 6, 2010, TVA made the following commitment:

> Unit 2 will have proceduralized testing and surveillance requirements consistent with those discussed in the August 31, 1992 response. Unit 2 will meet the requirements of RG 1.155, Revision 0, "Station Blackout," Appendices A and B".

The staff finds this commitment acceptable.

8.4.6 Emergency Diesel Generator Reliability Program

Section 2.6 of the NRC staff's SE dated March 18, 1993 states the following:

> TVA states that Watts Bar will utilize the target reliability of 0.975 in the reliability program that is to be established to conform to RG 1.155, Section 1.2, and NUMARC 87-00, Revision 1, Appendix D. TVA states that it understands that the resolution of Generic Issue (GI) B-56 could potentially impact the above guidance provided for the reliability-program.
>
> The staff accepts TVA's commitment. However, until GI B-56 is resolved, TVA should follow the guidelines of RG 1.155, Section 1.2, or NUMARC 87-00, Revision 1, Appendix E. The staff has not accepted Appendix D of NUMARC 87-00, Revision 1, in its entirety.

Based on its current configuration, TVA is using four EDGs to support WBN Unit 1 operations. The four EDGS were originally designed (two EDGs for each unit) to support WBN Unit 1 and Unit 2. However, WBN Unit 2 was not licensed at the time, and TVA used the four EDGs for WBN Unit 1 operations. The four EDGs have been maintained and covered by the existing WBN Unit 1 TS and surveillance requirements. The staff requested TVA confirm that the two EDGs credited for WBN Unit 2 are covered under the guidelines of RG 1.155, Section 1.2, or NUMARC 87-00, Revision 1, Appendix E, to maintain their target reliability of 0.975. In a letter dated December 6, 2010, TVA stated "The reliability goal for WBN and SQN EDGs is 0.975 in accordance with Station Blackout Analysis. Current reliability data indicates that the EDGs are meeting that goal." The staff finds this response acceptable.

8.4.7 Scope of Staff Review

The SBO Rule requires licensees to submit a response containing specifically defined information. It also requires utilities "to have baseline assumptions, analyses, and related information used in their coping evaluations available for NRC review." The NRC staff did not perform a detailed review of the proposed hardware and procedural modifications for WBN Unit 2. However, based on the staff's review of TVA's responses to its questions, the staff has identified the following areas for focus in follow-up NRC inspections or assessments done to verify WBN Unit 2 conformance with the SBO rule:

- hardware, if necessary, and procedural modifications
- SBO procedures are in accordance with RG 1.155, Position 3.4, and NUMARC 87-00, Section 4. Operator staffing and training follow the identified actions in the procedures.
- EDG reliability program meets, as a minimum, the guidelines of RG 1.155.
- Equipment and components required to cope with an SBO are incorporated in a quality assurance program that meets the guidance of RG 1.155, Appendix A.

- Actions taken pertaining to the specific information, commitments, and affirmation noted in this SE.

8.4.8 Summary and Conclusions

Based on the information provided by TVA regarding meeting the requirements of the SBO rule, the NRC staff concludes that TVA's completed and proposed actions, processes, and procedures to address an SBO event are acceptable, pending resolution before WBN Unit 2 startup of the open items noted above in Section 8 of this SSER.

9 AUXILIARY SYSTEMS

9.1 <u>Fuel Storage Facility</u>

9.1.2 Spent Fuel Storage

The spent fuel storage pit is a shared facility for Watts Bar Nuclear Plant (WBN) Units 1 and 2. The U.S. Nuclear Regulatory Commission (NRC) staff reviewed Section 9.1.2 of Final Safety Analysis Report (FSAR) Amendment 95, dated November 24, 2009, and determined that there were no changes from the spent fuel storage pit design described in Section 9.1.2 of FSAR Amendment 92, dated December 18, 2008, which was previously reviewed by the staff. Based on the previous staff evaluation documented in NUREG-0847, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2," issued June 1982 (hereafter referred to as NUREG-0847 or the SER), and its supplements, the review of the Watts Bar Unit 1 FSAR, and the staff evaluation of the submitted changes, the staff concludes that the spent fuel storage pit meets the relevant requirements of General Design Criterion (GDC) 2 ("Design Bases for Protection against Natural Phenomena"), GDC 4 ("Environmental and Dynamic Effects Design Bases"), GDC 5 ("Sharing of Structures, Systems, and Components"), GDC 61 ("Fuel Storage and Handling and Radioactivity Control"), 62 ("Prevention of Criticality in Fuel Storage and Handling"), and GDC 63 ("Monitoring Fuel and Waste Storage") of Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," and the requirements of 10 CFR 50.68, "Criticality Accident Requirements." Therefore, the design of the spent fuel storage pit described in Section 9.1.2 remains acceptable.

Section 9.1.2 of the WBN Unit 2 FSAR describes the spent fuel storage facility, which is shared by WBN Units 1 and 2. In NUREG--0847, the staff found the original design of the shared spent fuel pool acceptable with high-density underwater storage for up to 1,312 assemblies. The staff based its conclusion on its determination that the spent fuel storage facility was in conformance with the requirements of GDC 2, 4, 61, 62, and 63 for protection against natural phenomena, missiles, pipe break effects, radiation protection, prevention of criticality, and monitoring provisions, and that the facility conformed with the guidelines of Regulatory Guide (RG) 1.13, "Spent Fuel Storage Facility Design Basis," and RG 1.29,"Seismic Design Classification," for design and protection against seismic events.

Subsequently, the NRC staff reviewed other issues related to fuel storage. In Supplemental Safety Evaluation Report (SSER) 5 to NUREG-0847, the staff noted that Tennessee Valley Authority (TVA) had contracted for the U.S. Department of Energy to receive spent fuel from WBN. In SSERs 15 and 16, the staff accepted a reduction in allowed spent fuel storage capacity to 484 assemblies because of concerns with the neutron absorber panels and other issues related to the construction of some of the fuel storage racks. By letter dated July 28, 1997 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML020780158), the NRC issued Amendment No. 6 to the WBN Unit 1 operating license, which authorized installation of new spent fuel storage racks and increased the spent fuel storage capacity to 1,610 assemblies. The NRC issued Amendment Nos. 37, 40, 48, 67, and 77 to the WBN Unit 1 operating license on February 21, 2002 (ADAMS Accession No. ML020580612), September 23, 2002 (ADAMS Accession No. ML022540925), October 8, 2003 (ADAMS Accession No. ML032880062), January 18, 2008 (ADAMS Accession No. ML073520546), and May 4, 2009 (ADAMS Accession No. ML090920506), respectively.

These amendments authorized irradiation of tritium production burnable absorber rods (TPBARs) within the WBN Unit 1 core and transfer of these irradiated TPBARs through the shared WBN spent fuel pool. In addition, Amendment No. 40 to the WBN Unit 1 operating license authorized removal of smaller racks, which decreased the allowed storage capacity of the spent fuel storage racks to 1,386 assemblies in the remaining fuel storage racks.

The NRC staff reviewed the description of the spent fuel storage pit in Amendment 100 to the WBN Unit 2 FSAR and compared it with the description in Amendment 8 to the WBN Unit 1 FSAR. The staff found the descriptions to be essentially identical. Based on prior staff evaluation documented in NUREG-0847 and its supplements, the staff's review and acceptance of amendments to the WBN Unit 1 operating license, and the staff's comparison of the WBN Unit 1 FSAR with Amendment 100 to the WBN Unit 2 FSAR, the staff concluded that the spent fuel storage pool conforms to the relevant requirements of GDC 2, 4, 5, 61, and 63 for protection against natural phenomena, missiles, pipe break effects, radiation protection, and monitoring provisions. Therefore, the design of the shared spent fuel storage pool described in Section 9.1.2 of the WBN Unit 2 FSAR is acceptable.

9.1.4 Fuel Handling System

The fuel handling system consists of the equipment needed for refueling operations. This equipment consists of the reactor component hoisting equipment, fuel handling equipment, and the fuel transfer system. For WBN Unit 2, this equipment is located in the Unit 2 reactor building and shared portions of the auxiliary building containing the fuel storage facility for WBN Units 1 and 2. Section 9.1.4 of the WBN Unit 2 FSAR provided the description of the fuel handling system. In NUREG-0847, the NRC staff found the original design of the fuel handling system, excluding some heavy load handling provisions, to be acceptable. The staff based its conclusion on its determination that the fuel handling equipment was in conformance with the requirements of GDC 2 and 61 for protection against natural phenomena and safe fuel handling, and with the guidelines of RGs 1.13 and 1.29 for crane interlocks and seismic classification. In NUREG-0847, the staff noted that TVA would submit the results of its review against the guidelines of NUREG-0612, "Control of Heavy Load at Nuclear Power Plants—Resolution of Generic Technical Activity A-36," issued July 1980 (ADAMS Accession No. ML070250180), which the NRC staff had requested in a letter dated December 22, 1980, "Control of Heavy Loads" (now identified as NRC Generic Letter 80-113 (ADAMS Accession No. ML071080219)). In SSER 3 to NUREG-0847, the staff further evaluated the control of heavy load handling activities. Pending its review of information submitted by TVA addressing conformance with Sections 5.1.1 and 5.3 (Phase I) and Sections 5.1.2 through 5.1.6 (Phase II) of NUREG-0612, the staff intended to require through a license condition that the applicant meet the guidelines of Sections 5.1.1 and 5.3 of NUREG-0612 (Phase I) before the first refueling outage.

In SSER 13 to NUREG-0847, the NRC staff reviewed the TVA's response to NUREG-0612. By letter dated July 28, 1993 (ADAMS Accession No. ML073230481), TVA provided a revised response to the guidelines of NUREG-0612 that superseded its previous submittals. The staff concluded that TVA satisfied the guidelines of Phase I of NUREG-0612. In Generic Letter 85-11, "Completion of Phase II of 'Control of Heavy Loads at Nuclear Power Plants,' NUREG-0612," dated June 28, 1985 (ADAMS Accession No. ML031150689), the NRC staff concluded that a detailed review of the Phase II responses received from licensees was not necessary. Nevertheless, the staff performed a cursory review of TVA's response to Phase II of NUREG-0612 and concluded that the Phase II guidelines had been adequately addressed. Therefore, the staff found that the heavy load handling systems conformed to the guidelines of

NUREG-0612 and were acceptable.

In Amendment 8 to the WBN Unit 1 FSAR, TVA added a new Section 3.12, "Control of Heavy Loads." Section 3.12 described the commitments related to control of heavy loads at WBN, including those applicable to the shared auxiliary building crane and shared intake pumping station pedestal crane, and the new safety basis for handling the Unit 1 reactor vessel head using the Unit 1 containment polar crane under the guidelines of Nuclear Energy Institute (NEI) 08-05, "Industry Initiative on Control of Heavy Loads," issued July 2008 (ADAMS Accession No. ML082180666). TVA qualified the Unit 1 containment polar crane as single-failure-proof equivalent under NEI 08-05, Chapter 3, "Reactor Head Lift Single Failure Proof Equivalence." The NRC staff accepted the guidelines of NEI 08-05 for implementation of the industry initiative on control of heavy loads, as documented in its safety evaluation dated September 5, 2008 (ADAMS Accession No. ML082410532).

In Enclosure 1 to its letter dated August 30, 2010 (ADAMS Accession No. ML102510580), TVA described Unit 2 conformance with guidelines for control of heavy loads. TVA stated that WBN Unit 2 would comply with the Phase I guidelines of NUREG-0612 and qualify the Unit 2 polar crane as equivalent to single-failure-proof for reactor vessel head lifts, consistent with the guidelines of NEI 08-05. TVA stated that the method of compliance with Phase I guidelines would be substantially similar to the current Unit 1 program and that a new Section 3.12 will be added to the Unit 2 FSAR that will be materially equivalent to Section 3.12 of the current Unit 1 FSAR. This is Open Item 34 (Appendix HH).

Based on the above, the staff concludes that the design and proposed operation of the WBN Unit 2 fuel handling system is acceptable. The descriptions of equipment and operating procedures used for the handling of fuel within the reactor, refueling canal, and shared spent fuel storage facilities included in Section 9.1.4 of Amendment 100 to the WBN Unit 2 FSAR were approved by the NRC staff in the SER. Also, the NRC staff accepted the WBN Unit 1 heavy load handling program based on conformance with the Phase I guidelines of NUREG-0612, as documented in SSER 13 to NUREG-0847, and TVA enhanced the WBN Unit 1 program through implementation of the NEI 08-05 guidelines. Therefore, implementation of a materially equivalent program at WBN Unit 2 and incorporation of the program information in the WBN Unit 2 FSAR is acceptable for fuel and heavy load handling activities associated with the operation of WBN Unit 2.

9.2 Water Systems

9.2.3 Demineralized Water Makeup System

The demineralized water makeup system is a common system for WBN Units 1 and 2. The NRC staff reviewed Section 9.2.3 of Amendment 97 to the WBN Unit 2 FSAR and determined that there are no substantive changes from the system described in Section 9.2.3 of Amendment 92 to the WBN Unit 2 FSAR, which was approved by the staff in NUREG-0847. Based on the prior NRC staff evaluation documented in NUREG-0847 and the staff evaluation of the submitted changes, the staff concludes that the demineralized water makeup system meets the relevant requirements of GDC 2, for protection against natural phenomena, and the guidelines of RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," and Regulatory Position C.2 of RG 1.29, for quality group and seismic classification. Therefore, the design of the demineralized water makeup system described in Section 9.2.3 of the WBN Unit 2 FSAR is acceptable.

9.2.4 Potable and Sanitary Water Systems

By letter dated August 27, 2009 (ADAMS Accession No. ML092460757), TVA submitted Amendment 94 to the WBN Unit 2 FSAR and updated Section 9.2.4, "Potable and Sanitary Water Systems." The staff reviewed the revised information and found no differences between Units 1 and 2, except as discussed below.

One change to the potable water system as described in Section 9.2.4 is that the source of potable water is the Watts Bar Utility District rather than the previously used wells with a pumping and storage tank network. The potable water system is not essential for normal operation and the safe shutdown of the nuclear reactor.

TVA submitted Amendment 98 to the WBN Unit 2 FSAR by letter dated May 7, 2010 (ADAMS Accession No. ML101340794). Section 9.2.4 of Amendment 98 described one change to the sanitary water system. The sanitary waste is now pumped to the Spring City, TN, sewage treatment facility under contract with the Spring City Waterworks.

There are no cross-connections between the potable and sanitary water systems and the potentially radioactive systems; therefore, inadvertent contamination is prevented. Section 9.3.3 of this SSER discusses protection of safety-related equipment from flooding resulting from failure of the system. Failure of the system does not affect plant safety.

Based on its review of the information provided by TVA, the NRC staff concludes that the changes to the potable and sanitary water systems described above are acceptable. Based on the above information and the staff's previous evaluation documented in the SER and its supplements, the staff concludes that the potable and sanitary water systems meet the requirements of GDC 2 for protection against natural phenomena and meet the guidance of RGs 1.26 and 1.29 on seismic and quality group classifications and are, therefore, acceptable.

9.2.6 Condensate Storage Facilities

The NRC staff documented its review of the condensate storage facilities in Section 9.2.6 of NUREG-0847. The condensate storage facilities are nonsafety-related (Quality Group D, nonseismic). The staff concluded that the condensate storage facilities were not required to maintain the reactor in safe-shutdown condition or mitigate the consequences of accidents. The staff reviewed the licensee's system description and design criteria for the components of the condensate storage facilities and found them to be consistent with the requirements of GDC 2 and the guidelines of RG 1.26 and Regulatory Position C.2 of RG 1.29. Therefore, the condensate storage facilities were acceptable.

SSER 12, issued October 1993, stated that "the two condensate storage tanks reserved 200,000 gallons of condensate for each unit's auxiliary feedwater (AFW) system. In FSAR Amendment 72, [TVA] revised this reserved amount to 210,000 gallons." The AFW pumps take suction directly from the condensate storage tanks to supply treated water to the steam generators for cooldown of the reactor coolant system. A minimum of 200,000 gallons in each tank is reserved for AFW. However, FSAR Amendment 89 reduced the value from 210,000 gallons to 200,000 gallons. At that time, the FSAR was for both Unit 1 and Unit 2, so the revision applied to both units. The reason for the change was to correct the condensate storage tank minimum reserve volume for AFW use, based on TVA's Calculation HCG-LCS-043085, Revision 4 (referenced in TVA letter dated June 3, 2010; ADAMS Accession No. ML101600477).

In SSER 21, issued February 2009, the NRC staff reviewed existing license review topics to determine whether any topics remained open or were resolved for each section of the FSAR. No open topics were identified for FSAR Section 9.2.6, "Condensate Storage Facilities." The staff reviewed proposed changes to FSAR Section 9.2.6 in recent Amendments 95 through 100 and found no proposed changes that would challenge the system design or major changes to the system description that would change the staff's conclusion in the SER.

Therefore, the staff finds that the conclusions of the SER remain valid, and that WBN Unit 2 FSAR Section 9.2.6 is acceptable.

9.3 Process Auxiliaries

9.3.1 Compressed Air System

The NRC staff's review of the compressed air system is documented in Section 9.3.1 of the SER. The compressed air system is designed to provide a reliable supply of air for pneumatic instruments, valves, and controls. The system is shared by Unit 1 and Unit 2. The system is divided into two subsystems: the station control and service air system, and the auxiliary control air system (ACAS).

The station control and service air system is not required to function during accident conditions; therefore, the system is nonsafety-related (Quality Group D, nonseismic).

The ACAS is required to achieve safe reactor shutdown and to mitigate the consequences of an accident. It comprises two fully redundant trains. It is sized and equipped so that the required air supply is provided for both units under all design-basis accident conditions. The ACAS is designed to Quality Group B and C and seismic Category I requirements and is powered by separate and independent Class 1E power sources. All piping and components of those portions of the compressed air system required for containment isolation are designed to Quality Group B and seismic Category I requirements. The components of the ACAS are located in seismic Category I, tornado-missile-protected structures above the probable maximum flood level. Therefore, the system satisfies the requirements of GDC 2 and 5 and the guidelines of RGs 1.26 and 1.29.

In SSER 21, the NRC staff reviewed existing license review topics to determine whether items remained open or were resolved for each section of the FSAR. No open items were identified for FSAR Section 9.3.1.

In its letter dated July 31, 2010 (ADAMS Accession No. ML102290258), TVA stated the following:

> Evaluations of dual unit essential air system (Auxiliary Compressed Air System (ACAS)), demands have determined that the currently installed ACAS compressors have sufficient capacity to support dual unit operation. There are

no plans to either replace the existing ACAS air compressors or to add additional compressors.

The NRC staff reviewed proposed changes to Section 9.3.1 in FSAR Amendments 95 through 100 and found no proposed changes to the system description or design that would change the staff's conclusion in the original SER.

Based on the NRC staff's review of the compressed air system for compliance with the applicable GDC, RGs, and Branch Technical Positions (BTPs), the staff concludes that the compressed air system meets the requirements of (1) GDC 2 for against natural phenomena, and (2) GDC 5 for sharing of systems and components. Additionally, the system complies with the guidelines of RG 1.26 regarding its quality group and RG 1.29 regarding seismic classification. Therefore, the staff finds that the conclusions of the original SER remain valid, and FSAR Section 9.3.1 is acceptable.

9.3.3 Equipment and Floor Drainage System

The NRC staff reviewed the equipment and floor drainage system described in Section 9.3.3 of FSAR Amendment 95 and determined that there are no substantive changes to the system. Therefore, based on the prior staff evaluation documented in NUREG-0847, the staff concludes that the equipment and floor drainage system continues to meet the relevant requirements of GDC 2 for protection against natural phenomena, of GDC 4 for protection gainst natural environmental effects (flooding), and the guidelines of RGs 1.26 and 1.29, and BTP Auxiliary Systems Branch (ASB) 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," dated July 1981, on seismic and quality group classification and flooding as a result of piping failure. Therefore, the system is acceptable.

9.3.4 Chemical and Volume Control System

The NRC staff's review of the chemical and volume control system (CVCS) and its supporting systems for system performance during normal, abnormal, and accident conditions is documented in Section 9.3.4 of the SER. The staff concluded that the design of the CVCS and its support systems meets the requirements of GDC 1 ("Quality Standards and Records"), GDC 2, GDC 14 ("Reactor Coolant Pressure Boundary"), GDC 29 ("Protection against Anticipated Operational Occurrences"), GDC 33 ("Reactor Coolant Makeup"), GDC 35 ("Emergency Core Cooling"), GDC 60 ("Control of Releases of Radioactive Materials to the Environment"), and GDC 61 and, therefore, is acceptable.

Section 9.3.4 further states that the basis for the NRC staff's acceptance was the conformance of TVA's CVCS design with the following regulations and RGs:

> (1) the requirements of GDC 1 and the guidelines of Regulatory Guide 1.26 by assigning quality group classifications to system components in accordance with the importance of the safety function to be performed; (2) the requirements of GDC 2 and the guidelines of Regulatory Guide 1.29 by designing safety-related portions of the system to seismic Category I requirements; (3) the requirements of GDC 14 by maintaining reactor coolant purity and material compatibility to reduce corrosion and thus reduce the probability of abnormal leakage, rapid propagating failure, or gross rupture of the reactor coolant pressure boundary; (4) the requirements of GDC 29 as related to the reliability of the CVCS to

provide negative reactivity to the reactor by supplying borated water to the reactor coolant system in the event of anticipated operational occurrences; (5) the requirements of GDC 33 and 35 by designing the CVCS with the capability to supply reactor coolant makeup in the event of small breaks or leaks in the reactor coolant pressure boundary and to function as part of ECCS assuming a single-failure coincident with loss of offsite power; and (6) the requirements of GDC 60 and 61 with respect to confining radioactivity by venting and collecting drainage from the CVCS components through closed systems.

In FSAR Amendments 92 through 99, TVA made major changes to the wording and format of Section 9.3.4. However, based on its review of the FSAR amendments, the NRC staff found no changes to the (1) quality group classifications, (2) seismic requirements, (3) water chemistry or materials, (4) reliability to supply borated water to the reactor coolant system, (5) capability to supply makeup in the event of small breaks or leaks and to function as part of the ECCS (emergency core cooling system), and (6) confining of radioactivity. Therefore, the staff concludes that the changes made by TVA in FSAR Amendments 92 through 99 are acceptable, and that the conclusion made by the staff in the original SER, as noted above, remains valid.

9.3.8 Heat Tracing

In SSER 21, the NRC staff reviewed existing license review topics to determine whether items remained open or were resolved for each section of the FSAR. The original SER, NUREG-0847, did not include a Section 9.3.8. As a result, SSER 21 did not include a reference to FSAR Section 9.3.8.

The heat tracing system is not explicitly covered in the SER; therefore, TVA proposed to describe the system in FSAR Section 9.3.8, "Heat Tracing." The proposed FSAR section for heat tracing includes the purpose of the system and a list of the systems that use heat tracing. TVA does not take credit for heat tracing to maintain the reactor in a safe-shutdown condition or to mitigate the consequences of accidents. The system components were designed as nonseismic, nonsafety-related. In its letter dated February 8, 2008 (ADAMS Accession No. ML080770242, non-publiicly available), TVA proposed no significant changes to the heat tracing system.

The NRC staff reviewed proposed changes to Section 9.3 in FSAR Amendments 95 through 100. No changes to the heat tracing system were proposed.

Based on its review of the heat tracing system as described in Section 9.3.8 of WBN Unit 2 FSAR Amendments 95 through 100, the NRC staff concluded that the section conforms to the guidance in RG 1.151, Revision 1, "Instrument Sensing Lines," issued July 2010, on the relevant requirements to install heat tracing for freeze protection and to prevent boric acid from precipitating out of the fluid. Therefore, the staff concludes that FSAR Section 9.3.8 is acceptable.

9.4 Other Heating, Ventilation, and Air Conditioning Systems

9.4.1 Control Room Area Ventilation System

TVA revised FSAR Section 9.4.1 in FSAR Amendment 97, dated January 11, 2010. TVA clarified that the emergency air cleanup system automatically operates upon a safety injection signal, indication of high radiation, or smoke concentrations in the building fresh air supply. Also, during nontornado operation, power is removed from tornado isolation dampers, which are located in the ductwork connected to the two fresh air intakes. The dampers' control circuits remain de-energized during all plant conditions except tornado warning to preclude the possibility of a single failure in their control circuit isolating both air intakes. Based on its review of the information provided by TVA, the NRC staff concludes that FSAR Amendment 97 meets the environmental requirements of GDC 4 because these control circuits will provide protection against the effects of the tornado depressurization when needed, thus protecting the safety-related equipment of the control room area ventilation system. Therefore, the system is acceptable.

In FSAR Amendment 97, TVA also clarified the FSAR text for the control room area ventilation system as follows:

> Both emergency pressurizing fans (100 percent redundant) are started by the same accident signal that starts the air cleanup units. The capability is provided to place either of the operating air cleanup units and the associated emergency pressurizing fans in the standby mode. The standby components start automatically in the event of a failure of the operating air cleanup unit or its emergency pressurizing fan.
>
> During non-tornado operation, power is removed from tornado isolation dampers 0-FCO-31-21 and 0-FCO-31-34, which are located in the ductwork connected to the two fresh air intakes. The dampers' control circuits remain de-energized during all plant conditions, except tornado warning, to preclude the possibility of a single failure in their control circuit isolating both air intakes.
>
> Equipment used during the flood mode operation includes the MCR air-conditioning subsystem components on Control Building Elevation 755.0 and the water chillers and the chilled water circulating pumps on Auxiliary Building Elevation 737.0. Equipment located at floor Elevation 755 of the Control Building is unaffected by the design basis flood. The water chillers and chilled water circulating pumps serving the MCR air handling units located in the Auxiliary Building at floor Elevation 737 are functional for floods up to the design basis flood level.

Based on its review, the NRC staff concluded that the clarifying information is acceptable, because the control room area ventilation system continues to meet the guidance of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (hereafter referred to as NUREG-0800 or the SRP), Section 9.4.1, "Control Room Area System." Therefore, the system continues to meet the requirements of GDC 2, 4, 19 ("Control Room"), and 60.

Based on the NRC staff's previous evaluation, as documented in NUREG-0847 and its supplements, and on the staff's evaluation of the information provided by TVA in FSAR Amendment 97, the staff concludes that the control room area ventilation system continues to meet the relevant requirements of GDC 2, 4, 19, and 60 with respect to (1) protection against natural phenomena and environmental effects, (2) adequate access and occupancy of the control room under accident conditions, and (3) control of the release of gaseous radioactive effluents to the environment. It also meets the requirements of Item III.D.3.4 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980, and continues to meet the guidelines of RG 1.26, RG 1.29, RG 1.78, Revision 1, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," and BTP ASB 3-1 for (1) the quality group and seismic classification, (2) protection against chlorine release, and (3) high- and moderate-energy pipe breaks. Therefore, the system is acceptable.

9.4.2 Fuel Handling Area Ventilation System

As described in Section 9.4.2 of the SER, the fuel handling area ventilation system is a subsystem of the auxiliary building ventilating system and serves the spent fuel pool area, penetration rooms, and the fuel, waste, and cask handling areas. The system is designed to (1) maintain acceptable environmental conditions for personnel access, operation, inspection, maintenance, and testing, (2) protect mechanical and electrical equipment and controls, and (3) limit the release of radioactivity to the environment during all weather conditions.

During accident conditions, the fuel handling area ventilation system is shut down and all environmental control is handled by the auxiliary building gas treatment system (ABGTS), which is described in SER Section 6.2.3.

TVA revised FSAR Section 9.4.2 in FSAR Amendment 97 to clarify the description of the operation of the containment vent isolation (CVI) and auxiliary building isolation (ABI) following a fuel handling accident (FHA) in the auxiliary building or in containment during refueling operations. Also, TVA changed the FSAR to reflect the relocation of the spent fuel pool accident radiation monitor above the refuel floor elevation to ensure that an FHA in the transfer canal will be detected by the monitor.

Based on its review, the NRC staff concluded that there were no substantive changes to Section 9.4.2 in FSAR Amendment 97. The changes are acceptable because the fuel handling area ventilation system, including system operations during various FHA scenarios, continues to meet the guidance of NUREG-0800, Section 9.4.2, "Spent Fuel Pool Area Ventilation System." Therefore, the system continues to meet the requirements of GDC 2, 4, 60, and 61.

Based on the above and on the NRC staff's previous evaluation, as documented in NUREG-0847 and its supplements, the staff concludes that the fuel handling area ventilation system continues to meet the relevant requirements of GDC 2, 4, 60, and 61 for (1) protection against natural phenomena, (2) environmental effects, (3) control of releases of radioactive materials to the environment, and (4) appropriate containment, confinement, and filtering systems. The staff also concludes that the system continues to meet the guidelines of RGs 1.13, 1.26, 1.29, and 1.117, "Tornado Design Classification," for design of the ventilation system for the spent fuel storage facility, quality group and seismic classification, and the effects against tornado missiles. Therefore, the system is acceptable.

9.4.3 Auxiliary Building and Radwaste Area Ventilation System

As described in Section 9.4.3 of the SER, the auxiliary building and radwaste area ventilation system consists of the building air supply and exhaust system (general ventilation), the building cooling system (chilled water), safety features equipment coolers, the shutdown board room air conditioning system, the auxiliary board and battery room air conditioning system, the shutdown transformer room ventilation system, and the miscellaneous ventilation and air conditioning system.

In FSAR Amendment 92, TVA stated the following:

> The fifth vital battery room exhaust fans also have dampers capable of withstanding pressure differentials imposed by tornado conditions. The dampers are mounted below the Elevation 786.0 between the ceiling and the in-line fan.
>
> The fifth vital battery room is cooled by air which is drawn from the 480 Volt Board Room 1A through an opening in the common partition wall at the "T" Line and is exhausted directly to the outside. This configuration is similar to that of [the four battery rooms], with the exception that the exhaust fans are in-line axial fans and are located in the room. The cooling system is designed to maintain temperatures in this room within the range of 50°F to 104°F, and for continuous venting of hydrogen gas.

In FSAR Amendment 97, TVA revised as follows the description of the auxiliary building miscellaneous ventilation and air conditioning systems concerning main steam valve vault ventilation exhaust airflow:

> The miscellaneous ventilation and air-conditioning systems do not perform a safety function, however, the system components are designed to seismic category I(L) as necessary for the protection of safety related features.
>
> The main steam valve vault ventilation exhaust airflow is regulated to maintain an adequate temperature environment for the main steam safety valves. During low temperature conditions, the exhaust fans are shutdown and electric heating is provided. The ambient temperature in the valve vault is periodically monitored in accordance with the Technical Requirements Manual area temperature monitoring program.

In FSAR Amendments 92 and 97, TVA clarified the text about the shutdown transformer room ventilation systems concerning new loss-of-coolant accident (LOCA) temperatures for the auxiliary building 480-volt (V) transformer and heating, ventilation, and air conditioning rooms as follows:

> The shutdown transformers, located on elevation 772.0, are divided into two subareas with seven transformers in each subarea. These subareas are further divided into two enclosed areas with Train A emergency power routed to one transformer grouping and Train B emergency power to the other. [Amendment 92]
>
> When the outside air temperature decreases, exhaust fans in the individual transformer rooms are deactivated in staged series as determined by thermostatic control. As the room temperature increases above the predetermined control point, all exhaust fans are again activated in staged series. The shutdown transformer rooms' air is exhausted by electric motor-driven centrifugal-type roof ventilator fans. [Amendment 92]
>
> This ventilation system is designed to maintain the temperature in the transformer rooms within the range 19°F minimum and 110°F for which the

equipment is environmentally qualified. [Amendment 97]

In FSAR Amendment 97, TVA revised as follows the description of the building air supply and exhaust systems (general ventilation) concerning the CVI and ABI following an FHA in the auxiliary building or containment during refueling operations:

> During periods of high radiation in the fuel handling area or upon initiation of a containment isolation signal, or for high air temperature at the supply intake the Auxiliary Building supply and exhaust fans and the fuel handling exhaust fans are automatically stopped and isolation dampers located in the ducts that penetrate the Auxiliary Building Secondary Containment Enclosure (ABSCE) are closed. Additionally, during refueling operations when containment and/or the annulus is open to the Auxiliary Building ABSCE space, a Containment Vent Isolation (CVI) signal will automatically stop the above described fans and close the same isolation dampers as described above. Similarly, the high radiation signal in the fuel handling area can also automatically initiate a CVI during refueling operations when containment and/or the annulus is open to the Auxiliary Building ABSCE spaces.
>
> Ventilation supply air is heated or cooled at the air intake, as needed, to maintain suitable temperatures in the Auxiliary Building general spaces, for equipment protection and personnel comfort during normal operation.
>
> In the event of a fuel-handling accident, radiation monitors in the vicinity of the spent fuel pool initiate an Auxiliary Building isolation (ABI) signal which stops the building ventilation system and starts the ABGTS fans (see Sections 9.4.2 and 6.2.3). An ABI signal can also be initiated manually. In addition, during fuel handling operations when the containment and/or the annulus is open to the Auxiliary Building ABSCE spaces, a high radiation signal from the spent fuel pool radiation monitors will result in a containment ventilation isolation (CVI) in addition to an Auxiliary Building isolation and ABGTS start. Further, a CVI signal, including a CVI signal generated by a high radiation signal from the containment purge air exhaust radiation monitors, will initiate an Auxiliary Building isolation and start of ABGTS. These actions will ensure proper operation of the ABSCE.

Based on its review, the NRC staff concluded that the changes to Section 9.4.3 in FSAR Amendments 92 and 97 were not substantive or were editorial in nature. The changes are acceptable because the auxiliary building and radwaste area ventilation system, including system operations during various FHA scenarios, continues to meet the intent of NUREG-0800, Section 9.4.3, "Auxiliary and Radwaste Area Ventilation System." Therefore, the system continues to meet the requirements of GDC 2, 4, and 60.

Based on the NRC staff's previous evaluation, as documented in NUREG-0847 and its supplements, and on the staff's evaluation of the information provided by TVA in FSAR Amendments 92 and 97, the staff concludes that the auxiliary building and radwaste area ventilation system continues to meet the relevant requirements of GDC 2, 4, and 60 for (1) protection against natural phenomena, (2) environmental effects, and (3) control of the release of radioactive materials to the environment. It also continues to meet the guidelines of RGs 1.26, 1.29, and 1.117 on quality group and seismic classification and the effects against tornado missiles. Therefore, the system is acceptable.

9.4.4 Turbine Building Area Ventilation System

As described in Section 9.4.4 of the SER, the turbine building ventilation system, which is classified as nonessential Quality Group D, nonsafety-related, is designed to maintain an acceptable environment for personnel and the nonessential equipment served during normal plant operations. The system is separated from safety-related plant systems and areas.

In FSAR Amendment 94, TVA proposed design bases and editorial changes to Section 9.4.4 in the form of text, figures, and tables for the turbine building area ventilation system, including radiation protection information in FSAR Section 9.4.4.1, cold weather building pressurization in FSAR Section 9.4.4.2.4, and the system safety evaluation in FSAR Section 9.4.4.3.

In FSAR Amendment 94, TVA stated the following:

> During cold weather, all supply and exhaust systems can be isolated by closing the motor operated dampers to conserve heat. However, the two supply fans serving north elevation 708.0 floor may be operated at half speed since two hot water heating coils located in the supply duct connected to each of these fans heat the incoming air. With no exhaust fan running, the operation of these two supply fans will pressurize the entire Turbine Building to prevent infiltration of cold outside air. However the very slight positive pressure within the Turbine Building at the Main Control Room Habitability Zone (MCRHZ) elevation does not challenge the MCRHZ required positive minimum pressure of +1/8 inch water gage with respect to the outdoors and adjacent areas during both normal or emergency modes of operation.

The NRC staff asked TVA to provide a detailed assessment of the proposed changes describing how WBN Unit 2 continues to meet the acceptance criteria in NUREG-0800, Section 9.4.4, Revision 2, issued July 1981.

By letter dated June 3, 2010 (ADAMS Accession No. ML101600477), TVA stated the following:

> Five out of six of the Unit 2 FSAR changes are associated with administrative or editorial corrections in text created during the 10 CFR 50.54(f) FSAR re-verification efforts conducted between 1998 and 2000. These changes were incorporated to clarify the information in FSAR Section 9.4.4 to better address NUREG-0800 in areas associated with the function and qualification of Turbine Area Ventilation Systems (TAVS). Changes which add clarifying text are simply being carried over to the Unit 2 FSAR to maintain compliance fidelity established for the Unit 1 FSAR. Changes from the 50.54(f) re-verification effort which removed text considered to be excessive detail are not incorporated in the Unit 2 FSAR in order to maintain compliance with the "Level of Detail" requirements of 10 CFR 50.34(b) and the content requirements of Regulatory Guide 1.70, Revision 3, 1978. The turbine area (all elevations) is common to both units; therefore, establishment of the area design basis encompasses both units. Pursuant to NUREG-0847, Supplement 21, the Unit 1 design basis is considered to be sufficient for use as the basis for completion of Unit 2 which is especially true of common facilities.

> The sixth issue is in the area of maintaining required minimum positive pressure of 1/8" water gauge with respect to outdoors and adjacent areas during both normal and emergency modes of operation in the Main Control Room Habitability Zone (MCRHZ) during periods where Turbine Building heating is required (Outdoor temperature is 35° Fahrenheit or less). During cold weather heating periods, certain dampers are closed with air flow directed across heating coils for maintaining interior heat. A potential exists for the Turbine Building to become pressurized with this system operating configuration which creates a challenge to the required minimum MCRHZ pressure since they share a common wall and penetrations at the "N" line. The potential for this anomaly is documented in [TVA WBN] Problem Evaluation Report (PER) 4947. As part of the corrective measures established for resolution of this PER, testing was performed to determine actual Turbine Building pressurization during worst case scenarios. This testing confirmed that the maximum turbine area pressure at the MCHRZ elevation (recorded at 0.01 inches of water) will not challenge minimum pressure requirements of the habitability zone.
>
> Additionally, normal day to day pressure of the MCRHZ is maintained at 0.4 inches of water to assure continuous compliance with minimums. [TVA] System description document (SDD) N3-44-4002 was revised to provide a detailed description of this situation. This information has been translated to the Unit 2 FSAR as reviewed under 10 CFR 50.59 for Unit 1.

Based on its review of the information provided by TVA in FSAR Amendment 94, the NRC staff concluded that there are no substantive changes to the turbine building area ventilation system as described in Section 9.4.4. Therefore, the turbine area ventilation system of WBN Unit 2 will continue to meet the intent of the acceptance criteria in NUREG-0800, Section 9.4.4, Revision 2.

Based on the NRC staff's previous evaluation, as documented in NUREG-0847 and its supplements, and on the staff's evaluation of the information provided by TVA in FSAR Amendment 94, the staff concludes that the turbine building area ventilation system continues to meet the relevant requirements of GDC 2 for protection against natural phenomena and continues to meet the guidelines of RGs 1.26 and 1.29 on quality group and seismic classification. Therefore, the system is acceptable.

9.4.5 *Engineered Safety Features Ventilation System*

As described in Section 9.4.5 of the SER, the engineered safety features (ESF) ventilation system provides a suitable and controlled environment for essential equipment and components during normal plant operating conditions and accident conditions. These components include (1) the essential raw cooling water (ERCW) pumps and ancillary equipment, (2) the diesel generator and associated components, and (3) ESF equipment such as the containment spray pumps, the residual heat removal pumps, and the centrifugal charging pumps.

In FSAR Amendment 97, TVA stated that analysis of the ERCW intake pumping station ventilation system shows the following:

> Adequate flow-through ventilation is provided for the ERCW and HPFP [high pressure fire protection] pump area by natural convection during all credible

> environmental conditions. Compensatory actions are taken during severe environmental conditions. A structural failure of the grillage roof will not prevent supply of adequate ventilation air to the pump deck.

TVA also described the operation logic of the diesel generator building exhaust fans:

> One diesel generator room exhaust fan automatically starts upon diesel startup. The second exhaust fan starts when the upper setpoint of a temperature switch mounted in the air exhaust-room is reached or on low flow of the first fan. The generator and electrical panel cooling fan can start along with either exhaust fan. The temperature switches mounted in the air exhaust room monitor the temperature of the air as it leaves the diesel generator room. Each switch may actuate its respective room exhaust fan upon detection of high diesel generator room temperature conditions or may deenergize its respective fan, as necessary, in order to maintain the diesel generator room exhaust temperature between 50°F and 120°F.

Based on its review of the information provided by TVA in FSAR Amendment 97, the NRC staff concludes that the system meets the requirements of GDC 4, because the operation logic will maintain environmental conditions for optimum equipment operation during all modes of operation, including normal, transient, and accident conditions. Therefore, the system is acceptable.

In FSAR Amendment 97, TVA also stated the following:

> During tornadoes, the essential components of the system remain functional because the components are located in a Seismic Category I structure that is designed to resist damage by tornado missiles. For tornado depressurization mitigation, intake, and exhaust dampers are opened to assist in pressure equalization.

Based on its review of the information provided by TVA in FSAR Amendment 97, the NRC staff concludes that the system meets the requirements of GDC 4, because these dampers provide protection to the safety-related equipment against the effects of the tornado depressurization. Therefore, the system is acceptable.

In addition, in FSAR Amendment 97, TVA revised the description of the ESF ventilation system to include operation following an FHA as follows (in part):

> In the event of a fuel-handling accident, radiation monitors in the vicinity of the spent fuel pool initiate an Auxiliary Building isolation (ABI) signal which stops the building ventilation system...The containment isolation Phase A signal, high radiation in the spent fuel pool area, a CVI signal when containment and/or the annulus is open to the Auxiliary Building ABSCE spaces, and high air temperature in the Auxiliary Building air intakes provide for a two-train isolation signal for the Auxiliary Building. Isolation of the general ventilation system...results in the disruption of normal airflow patterns; and thus, provide for an effectively sealed ABSCE boundary.

Based on its review of the information provided by TVA in FSAR Amendment 97, the NRC staff concludes that the system continues to meet the requirements of GDC 60 for suitable control of the release of radioactive gaseous effluents to the environment.

Based on the NRC staff's previous evaluation, as documented in NUREG-0847 and its supplements, and on the staff's evaluation of the information provided by TVA in FSAR Amendment 97, the staff concludes that the ESF ventilation system meets the relevant requirements of GDC 2, 4, and 60 for protection against natural phenomena and missiles and continues to meet the guidance of RGs 1.26 and 1.29 for quality group and seismic classification and the effects against tornado missiles. Therefore, the system is acceptable.

9.4.6 Reactor Building Purge Ventilation System

In Section 9.4.6 of FSAR Amendment 97, TVA described the design and operation of the reactor building purge ventilation system (RBPVS). The NRC staff reviewed the information provided in FSAR Amendment 97 to confirm that the design meets the relevant requirements of GDC 2, 4, 60, and 61 for protection against natural phenomena, environmental effects, and control of releases of radioactive materials to the environment.

The RBPVS is designed to maintain the environment in the primary and shield building annulus within acceptable limits for equipment operation and for personnel access during inspection, testing, maintenance, and refueling operations. The RBPVS also provides a filtration path for any through-duct outleakage from the primary containment to limit the release of radioactivity to the environment. Based on its review, the NRC staff determined that the FSAR changes to the RBPVS system were nonsubstantive or editorial.

TVA clarified the FSAR description of the CVI and ABI following an FHA in the auxiliary building or containment during refueling operations. Also, TVA added a description of the containment vent air cleanup units, which filter the containment vent air before it is released into the annulus. The NRC staff reviewed TVA's changes to the FSAR and concludes that the changes are acceptable because the RBPVS operations during various FHA scenarios continue to meet the requirements of GDC 2, 4, 60, and 61 for protection against natural phenomena, environmental effects, and control of releases of radioactive materials to the environment.

9.4.7 Containment Air Cooling System

The NRC staff reviewed Section 9.4.7, "Containment Air Cooling System," of FSAR Amendment 97 to ensure that the system meets the relevant requirements of GDC 2, 4, and 60 for protection against natural phenomena, environmental effects, and control of releases of radioactive materials to the environment.

On the basis of its review, the NRC staff concluded that the FSAR changes to the description of the air cooling system were nonsubstantive or editorial changes. The containment air cooling system consists of the lower compartment air cooling system, the control rod drive mechanisms (CRDM) building cooling system, the upper compartment air cooling system, and the reactor building instrument room air cooling system. In FSAR Amendment 97, TVA revised the description of the containment air cooling systems to clarify conflicting requirements for the CRDM coolers and the lower compartment coolers as follows:

> The lower compartment cooling (LLC) [sic] air system manual dampers are adjusted to provide sufficient air flow through the reactor well to maintain an

approximate air temperature of 120°F. The system is designed for two of four units to recirculate air through the lower containment and equipment compartments, although all four may be operated, anytime there is a loss of normal containment cooling following any non-LOCA design basis event which results in a hot standby condition. The LCC system is not required to operate after a LOCA.

After a MSLB [main steamline break], all four LCCs [lower compartment cooling units] are started (only two are required) within 1.5 to 4 hours after the MSLB, to recirculate air throughout the lower compartment spaces to prevent hot spots from developing. This is a safety function.

In FSAR Amendment 97, TVA clarified the description of the LCC air system as follows:

For Unit 1 only, LCC 1D-B has a reduced cooling capacity of approximately 12.5% due to isolation of ERCW to one of its eight cooling coils.

Connections are available in the A-Train ERCW supply and return headers for the lower compartment coolers that will allow chilled water from a non-safety related chiller to be used to provide additional cooling of the Unit 1 Reactor Building during outages.

Based on its review of FSAR Amendment 97 and the staff's previous evaluation, as documented in the SER and its supplements, the NRC staff concludes that the containment air cooling system is acceptable because the system continues to meet the requirements of GDC 2, 4, and 60 for protection against natural phenomena, environmental effects, and control of releases of radioactive materials to the environment.

9.4.8 Condensate Demineralizer Waste Evaporator Building Environmental Control System

In FSAR Amendment 94 (ADAMS Accession No. ML092460757), TVA proposed design-basis changes for cold weather building pressurization in FSAR Section 9.4.8, "Condensate Demineralizer Waste Evaporator Building Environmental Control System." TVA clarified in Amendment 94 that the condensate demineralizer waste evaporator (CDWE) building environmental control system (ECS) is a separate, nonsafety air conditioning system that is not required for Unit 2 operation. The CDWE building is inside the auxiliary building secondary containment enclosure (ABSCE) boundary; therefore, it is connected to the auxiliary building ventilation exhaust system. The ventilation exhaust system provides a negative pressure inside the CDWE building.

Based on its review, the NRC staff concluded that there were no substantive changes in Section 9.4.8 of FSAR Amendment 94 except to clarify that negative pressure inside the CDWE building is provided by connecting the system to the auxiliary building ventilation exhaust system. The staff asked TVA to provide a detailed assessment of the proposed changes that addressed the acceptance criteria in NUREG-0800, Section 9.4.3, Revision 2, issued July 1981.

By letter dated June 3, 2010 (ADAMS Accession No. ML101600477), TVA stated the following:

The CDWE Building environmental control system is no longer required for either Unit 1 or Unit 2 due to the fact that the Condensate Demineralizer Waste

Evaporator (CDWE) was abandoned in favor of Mobile Demineralizers provided by vendors. The physical space of the CDWE Building (CDWEB) is considered, therefore, a clean environment since no activity which would tend to produce radioactivity will be accomplished in the space. The space is connected to the Auxiliary Building Secondary Containment Enclosure (ABSCE) through exhaust ducting and a set of airlock doors in the tunnel between the waste packaging area and the CDWEB. The ventilation system is configured such that there is no supply air to the CDWEB (the supply fan and associated ducting are isolated by two closed and abandoned fire dampers and the fan is disabled).

The fuel handling area exhaust fans take suction from the CDWEB during normal operation and the area is maintained at a slight negative pressure relative to the outside environment. During accident conditions, the fuel handling area exhaust is isolated and the Auxiliary Building Gas Treatment System (ABGTS) takes suction from the same exhaust ductwork. Exhaust from the ABGTS is filtered and exhausted through the Shield Building exhaust stack. The exhaust function is not considered to be part of the CDWEB Environmental Control System (ECS) as it does nothing to moderate building temperature or air quality. The exhaust air flow creates a negative pressure condition in the CDWE space during both normal and post accident conditions.

The CDWE Building is self contained with external walls and roof not shared with any other features which, when combined with negative pressurization, precludes transportation of potentially contaminated air to the environment. The abandoned CDWEB ECS components within the CDWEB, such as air coolers and space heaters, are not needed to control temperature since the original cooling loads are no longer present. Doors are treated as ABSCE boundary controlled access and egress points which must have clearance authorization from operations prior to opening.

Because the CDWE building ECS is a nonsafety-related separate system and is not required for operation of either WBN Unit 1 or 2, the NRC staff concludes that the proposed changes in FSAR Amendment 94 to the WBN Unit 2 CDWE continue to meet the acceptance criteria in NUREG-0800, Section 9.4.3, Revision 2.

Based on the NRC staff's previous evaluation, as documented in NUREG-0847 and its supplements, and on the staff's evaluation of the information provided by TVA in FSAR Amendment 94, as supplemented by letter dated June 3, 2010, the staff concludes that the CDWE building ECS meets the relevant requirements of GDC 2 and 4 for protection against natural phenomena and environmental effects and missiles and continues to meet the guidelines of RGs 1.26, 1.29, and 1.117 on quality group and seismic classification and the effects against tornado missiles. Therefore, FSAR Section 9.4.8 is acceptable.

9.5 Other Auxiliary Systems

9.5.3 Lighting System

The lighting system for a nuclear power plant provides adequate lighting during all plant operating conditions (e.g., normal operation and anticipated fire, transient, and accident conditions). This section includes design criteria, provisions for lighting in areas required for firefighting and provisions for lighting needed in areas for control and maintenance of safety-related equipment and in access routes to and from these areas. The NRC staff reviewed the lighting systems with respect to the capability of the normal, standby, and emergency lighting systems to provide adequate lighting during all plant operating conditions, including fire, transients, and accident conditions, and the effect of the loss of normal power on the emergency lighting system.

Section 9.5.3 of the WBN Unit 2 FSAR, "Lighting System," describes the lighting systems as follows:

> There are three basic lighting systems in the plant designated as follows: normal, standby, and emergency. These systems are designed in accordance with TVA design guides and standards which use the recommendations of the Illuminating Engineering Society of North America [IESNA] as their basis, and good engineering practice to provide the required illumination necessary for safe conduct of plant operations and under normal conditions to make the plant personnel as comfortable as possible.
>
> All plant lighting systems have the following features in common: adequate capacity and rating for the operation of the loads connected to the systems, independent wiring and power supply, overcurrent protection for conductor and equipment using nonadjustable inverse time circuit breakers, copper conductor with 600-volt insulation run in metal raceways.

The normal lighting system is designed to provide sufficient illumination to meet normal plant operations and maintenance requirements. The normal lighting is for general-purpose lighting and is powered by two alternate feeders from the 6.9 kilovolt (kV) common boards A and B via three-phase 6900-120/208-V transformers. The normal lighting boards are located in the turbine and auxiliary buildings of the main plant. In the main control room (MCR), alternate rows of fixtures or alternate fixtures are fed from different lighting boards to prevent total blackout in a particular area in case of failure of one of the lighting boards or cabinets.

Upon loss of the normal lighting system, the standby lighting system provides adequate illumination for the safe shutdown of the reactor and the evacuation of personnel from vital areas. The standby lighting system forms an integral part of the normal lighting system and is normally energized at all times. The standby lighting system is fed from an independent source. This system is fed from 480-V reactor motor-operated valve boards via 480-120/208-V transformers to each standby lighting cabinet. The reactor motor-operated valve boards have a normal and alternate alternating current (ac) power supply and, in the event of their failure, are fed from the standby diesel generators. The cable feeders to the standby cabinets are located in the seismic Category I structure and are routed in redundant raceways. The fixtures are dispersed among the normal lighting fixtures.

The emergency lighting system is composed of two separate systems: (1) the 125-V direct current (dc) lighting system, which is designed to provide immediately the minimum illumination level in areas vital to the safe shutdown of the reactor for the period required for diesel loading or, upon loss of ac auxiliary power, for the duration of the capacity of the 125-V vital dc batteries, and (2) an individual 8-hour battery pack network, which is used to supplement the 125-V dc emergency lighting to provide emergency lighting in areas that must be manned for safe shutdown, and for access and egress to and from fire areas, which meets the requirements of Section III.J of Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," to 10 CFR Part 50. Other battery pack units are provided for building egress for personnel safety purposes.

As described in Section 9.5.3.3 of the FSAR, the diesel generator building lighting cabinets are fed through 480-208/120-V three-phase lighting transformers that are fed from the diesel 480-V auxiliary boards. Each of these auxiliary boards has dual feeders from the 480-V shutdown boards. Each diesel generator unit has a lighting cabinet that supplies the normal lighting for that unit. Low-level lighting required for maintenance or operating procedures and ingress and egress in the event of loss of normal lighting is supplied by fixtures with a self-contained battery and inverter charger and also by individual 8-hour battery pack lighting units.

Section 9.5.3.4 of the WBN Unit 2 FSAR describes the safety-related functions of the lighting system as follows:

> The lighting system is adequate for the operation and evacuation of the plant to the extent that the supports for the components of the system, that are located in areas of Seismic Category I structures containing safety-related equipment are qualified to prevent failure that could impair the functioning of any safety-related plant feature. Lighting systems are classified as non-safety related. However, due to their functions, standby and emergency lighting systems shall be of a high reliability design so as to ensure necessary illumination in areas of the plant needed for operation of safe shutdown equipment and in access and egress routes thereto.

The NRC staff's acceptance criteria in SRP Section 9.5.3 state that the acceptability of the normal, standby, and emergency lighting is based on the degree of similarity of the systems designed to those of previously reviewed plants with satisfactory operating experience. There are no general design criteria or RGs that directly apply to the plant lighting system. However, the lighting levels recommended in the 1981 edition of the *IESNA Lighting Handbook* were adopted by the NRC in NUREG-0700, Revision 0, "Guidelines for Control Room Design Review," issued September 1981, as guidance for providing the appropriate illumination levels in the control room.

The NRC staff reviewed Section 9.5.3 of WBN Unit 2 FSAR Amendment 94 to evaluate the acceptability of the normal, standby, and emergency lighting with respect to the above regulatory guidance.

Section 9.5.3 of the NRC staff's SER (NUREG-0847) states the following:

> The lighting system for Watts Bar is designed to provide adequate lighting in all areas of the station and consists of normal and standby ac lighting systems and an emergency dc lighting system. The design is based on illumination levels that equal or exceed those recommended by the IESNA Lighting Handbook for central stations.
>
> The normal lighting system is for general lighting of the plant. The major power supply is through two alternate feeders from the 6.9-kV common boards A and B to selective and interrupter switch and 3-phase 6900-120/208-V ac transformers, feeding a lighting board. These lighting boards are located in the turbine and

auxiliary building of the main plant. Other lighting boards in the service building, office building, gatehouse, and so forth, are fed from 480-V boards through 3-phase 480-120/208-V ac transformers. These lighting boards feed the normal lighting cabinets distributed throughout the main plant. In the power boards and control rooms, alternate rows of fixtures or alternate fixtures are fed from different lighting boards to prevent total blackout in a particular area in case of failure of one or the other lighting boards or cabinets.

The standby lighting system forms a part of the normal lighting requirements and is normally energized at all times. This system is fed from 480-V control and auxiliary building vent boards to 3-phase 480-120-V ac transformers to each standby lighting cabinet. The control and auxiliary building vent boards have a normal and alternate ac power supply and, in case of the failure of these supplies, the vent boards are fed from the diesel generators. The cable feeders to the standby cabinets located in the seismic Category 1 structure are routed in redundant raceways, and the fixtures are dispersed among the normal lighting fixtures.

The third lighting system is the emergency system. The feeder to this system is electrically held in the off position until a power failure occurs on the ac systems. Then the emergency lighting cabinets are automatically energized from the 125-V dc vital battery boards. This system is an essential supporting auxiliary system for the ESF, and the cable feeders to the lighting cabinets are routed in the redundant ESF cable tray system or in conduit. The fixtures are incandescent type and are dispersed among the normal and standby fixtures with alternate emergency fixtures being fed from redundant power trained lighting cabinets. Each diesel generator unit has a lighting cabinet which supplies the normal lighting for that unit. The diesel generator building lighting cabinets are fed through 480-208/120-V 3-phase local lighting transformers, which in turn are fed from the associated diesel generator 480-V shutdown boards. The diesel generator will start within the prescribed time to provide the 480-V ac power requirements to the 480-V shutdown boards, thus supplying power again to the diesel generator building lighting transformers. Low-level lighting required for maintenance or operating procedures and ingress/egress in the event of loss of normal lighting is supplied from fixtures with a self contained battery and inverter charger.

The plant lighting systems are designed so that a single failure cannot degrade essential lighting below a safe level. The plant lighting systems are tested at installation and provisions are installed for testing the dc emergency system.

The scope of the review of the lighting system for WBN Unit 2 included assessment of all components necessary to provide adequate lighting during both normal and emergency operating conditions, the adequacy of the power sources for the normal and emergency lighting systems, and verification of the lighting system's functional capability to provide the appropriate illumination levels under all conditions of operation. Based on operating experience and the degree of similarity to previously reviewed plants, the particular areas of interest for proper illumination levels are the MCR, remote shutdown consoles, and at the safety-related panels for interior maintenance.

Section 9.5.3 of the WBN Unit 2 FSAR does not provide information on the typical luminance ranges for normal lighting in all areas and rooms of the plant. In particular, the FSAR does not contain a description of lighting illumination levels for the MCR, remote shutdown consoles, and at the safety-related panels for interior maintenance. The NRC staff asked TVA to provide information on illumination levels for these areas in order to determine the adequacy of the lighting system to provide the appropriate illumination levels under all conditions of operation. By letter dated July 31, 2010 (ADAMS Accession No. ML102290258), TVA stated that the luminance for normal and emergency lighting in all areas and rooms of the plant is delineated in the TVA design standard for lighting standards and practices. The design standard is based on the 1993 edition of the *IESNA Lighting Handbook* and Appendix E-2 to NUREG-0700, Revision 0, September 1981. The design illumination levels conform to these references. Average illumination levels in footcandles for various plant areas and rooms are based on these references. The following are the normal lighting illuminance levels for the areas and rooms:

- aisles, corridors, and stairways: 10–20 footcandles
- auxiliaries, pumps, tanks, and compressors: 20 footcandles
- battery and battery board room: 20 footcandles
- communications room: 40 footcandles
- conference room: 30–50 footcandles
- equipment rooms, mechanical, and miscellaneous: 75–100 footcandles

The NRC staff reviewed the illuminance levels provided by TVA and concluded that the illuminance levels are in conformance with the guidance in the 1993 edition of the *IESNA Lighting Handbook* and NUREG-0700, Revision 0. Therefore, the luminance for normal and emergency lighting in all areas and rooms of the plant is acceptable.

With regard to the illumination levels for the work areas or types of tasks in the MCR, safety-related panels in the MCR, and remote shutdown consoles, TVA stated that the WBN Units 1 and 2 MCR and auxiliary control room (ACR) lighting system was modified in the 1989–1991 timeframe. TVA made these modifications to comply with the requirements for minimum average illumination for the MCR and the ACR based on the following documents:

- *IESNA Lighting Handbook*, 1981 application volume and 1981 reference volume

- NUREG-0700, Revision 0, Sections 6.1.5.3 and 6.1.5.4 and Appendix E-2

- TVA Design Standard DS-E17.1.1, Revision 2, "Lighting Design Standards and Practices"

The following are the normal lighting luminance levels for the work areas in the MCR and ACR:

- vertical face of switchboard (66 inches above the floor), 20–50 footcandles
- benchboard (horizontal level), 20–50 footcandles
- rear of the switchboard (vertical 60 inches above floor), 10 footcandles
- unit operators' desk, 50-100 footcandles

After the modifications were implemented, TVA conducted a survey of both the MCR and the ACR to determine the actual illumination levels and to ascertain whether the acceptance criteria were met. TVA's survey analysis concluded that the illumination levels meet the acceptance criteria given in the documents listed above.

The NRC staff reviewed the illuminance levels in the MCR and ACR as provided by TVA and concluded that the illuminance levels in the MCR and ACR are in conformance with the guidance given in the 1993 edition of the *IESNA Lighting Handbook* and NUREG-0700, Revision 0. Therefore, the luminance in the MCR and ACR areas is acceptable.

The NRC staff also asked TVA to discuss whether the emergency lighting in the MCR, safety-related panels in the MCR, and remote shutdown consoles provides illumination levels in these areas equal to or greater than those recommended by the *IESNA Lighting Handbook* or NUREG-0700 for at least 8 hours. In its letter dated July 31, 2010, TVA stated that the WBN Units 1 and 2 standby and emergency lighting in the MCR and remote shutdown consoles meet the requirements of the *IESNA Lighting Handbook* for central stations or of NUREG-0700. The lighting is designed to provide illumination levels for (1) standby lighting (all tasks, each train) of 10 footcandles and (2) emergency lighting of 3 footcandles, and as provided in Table 9.5.3-1 below:

Table 9.5.3-1 Acceptance Criterion for the Task Area Luminance Ratio

Area	**Luminance Ratio**
Task area versus adjacent darker surroundings	3:1
Task area versus adjacent lighter surroundings	1:3

Based on the information provided by TVA, the NRC staff concludes that the illuminance levels for emergency lighting in the MCR, safety-related panels in the MCR, and remote shutdown consoles conform to the guidance given in the 1993 edition of the *IESNA Lighting Handbook* and NUREG-0700 and are, therefore, acceptable.

Based on its review of the information provided by TVA, the NRC staff concludes that (1) the plant lighting systems described in Section 9.5.3 of the WBN Unit 2 FSAR conform to the industry standard *IESNA Lighting Handbook*, NUREG-0700, and the acceptance criteria of SRP Section 9.5.3, and (2) the systems can perform their safety-related functions. Therefore, the plant lighting systems are acceptable.

9.5.4 Emergency Diesel Engine Fuel Oil Storage and Transfer System

The NRC staff reviewed the emergency diesel engine fuel oil storage and transfer system as described in FSAR Amendment 95 and found no differences between WBN Units 1 and 2. Therefore, based on its review of FSAR Amendment 95 and the previous staff evaluation, as documented in NUREG-0847 and its supplements, the staff concludes that the emergency diesel fuel oil storage and transfer system meets the relevant requirements of GDC 2, 4, 5, and 17, "Electric Power Systems," and is acceptable.

9.5.6 Emergency Diesel Engine Starting Systems

The NRC staff reviewed the emergency diesel engine starting system as described in FSAR Amendment 95 and found no differences between WBN Units 1 and 2. Therefore, based on its review of FSAR Amendment 95 and the previous staff evaluation, as documented in NUREG-0847 and its supplements, the staff concludes that the emergency diesel engine starting system meets the relevant requirements of GDC 2, 4, 5, and 17 and is acceptable.

9.5.7 Emergency Diesel Engine Lubricating Oil System

The NRC staff reviewed the emergency diesel engine lubricating oil system as described in Section 9.5.7 of FSAR Amendment 94 and determined that there are no substantive changes to the information previously reviewed. Therefore, based on its review of FSAR Amendment 94 and the previous staff evaluation, as documented in NUREG-0847 and its supplements, the staff concludes that the emergency diesel engine lubricating oil system meets the relevant requirements of GDC 2, 4, 5, and 17 and is acceptable.

9.5.8 Emergency Diesel Engine Combustion Air Intake and Exhaust System

The NRC staff reviewed the emergency diesel engine combustion air intake and exhaust system described in Section 9.5.8 of FSAR Amendment 94 and determined that there are no substantive changes to the information previously reviewed. Therefore, based on its review of FSAR Amendment 94 and the previous staff evaluation, as documented in NUREG-0847 and its supplements, the staff concludes that the emergency diesel engine combustion air intake and exhaust system meets the relevant requirements of GDC 2, 4, 5, and 17 and is acceptable.

10 STEAM AND POWER CONVERSION SYSTEM

10.2.1 Turbine Generator Design

In Section 10.2.1 of NUREG-0847, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2," issued June 1982 (hereafter referred to as NUREG-0847 or the SER), the U.S. Nuclear Regulatory Commission (NRC) staff reviewed the Watts Bar Nuclear Plant (WBN) turbine generator design. The turbine generator converts steam power into electrical power and has a turbine control and overspeed protection system.

The design function of the turbine control and overspeed protection system is to minimize the probability of generation of turbine missiles. The turbine generator control and overspeed protection system is not required to maintain the reactor in safe-shutdown condition or mitigate the consequences of accidents; however, the staff's evaluation identified the turbine control and overspeed protection system as essential for the overall safe operation of the plant.

The NRC requires that structures, systems, and components important to safety be appropriately protected against the effects of missiles that could result from equipment failures in accordance with General Design Criterion (GDC) 4, "Environmental and Dynamic Effects Design Bases," of Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities." The NRC provides additional guidance in Regulatory Guide (RG) 1.115, "Protection against Low-Trajectory Turbine Missiles." The licensee proposed preoperational and startup tests of the turbine generator in accordance with RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants." The staff evaluates the adequacy of the test program in Chapter 14 of this report. The staff evaluated the turbine generator overspeed protection system for compliance with the recommendations of Branch Technical Position (BTP) Auxiliary Systems Branch (ASB) 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," dated July 1981, and BTP Mechanical Engineering Branch (MEB) 3-1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," dated July 1981, for protection against cracks and breaks in high-energy and moderate-energy piping systems. The staff evaluated protection against the dynamic effects associated with the postulated pipe system failure in SER Section 3.6. Based on the design of the turbine generator system's conformance with the design criteria and bases for the prevention of the generation of turbine missiles, the NRC staff found the design of the turbine generator system to be acceptable as documented in the SER.

The NRC staff reviewed changes that the Tennessee Valley Authority (TVA) made to Section 10.2.1 of the SER in Final Safety Analysis Report (FSAR) Amendments 95 through 100. TVA made no changes that would affect the staff's conclusions in the SER.

Based on its review, the NRC staff concludes that the description of the turbine generator system in FSAR Section 10.2.1 continues to conform to the above requirements and guidance, and that the system can perform its function as designed. Therefore, the staff finds the conclusions of the SER to remain valid, and FSAR Section 10.2.1 is acceptable.

10.3 Main Steam Supply System

FSAR Section 10.3.1, "Main Steam Supply System (Up to and Including the Main Steam Isolation Valves)," evaluates the safety-related portion of the main steam system, up to and including the main steam isolation valves (MSIVs). FSAR Section 10.3.2, "Main Steam Supply System," evaluates the nonsafety-related portion of the main steam system downstream of the MSIVs, up to and including the turbine stop valves.

10.3.1 Main Steam Supply System (Up to and Including the Main Steam Isolation Valves)

In the SER, the NRC staff reviewed FSAR Section 10.3.1. Within the scope of the SER, the staff reviewed the main steam supply system up to the isolation valves for compliance with the applicable GDC, RGs, and BTPs. The staff concluded that the system design conforms to (1) the requirements of GDC 2, "Design Bases for Protection against Natural Phenomena," and GDC 4 for protection against natural phenomena and environmental effects and missiles, (2) the requirements of GDC 34, "Residual Heat Removal," for residual heat removal capability and the single failure criterion, and (3) the guidelines of RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," RG 1.29, "Seismic Design Classification," RG 1.117, "Tornado Design Classification," and BTP ASB 3-1 for quality group and seismic classification, the effects of tornado missiles, high-energy pipe breaks, and the MSIV closure time position. Therefore, the staff found FSAR Section 10.3.1 to be acceptable.

In Supplemental Safety Evaluation Report (SSER) 19, the NRC staff reviewed and accepted changes to FSAR Section 10.3.1. In FSAR Amendment 91, TVA revised the MSIV closing time to 6 seconds. In Section 10.3.1 of SSER 19, the staff stated that the MSIVs were capable of closing in 5 seconds after receipt of a closure signal. The staff reviewed the results of TVA's sensitivity study and concluded that a total time of 8 seconds for main steamline isolation, including generation, processing, and delay of the isolation signal and valve closure, was acceptable for all of the related design-basis accidents. Therefore, the staff concluded that TVA's revised maximum closure time of 6 seconds for the MSIVs was within the design acceptable limits and was acceptable.

In SSER 21, the NRC staff reviewed existing license review topics to determine whether the topics remained open or were resolved for each section of the FSAR. No open topics were identified for FSAR Section 10.3.1.

The NRC staff reviewed changes to Section 10.3.1 that TVA made in FSAR Amendments 95 through 100. TVA did not identify any significant changes to the main steam system up to the isolation valves and did not make any changes to the safety function provided by the main steam system up to the isolation valves that would change the staff's conclusion in the SER.

Based on its review, the NRC staff concludes that FSAR Section 10.3.1 continues to comply with the applicable GDC, RGs, and BTPs as evaluated in SER, and that the conclusions of the SER remain valid.

10.3.2 Main Steam Supply System (Downstream of the Main Steam Isolation Valves)

In the SER, the NRC staff reviewed FSAR Section 10.3.2. This section of the FSAR included the main steam system downstream of the MSIVs. The staff concluded that this portion of the main steam system was not required to maintain the reactor in safe-shutdown condition or mitigate the consequences of accidents and, therefore, was nonsafety-related. The system components were designed to the requirements of American National Standards Institute

(ANSI) B31.1, "Power Piping," and are nonseismic. Based on its review, the staff concluded that FSAR Section 10.3.2 was acceptable.

The NRC staff reviewed changes that TVA made to Section 10.3.1 in FSAR Amendments 95 through 100. TVA did not identify any significant changes to the main steam system downstream of the isolation valves and did not make any changes to the safety function provided by the main steam system downstream of the isolation valves that would change the staff's conclusion in the SER.

Therefore, the NRC staff finds that the conclusions of the SER remain valid for the main steam supply system downstream of the MSIVs described in FSAR Section 10.3.2.

10.3.3 Steam and Feedwater System Materials

By letter dated November 24, 2009, TVA submitted FSAR Amendment 95, which included information to support the NRC staff's review of SER Section 10.3.3, "Steam and Feedwater System Materials," for WBN Unit 2. The submittal included fracture toughness, materials selection, and fabrication requirements used in the steam and feedwater system components. For this section of the current safety evaluation, the NRC staff reviewed Section 10.3.6 of FSAR Amendment 95, which is related to materials used for the steam and feedwater system components. In conducting its review, the staff referred to the relevant requirements identified in GDC 1 and Section 10.3.6 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (hereafter referred to as NUREG-0800 or the SRP).

For the NRC staff's review of Section 10.3.6, the applicable criterion is GDC 1, "Quality Standards and Records." GDC 1 states, in part, that "structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed." GDC 1 also requires that "appropriate records of the design, fabrication, erection, and testing of structures, systems, and components...be maintained...throughout the life of the unit."

In Staff Requirements Memorandum SECY-07-096, "Possible Reactivation of Construction and Licensing Activities for the Watts Bar Nuclear Plant Unit 2," dated July 25, 2007, the Commission directed the NRC staff to use the current licensing basis for WBN Unit 1 as the reference basis for the review and licensing of WBN Unit 2. Therefore, the NRC staff's review of Section 10.3.6 of WBN Unit 2 FSAR Amendment 95 focused on resolving variations between the WBN Unit 1 and Unit 2 baseline FSAR and items found in the SER and SSERs that are applicable to this section. In the SER, the staff approved the fracture toughness, materials selection, and fabrication requirements used in the steam and feedwater system components for WBN Units 1 and 2. In FSAR Amendment 95, TVA modified the materials selection and fabrication requirements documented in FSAR Section 10.3.6.

TVA modified the requirements in order to allow the use of low-alloy steels in the steam and feedwater systems. The use of low-alloy steels provides better resistance to the degradation mechanism of flow-accelerated corrosion (FAC). FAC has been detected on WBN Unit 1, and, in some locations, the original carbon steel piping has been replaced with a more corrosion-resistant chrome-molybdenum low-alloy steel. Because of the use of low-alloy steels, RG 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," is now applicable. The changes in Section 10.3.6 of FSAR Amendment 95 would allow this corrosion-resistant material to be used in WBN Unit 2. The changes describe the application of RG 1.50 to the steam and feedwater systems. The changes in Section 10.3.6 of FSAR Amendment 95 are consistent with the WBN Unit 1 FSAR. Based on its review of FSAR Amendment 95, the NRC staff concludes that the changes to the steam and feedwater system materials controls documented in Section 10.3.6 of the WBN Unit 2 FSAR are acceptable because of the improved corrosion resistance provided by the low-alloy steel piping when used in these systems, and because the NRC staff previously accepted these methodologies for WBN Unit 1.

Based on its review, the NRC staff concludes that the steam and feedwater system materials requirements in WBN Unit 2 FSAR Amendment 95 are consistent with the staff-approved steam and feedwater system materials controls used in WBN Unit 1. Based on its previous evaluation documented in the SER and SSERs, and on its evaluation of FSAR Amendment 95, the NRC staff concludes that the steam and feedwater system materials controls meet the relevant requirements identified in GDC 1 and Section 10.3.6 of NUREG-0800, and are acceptable.

10.3.4 Secondary Water Chemistry

In the SER, the NRC staff concluded that TVA's secondary water chemistry monitoring and control program met (1) the requirements of GDC 14, "Reactor Coolant Pressure Boundary," insofar as control of secondary water chemistry ensures primary boundary material integrity, (2) Acceptance Criterion 3 of SRP Section 5.4.2.1, (3) Positions 2 and 3 of BTP Materials Engineering Branch (MTEB) 5-3, Revision 2, "Monitoring of Secondary Side Water Chemistry in PWR Steam Generators," July 1981, and (4) the program criteria in the staff's position and was, therefore, acceptable. The staff further stated that the operating license will be conditioned to require that the proposed secondary water chemistry monitoring and control program be implemented. In SSER 5, the staff stated that the program would be included in the administrative section of the technical specifications (TS), thus eliminating the need for a license condition. In SSER 21, Section 10.3.4 remained open.

By letter dated February 2, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML100550367), TVA submitted developmental version B of the WBN Unit 2 TS. TVA stated that "this submittal provides the final version of the TS, TS Bases...required in support of licensing WBN Unit 2." WBN Unit 2 administrative TS 5.7.2.13, "Secondary Water Chemistry Program," is the same as the secondary chemistry water program TS requirement for Unit 1.

The NRC staff also reviewed WBN Unit 2 FSAR Amendments 92 through 99. In these amendments, TVA made minor changes to the wording and format of Section 10.3.5, "Water Chemistry." These changes were nonsubstantive.

Based on the NRC staff's review of FSAR Amendments 92 through 99, and because the applicable proposed TS for WBN Unit 2 is the same as that already approved by the staff for Unit 1, the staff concludes that the WBN Unit 2 secondary water chemistry program is acceptable, and that Section 10.3.4 is resolved.

10.4 <u>Other Features</u>

10.4.1 Main Condenser

Section 10.4.1 of the SER states that the main condenser is designed to function as a heat sink for the turbine exhaust steam, turbine bypass steam, and other turbine cycle flows, and to receive and collect condensate flows for return to the steam generators. The main condenser transfers heat to the circulating water systems, which use cooling towers to dissipate the rejected heat. Similar to Unit 1, the Unit 2 main condenser is not required to maintain the reactor in safe-shutdown condition or mitigate the consequences of accidents. FSAR Amendment 95 describes the Unit 2 main condenser as a single-shell, triple-pressure type with a divided water box. It is designed to produce turbine back pressures of 1.92, 2.70, and 3.75 inches of mercury (absolute) for the three pressure zones when operating at rated turbine output.

The turbine back pressures for WBN Unit 1 and Unit 2 are different. In its letter dated July 31, 2010 (ADAMS Accession No. ML102290258), TVA explained that the Unit 1 design pressures of 1.63, 2.38, and 3.40 inches of mercury were based on a cooling water inlet temperature of 70 degrees Fahrenheit, with 95-percent clean tubes. Subsequent design changes to the Unit 1 condenser included retubing of the condenser with SEA-CURE® stainless steel tubes to remove copper from the secondary side for steam generator preservation. TVA also rerouted the condenser zone scheme to mitigate the impact of cooling tower performance. TVA stated that, along with the approved 1.4-percent power uprate inducing increased heat duty onto the condenser zones, the cleanliness of the stainless steel tubing could only be contained between 80- to 85-percent cleanliness, which projected the back pressures of 1.92, 2.70, and 3.75 inches of mercury. With the exception of the rerouting of the condenser zone scheme, similar changes will be applied to the main condenser for Unit 2. In place of rerouting the condenser zone scheme to mitigate the impact of the cooling tower for the Unit 2 main condenser, the licensee will use the supplemental condenser cooling water from Watts Bar Lake to reduce the cooling water inlet temperature, thereby reducing the hotwell temperature. The Unit 2 high- and low-pressure turbines and moisture separator reheaters were replaced and calibrated with the zone pressures from Unit 1.

The NRC staff compared the Unit 2 main condenser specifications with the specifications in TVA's FSAR Amendment 7 for WBN Unit 1. There were differences in the Birmingham Wire Gauge (BWG) balance of tube sizes used for Unit 1 and Unit 2. TVA confirmed in its July 31, 2010, letter that the Unit 2 BWG balance of tube sizes were 22 BWG, which are the same as in Unit 1, and that the FSAR had been corrected in Amendment 98. The licensee also confirmed in its July 31, 2010, letter that the SEA-CURE® stainless steel tubes were used for the Unit 2 main condenser. The staff also inquired about the differences between the design of the Unit 1 main condenser, which allows for water storage capacity of 3 minutes at full-load operation, and the design of the Unit 2 main condenser, which allows for 3½ minutes. Also in its July 31, 2010, letter, TVA clarified that the Unit 1 capacity for the main condenser is also capable of supporting 3½ minutes; TVA revised the information in FSAR Amendment 8 for WBN Unit 1.

In FSAR Amendment 97, TVA provided descriptive details about the design performance of the main condenser, methods to be used to reduce corrosion or erosion of main condenser tubes and components, the inventory of radioactive contaminants in the main condenser, mitigation of hydrogen buildup in the main condenser, and any potential effects of a main condenser failure on safety-related equipment. TVA is applying its previous analysis of the Unit 1 main condenser, which was accepted by the NRC staff, to the Unit 2 main condenser without any changes, other than the exclusion of the condenser zone scheme rerouting described above. TVA will continue to test the Unit 2 main condenser for leakage in the same manner as for Unit 1. TVA also stated that manways in Unit 2 provide access to water boxes, tube sheets, lower steam inlet section, shell, and hotwell for purposes of inspection, repair, or tube plugging.

The NRC staff used the acceptance criteria in SRP Section 10.4.1, "Main Condensers," as guidance to validate that TVA's analysis of the Unit 2 main condenser was consistent with the analysis performed for Unit 1. The staff found no substantive changes or deviations from SRP Section 10.4.1.

Based on its review of the FSAR and the information provided by TVA in its letter dated July 31, 2010, the NRC staff concludes that the Unit 2 main condenser design and performance will meet the acceptance criteria established for the Unit 1 main condenser. Therefore, the conclusions of the SER remain valid, and FSAR Section 10.4.1, "Main Condenser," is acceptable for WBN Unit 2.

10.4.2 Main Condenser Evacuation System

In FSAR Amendment 97, TVA describes the design of the Unit 2 main condenser evacuation system (MCES) to create and maintain condenser back pressure at 1.0 inch of mercury (absolute) by removing noncondensable gases from the condenser. Section 10.4.2 of the SER states that the MCES, one for each unit, included three mechanical vacuum pumps, an electrical heating coil, a high-efficiency particulate air (HEPA) filter, and a carbon adsorber. The NRC staff requested additional information about the components that constitute the Unit 1 and Unit 2 MCESs. In its letter dated July 31, 2010, TVA responded that both the Unit 1 and 2 MCESs comprise three condenser vacuum pumps, radiation monitors for the condenser vacuum pumps, and vents to the roof. TVA also indicated in this letter that the internals for the HEPA filter and carbon absorber train were removed and the duct heater was unplugged.

In the WBN Unit 2 FSAR, TVA proposed that Unit 2 uses three types of radiation monitors to detect radiation at the vacuum pump exhaust; however, WBN Unit 1 FSAR Amendment 7 states that Unit 1 uses two types of radiation monitors. In its July 31, 2010, letter, TVA explained the differences between the Unit 1 and Unit 2 MCES radiation monitors. TVA stated that the Unit 1 MCES originally had three physical monitors with four detectors that monitored the vacuum pump exhaust. TVA removed the channel for the radiation monitor that provided the midrange capability at the Unit 1 vacuum pump exhaust dual detection ranges, because another radiation monitor is capable of providing dual detection ranges, including the midrange. The three older dual detection, dual radiation monitors found in Unit 1 are obsolete. TVA will install two newer separate monitors in the Unit 2 MCES that are capable of detecting the necessary ranges. Because of the difference in equipment identifiers between the two units, it gives the appearance that Unit 2 has three detectors and Unit 1 has only two. Both units will have radiation monitors available to detect the necessary ranges at the vacuum pump exhaust of the MCES.

In reviewing the Unit 2 MCES, the NRC staff compared TVA's Unit 1 analysis to its Unit 2 analysis and reviewed the system using the acceptance criteria in SRP Section 10.4.2. Based on its review of the information provided by TVA, the staff concluded that the MCES analysis for Unit 2 is consistent with the MCES analysis for Unit 1, which was previously approved by the staff. Therefore, the conclusions of the SER remain valid, and FSAR Section 10.4.2 is acceptable for WBN Unit 2.

10.4.3 Turbine Gland Sealing System

In Section 10.4.3 of the SER, the NRC staff reviewed the turbine gland sealing system (TGSS). The system components are designed to Quality Group D, nonsafety-related and nonseismic. The staff reviewed the system description and design criteria for the components of the TGSS and found them to be consistent with the guidance in RG 1.26.

In SSER 21, the NRC staff reviewed existing license review topics to determine whether the topics remained open or were resolved for each section of the FSAR. No open topics were identified for FSAR Section 10.4.3, "Turbine Gland Sealing System." The staff also reviewed TVA's proposed changes to FSAR Section 10.4.3 in FSAR Amendments 95 through 100 and found no proposed changes to the system design or to the system description that would change the staff's conclusion in the SER.

Based on its review, the NRC staff concludes that the description of the TGSS, design criteria, and design bases provided in FSAR Section 10.4.3 remains consistent with the criteria given in RG 1.26. Therefore, the conclusions of the SER remain valid, and FSAR Section 10.4.3 is acceptable for WBN Unit 2.

10.4.4 Turbine Bypass System

In Section 10.4.4 of the SER, the NRC staff reviewed the turbine bypass system. The staff concluded that the turbine bypass system was not required to perform during accident conditions. Therefore, the turbine bypass system components are nonsafety-related. The staff evaluated the turbine bypass system using the guidance provided in BTP ASB 3-1 and BTP MEB 3-1 for breaks and cracks in high-energy and moderate-energy system piping. Evaluation of protection against the dynamic effects associated with the postulated pipe system failures is covered in Section 3.6 of this SSER. In the SER, the staff stated that TVA will include preoperational and startup tests of the turbine bypass system in accordance with the recommendations of RG 1.68. Therefore, the staff concluded that the description of the design criteria and bases of the turbine bypass system in FSAR Section 10.4.4, "Turbine Bypass System," conformed to the acceptance criteria in Section II of SRP Section 10.4.4, and to industry standards, and was acceptable.

In SSER 21, the staff reviewed existing license review topics to determine whether the topics remained open or were resolved for each section of the FSAR. No open topics were identified for FSAR Section 10.4.4. The staff reviewed TVA's proposed changes to FSAR Section 10.4.4 in recent Amendments 95 through 100 and found no changes to the design or description of the system that would change the staff's conclusion in the SER. Therefore, the conclusions of the SER remain valid, and FSAR Section 10.4.4 is acceptable for WBN Unit 2.

10.4.5 Condenser Circulating Water System

Section 10.4.5 of the SER describes the condenser circulating water (CCW) system as being designed to remove the heat rejected from the main condenser to the environment via two natural draft cooling towers. The design and operation of the CCW system for Unit 2 is similar to the design and operation for Unit 1, which the NRC staff previously accepted. The CCW system is nonsafety-related, Quality Group D, and nonseismic. At WBN, the CCW pumping station is an independent structure located in the yard between the cooling towers and the turbine building. Based on its review, the staff found that the CCW system meets the guidelines in RG 1.26.

The CCW system is not a shared system. All components of the system are unitized and are therefore independent of the other unit. Each unit's natural draft cooling tower is designed to reject the full heat load waste of a single unit's main condenser and the associated heat load from the essential raw cooling water (ERCW) and raw cooling water (RCW) systems. Makeup for the CCW system is provided by the ERCW and RCW system discharges. The cooling towers, condenser cooling water pumping station, pumps, motors, and conduits are not required to be designed to seismic Category I, tornado wind, or maximum flood requirements, because they are not required for the safe shutdown of the plant. The cooling towers are located so that a tower's structural failure as a result of a seismic event, tornado, or any other natural phenomenon could not damage any safety-related structure, system, or component. Therefore, the system design is consistent with the requirements of GDC 2 and the guidelines of RG 1.28, "Quality Assurance Program Criteria (Design and Construction)." The SER states that TVA verified that, in the event of a CCW pipe rupture in the turbine building and assuming continuous makeup to the CCW system, flooding of the turbine or service buildings will not endanger any safety-related equipment, including essential electrical systems. All penetrations (doorways, mechanical piping, and electrical penetrations) from the turbine or service buildings to the auxiliary or control buildings are sealed up to elevation 729 feet main sea level (1 foot above grade level). Piping of any system that conveys flow (makeup) to the heat rejection system has provisions to prevent backflow of the CCW into any area where flooding of safety-related components would result from a failure of the system providing the flow.

The NRC staff reviewed the CCW system for compliance with the applicable GDC, RGs, and BTPs and concluded that the CCW system conforms to the requirements of GDC 2 and 4 for protection against natural phenomena and environmental effects due to pipe breaks, and to the guidelines of RG 1.26 and Regulatory Position C.2 of RG 1.29 for the quality group classification and the protection of safety-related systems from failures in nonsafety-related systems. The staff also reviewed TVA's proposed changes to the system in FSAR Amendments 92 through 99 and found no changes that affect the conclusions made by the staff in the SER. Therefore, the conclusions of the original SER remain valid, and FSAR Section 10.4.5, "Condenser Circulating Cooling Water System," is acceptable for WBN Unit 2.

10.4.6 Condensate Cleanup System

Section 10.4.6 of the SER states the following staff findings:

(1) The CCS is capable of removing contaminants from the full condensate flow during normal operation and anticipated operational occurrences to produce feedwater purity in accordance with BTP MTEB 5-3. Therefore, the CCS meets the requirements of GDC 14 as it relates to controlling secondary water chemistry to reduce corrosion so that the integrity of the primary coolant boundary (steam generator tubes) will be maintained.

(2) The instrumentation and sampling equipment provided is adequate to monitor and control process parameters in accordance with BTP MTEB 5-3.

Based on its evaluation performed in accordance with SRP Section 10.4.6 (NUREG-0800), the staff concludes that the proposed condensate cleanup system is acceptable.

In WBN Unit 2 FSAR Amendments 92 through 99, TVA made changes to the wording and format of Section 10.4.6, which is now titled "Condensate Polishing Demineralizer System." The NRC staff found that changes to the condensate cleanup system (CCS) instrumentation do not affect the staff's conclusion in the SER that the instrumentation and sampling equipment provided is adequate to monitor and control process parameters in accordance with BTP MTEB 5-3.

However, the staff notes that the reference to Table 10.3.2, "Feedwater Chemistry Specification," and the table itself have been removed. As a result, the staff can no longer conclude that the CCS is capable of producing feedwater purity in accordance with BTP MTEB 5-3.

TVA should provide information to the NRC staff that the CCS will produce feedwater purity in accordance with BTP MTEB 5-3 or, alternatively, provide justification for producing feedwater purity to another acceptable standard. This is Open Item 35 (Appendix HH).

10.4.7 Condensate and Feedwater Systems

In Section 10.4.7 of the SER, the NRC staff reviewed the condensate and feedwater systems. The SER describes the condensate and feedwater systems to be designed to supply a sufficient quantity of feedwater to the steam generator secondary side inlet during all normal operating conditions and to guarantee that feedwater will not be delivered to the steam generators when feedwater isolation is required. Complete isolation of the main feedwater system is provided when it is required to mitigate the consequences of a steam or feedwater line break. To ensure feedwater system isolation in accident situations, the portion of the condensate feedwater system between the check valves located outside the containment, including the containment isolation valves and up to the steam generators, is classified as safety-related and is designed to seismic Category I, Quality Group B requirements. This portion of the system is located in seismic Category I, flood- and tornado-protected structures. These structures provide protection against tornado missiles. The essential equipment is separated from the effects of internally generated missiles and is not affected by failures in high-energy piping, thus satisfying the requirements of GDC 2 and GDC 4 and the guidelines of RG 1.117. The main feedwater regulating valves and the main feedwater isolation valves are credited to close on receipt of the feedwater isolation signal within 5.0 seconds and 6.5 seconds, respectively. The remainder of the system piping serves no safety function and is classified as nonsafety-related Quality Group D and nonseismic Category I.

In the SER, the NRC staff reviewed the condensate and feedwater systems for compliance with the applicable GDC, RGs, and BTPs and concluded that the condensate and feedwater systems conform to the requirements of GDC 2 and GDC 4 for protection against natural phenomena and missiles and meet the guidelines of RGs 1.26, 1.29, and 1.117 for quality group and seismic classification and tornado missile protection. Therefore, the staff found FSAR Section 10.4.7, "Condensate and Feedwater Systems," to be acceptable.

In SSER 21, the NRC staff reviewed existing license review topics to determine whether the topics remained open or were resolved for each section of the FSAR. No open topics were identified for FSAR Section 10.4.7.

In SSER 14, issued December 1994, the NRC staff noted that in the SER a feedwater isolation signal was initiated by (1) a high-high steam generator level, (2) an engineered safety feature safety injection actuation signal, or (3) a reactor trip. In FSAR Amendment 82, TVA revised FSAR Section 10.4.7 to identify a new feedwater isolation signal and to clarify the isolation signal generated by a reactor trip. TVA added that a signal from the high-flood-level detection in either the south or north main steam valve vault rooms will generate a main feedwater isolation signal. Also, a main feedwater isolation signal would be generated by a reactor trip only if the reactor trip were coincident with a low reactor-coolant-average-temperature signal. Based on its review, the staff concluded that the changes were acceptable and that the conclusions of the SER remained valid.

Additionally, the staff noted in SSER 14 that it had found an unrelated error during its review of FSAR Amendment 82, dated January 31, 1994. In the SER, the staff stated that, upon receipt of a feedwater isolation signal, the main feedwater regulating valves and the main feedwater isolation valves will close within 5.0 seconds and 6.5 seconds, respectively. However, according to the SER, both the feedwater regulating valves and feedwater isolation valves will close within 6.5 seconds of initiation of the feedwater isolation signal. Because the accident and containment analyses are based on a closure time 6.5 seconds from the initiation of the feedwater isolation signal, the staff concluded that 6.5 seconds was acceptable for both valves and considers this a matter of clarification of the SER.

The NRC staff reviewed TVA's changes to Section 10.4.7 in FSAR Amendments 95 through 100 and found no changes that would affect the system design, system description, or the conclusions reached in the SER.

Based on its review, the NRC staff concludes that the description of the condensate and feedwater systems, design criteria, and design bases in FSAR Section 10.4.7 is consistent with the criteria given in RG 1.26 and complies with the regulatory requirements noted above. Therefore, the conclusions of the SER remain valid, and FSAR Section 10.4.7 is acceptable for WBN Unit 2.

10.4.8 Steam Generator Blowdown System

Section 10.4.8, "Steam Generator Blowdown System," (SGBS) of the SER, states the following (in part):

> The components of the SGBS downstream from the outer isolation valves are considered to be nonsafety-related and are constructed according to the quality standards of Quality Group D, as augmented by Regulatory Position C.1.1 of Regulatory Guide 1.143 ("Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants"), thereby meeting the quality standards requirements of GDC 1.
>
> Instrumentation and automatic controls are provided to monitor and control the operation of the blowdown system, with provision for sampling of the blowdown in conformance with the guidelines of BTP MTEB 5-3.

Based on the above, the NRC staff concluded in the SER that the SGBS was acceptable.

In WBN Unit 2 FSAR Amendments 92 through 99, TVA made changes to the wording and content of Section 10.4.8. TVA removed the list of design code requirements and added a reference to Section 3.2.2. A review of Section 3.2.2, including Table 3.2.2, shows that the components downstream from the outer isolation valves are not constructed to the quality standards of Quality Group D, as augmented by Regulatory Position C.1.1 of RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants." Therefore, the NRC staff considers that the conclusion of the SER is no longer valid.

TVA did not provide any justification for meeting another standard or criterion for this portion of the system. The NRC staff therefore has no basis for determining the acceptability of the SGBS or whether it meets the requirements of GDC 1.

TVA also removed from the FSAR the requirement for daily steam generator blowdown samples and added a reference to sampling requirements in accordance with the offsite dose collection manual. Therefore, the previous staff conclusion that the provision for sampling of the blowdown is in conformance with the guidelines of BTP MTEB 5-3 (specifically, Section II.3.d) is no longer valid.

TVA did not provide any justification for meeting another standard or criterion for sampling. The staff therefore has no basis for determining the acceptability of the SGBS or whether it meets the requirements of GDC 14 for maintaining acceptable secondary water chemistry control by reducing the corrosion rate.

TVA should provide information to the NRC staff to enable verification that the SGBS meets the requirements and guidance specified in the SER or provide justification that the SGBS meets other standards that demonstrate conformance to GDC 1 and GDC 14. This is Open Item 36 (Appendix HH).

10.4.10 Heater Drains and Vents

In its letter dated July 31, 2010 (ADAMS Accession No. ML102290258), TVA stated the following:

> Heater Drains and Vents (HDV) are designed to remove all condensate from the feedwater heaters, moisture separators, 1st stage (low pressure) reheaters, 2nd stage (high pressure) reheaters, main feed pump turbine condensers and gland steam condensers. These drains and vents are not required for the safe shutdown of the plant or to mitigate the consequences of an accident.
>
> During the re-constitution of the Unit 2 FSAR, one of the underlying tenets was to have the Unit 2 FSAR correlate as closely as possible to the Unit 1 UFSAR. Section 10.4.10 was removed from the Unit 1 UFSAR by Amendment 1; documented in Change Package 1569 and the Safety Evaluation for EDC E-50038-A.

RG 1.70, Rev. 3, "Standard Format and Content of Safety Analysis for Nuclear Power Plants (LWR Edition)" and NUREG-0847 (series) "Safety Evaluation Reports Related to Watts Bar Nuclear Plant" do not require a discussion about HDVs. Amendment 94 to the Unit 2 FSAR removed 10.4.10 from the Table of Contents.

There are no regulatory requirements or guidance in RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," or in the SER for the licensee to provide a description of the heater drain and vent system in the FSAR; therefore, the NRC staff finds the omission of this section from the FSAR to be acceptable.

10.4.11 Steam Generator Wet Layup System

In its letter dated July 31, 2010 (ADAMS Accession No. ML102290258), TVA stated that the steam generator wet layup system is no longer used in WBN Unit 1 and will not be used in Unit 2. The licensee has adopted an alternate method of corrosion control for the steam generators while in a wet layup condition, as described in FSAR Section 10.3.5.

Because the steam generator wet layup system is not used at WBN, the NRC staff did not review FSAR Section 10.4.11.

13 CONDUCT OF OPERATIONS

13.1 Organizational Structure of Applicant

In the safety evaluation report (SER), the U.S. Nuclear Regulatory Commission (NRC) staff found the organizational structure of the Tennessee Valley Authority (TVA) acceptable. Since then, TVA has revised Section 13.1.1 of the final safety analysis report (FSAR) to state that organizational information is as presented in TVA Topical Report TVA-NPOD89-A, "TVA Nuclear Power Group Organization Description." In Section 13.1 of Supplemental Safety Evaluation Report (SSER) 16, the staff found TVA's organizational structure acceptable based on the staff's approval of TVA Topical Report TVA-NPOD89 and annual updates to the topical report through Revision 6. The staff's approval of the topical report and its updates supersedes the approval given by the staff in the SER. The revision reviewed by the staff in this SSER of TVA-NPOD89-A is Revision 18, issued August 31, 2009.

13.1.3 Plant Staff Organization

In Section 13.1.2 of the SER, the NRC staff stated in its review of "plant staff and qualifications" that a license condition (Proposed License Condition 26) would be imposed in the operating license (OL) to ensure that TVA will augment the Watts Bar shift staff with individuals who have experience with large pressurized-water reactor (PWR) operations. By FSAR Amendment 63 and TVA's Nuclear Quality Assurance (NQA) Plan, TVA committed to complying with the guidelines of Regulatory Guide (RG) 1.8, "Personnel Selection and Training," Revision 2, issued April 1987. In SSER 8, the staff determined that this commitment provided adequate assurance that the intended requirements of Proposed License Condition 26 would be met and so eliminated the need for the proposed license condition in the WBN Unit 1 operating license.

In Section 13.1.3 of FSAR Amendment 101 for WBN Unit 2, TVA states that "Nuclear Power (NP) personnel at the Watts Bar plant will meet the qualification and training requirements of NRC RG 1.8 with the alternatives as outlined in the Nuclear Quality Assurance Program, TVA-NQA-PLN89-A." TVA's commitment to RG 1.8 for WBN Unit 2 provides adequate assurance that TVA will augment the shift staff with individuals who have experience with large PWR operations and is, therefore, acceptable to the staff.

In order to complete its evaluation of TVA's plant staff organization, TVA should provide information to the NRC staff to allow the staff to confirm that:

1) The education and experience of management and principal supervisory positions down through the shift supervisory level conform to RG 1.8. The staff will review the resumes to confirm this.

2) TVA has an adequate number of licensed and non-licensed operators in the training pipeline to support the preoperational test program, fuel loading, and dual unit operation.

3) The plant administrative procedures clearly state that when the Assistant Shift Engineer assumes his duties as Fire Brigade Leader, his control room duties are temporarily assumed by the Shift Supervisor (Shift Engineer), or by another SRO, if one is available. The staff will confirm that the plant administrative procedures clearly describe this transfer of control room duties.

These are Open Items 9, 10, and 11 (Appendix HH).

13.2 Training

13.2.1 Licensed Operator Training Program

In the SER, the NRC staff found TVA's plant staff training program acceptable based on a review of the licensee's conformance to positions in Sections 13.2.1 and 13.2.2 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (also referred to as the SRP); RG 1.8; NUREG-0737, "Clarification of TMI Action Plan Requirements," issued November 1980; and the 1981 edition of Title 10 of the *Code of Federal Regulations* (10 CFR), Sections 55.20, "Scope of Examinations," 55.21, "Content of Operator Written Examination," 55.22, "Content of Senior Operator Written Examination," 55.23, "Scope of Operator and Senior Operator Operating Tests," and 55.25, "Administration of Operating Test Prior to Initial Criticality."

In SSER 9, the NRC staff accepted TVA's choice to substitute an accredited training program, based on a system approach to training (SAT) for initial and requalification training programs previously approved by the NRC. The staff's acceptance was based on TVA's written certification that the substitute licensed operator training programs have been developed using an SAT-based training program, are accredited by the National Nuclear Accrediting Board, and have been implemented using a certified simulation facility. In FSAR Amendment 70, the licensee updated Chapter 13 to reflect the certified licensed operator training programs. The licensee's commitment to substitute an accredited, SAT-based training program represents an increase in the licensed operator training program requirements and is acceptable.

In SSER 10, the NRC staff concluded that TVA had satisfied the requirement for licensed operators to train on natural circulation mode (NUREG-0737, Item I.G.1) by use of an SAT-based training program. The SAT-based training program should ensure that the licensed operator training program includes the activities listed in 10 CFR 55.59(c)(3)(i), as appropriate to the facility. Specifically, this includes "loss of core coolant flow/natural circulation," as listed in 10 CFR 55.59(c)(3)(i)(J).

In Section 13.2.1 of FSAR Amendment 97, dated January 11, 2010, TVA states the following:

> The Watts Bar Nuclear Plant (WBN) training programs have been developed in accordance with the Systems Approach to Training as prescribed by the Institute of Nuclear Power Operations (INPO). The National Academy for Nuclear Training, through a formal accreditation process, verifies that WBN training programs meet the established criteria. WBN is a branch of the National Academy and has achieved accreditation of the following programs:
>
> Non-licensed operator
> Reactor operator
> Senior reactor operator
> Continuing training for licensed personnel
> Shift manager
> Shift technical advisor
> ...

Based on (1) its review of the information provided by TVA in WBN Unit 2 FSAR Amendment 97 and the staff's previous review as documented in the SER and supplements, (2) the industry accreditation, as described in RG 1.8, of the TVA training programs, and (3) the results of the NRC's periodic examinations of TVA licensed operators and inspections of the training program at WBN Unit 1, the NRC staff finds that TVA's plant staff training program continues to be acceptable.

13.3 Emergency Preparedness

13.3.1 Introduction

As described in Section 13.3, "Emergency Preparedness," of FSAR Amendment 101, the TVA Radiological Emergency Plan (REP) has been developed to provide protective measures for TVA personnel, and to protect the health and safety of the public in the event of a radiological emergency resulting from an accident at Watts Bar Nuclear Plant (WBN). The REP contains site-specific appendices for each unit. The TVA REP consists of two portions: (1) a site-independent Nuclear Power Radiological Emergency Plan (NP-REP) that applies to all of TVA's licensed nuclear power reactors and (2) a series of site-specific appendices for each of the TVA nuclear sites—Browns Ferry Nuclear Plant (Appendix A), Sequoyah Nuclear Plant (Appendix B), and the WBN Plant (Appendix C). The TVA REP and the WBN-specific Appendix C, but omitting Appendices A, B, D, and E, form the WBN Unit 2 "emergency plans," as referred to in 10 CFR 50.34(b)(6)(v), 10 CFR 50.47(b), and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities." The generic NP-REP and the WBN-specific Appendix C are collectively referred to in this SER supplement as the "WBN REP."

The WBN REP expresses the overall concept of operation and describes the essential elements of advance planning that have been considered and the provisions that have been made to cope with radiological emergencies. The WBN REP describes the site's emergency response organization (ERO), the emergency response facilities, and other capabilities, methods, and resources by which TVA complies with the planning standards of 10 CFR 50.47(b) and the requirements of 10 CFR Part 50, Appendix E. The emergency plan implementing procedures (EPIPs), which provide detailed instructions for implementing the provisions of the WBN REP, support these plans.

The objective of the NRC staff review documented here is to determine whether the proposed extension of the existing WBN REP to incorporate Unit 2 has adequately addressed the differences between the two units and any dual-unit issues that arise from the licensing and operation of Unit 2. The NRC will use the results from this review to make its finding, under 10 CFR 50.47(a)(1)(i), that adequate protective measures can and will be taken in a radiological emergency at Unit 2. TVA should evaluate the impact of Unit 2 related changes on the effectiveness of the WBN REP, as it applies to Unit 1, under 10 CFR 50.54(q).

13.3.1.1 History of the Licensing Proceeding

TVA originally submitted the WBN REP on January 28, 1982, for Unit 1 and Unit 2. The staff's review of that plan is discussed in NUREG-0847, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2," issued June 1982 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML072060490). TVA halted construction and licensing activities at WBN in 1985. Construction and licensing activities resumed later in 1985 for Unit 1; construction and licensing activities for Unit 2 remained suspended. TVA withdrew the previously approved WBN REP and on February 12, 1993, submitted Revision 15 of the WBN REP on both WBN dockets (ADAMS Accession No. ML073230252, non-publicly available). The staff performed a complete review of the WBN REP on the Unit 1 docket and documented the results of that review, with open items, in SSER 13 issued April 1994 (ADAMS Accession No. ML072060484). The staff determined that the WBN REP provided an adequate planning basis for an acceptable state of onsite emergency preparedness that met the planning standards of 10 CFR 50.47(b) and the requirements of Appendix E to 10 CFR Part 50.

In SSER 20, issued February 1996 (ADAMS Accession No. ML072060498), the NRC staff documented the completion of its review of WBN REP. The staff determined, pursuant to 10 CFR 50.47(a), that the overall state of onsite and offsite emergency preparedness provided reasonable assurance that adequate protective measures could and would be taken in the event of a radiological emergency at WBN. The staff based its determination on its assessment of the adequacy of the onsite plans, TVA's ability to implement those plans, and the staff's review of the Federal Emergency Management Agency (FEMA) assessment of the adequacy of the offsite plans and the State and local governments' ability to implement those plans. The staff concluded that emergency preparedness at WBN was adequate to support full-power operations.

Although the WBN REP and SSERs 13 and 20 do not explicitly exclude Unit 2, the staff at that time performed its review only in the context of Unit 1, since licensing activities for Unit 2 had been suspended at TVA's request.

On August 3, 2007, TVA notified the NRC of its intention to reactivate construction activities at Unit 2 (ADAMS Accession No. ML072190047). On January 29, 2008, TVA submitted a proposed regulatory framework for the completion of construction and licensing activities (ADAMS Accession No. ML080320443). Table 1 in TVA's letter stated its position that emergency preparedness was a closed issue for Unit 2 based on the NRC conclusion in Supplement No. 20 to the SER. By letter dated May 8, 2008, the NRC staff provided its initial assessment of the remaining OL review scope (ADAMS Accession Nos. ML080860026 and ML080860040). In Table 2 in the staff's letter, the staff identified FSAR Chapter 13.3, "Emergency Plan," as being an open item. The staff took this position because it had reviewed the emergency plan evaluated in Supplements No. 13 and 20 of the SER only in the context of Unit 1, and the staff would need to make an explicit finding for Unit 2 in this regard.

The NRC staff discussed this licensing proceeding with FEMA staff as proscribed in the memorandum of understanding (MOU) between the NRC and FEMA (44 CFR Part 353, "Fee for Services in Support, Review and Approval of State and Local Government or Licensee Radiological Emergency Plans and Preparedness," Appendix A, "Memorandum of Understanding Between Federal Emergency Management Agency and Nuclear Regulatory Commission Relating to Radiological Emergency Planning and Preparedness," dated September 14, 1993). In a meeting in October 2009, FEMA advised the staff that it would review the offsite plans. As stated in the MOU, the NRC's responsibilities are as follows:

1. To assess licensee emergency plans for adequacy...

2. To verify that licensee emergency plans are adequately implemented...

3. To review the FEMA findings and determinations...

4. To make radiological health and safety decisions with regard to the overall state of emergency preparedness...

13.3.1.2 Content of the Application

By letter dated March 4, 2009 (ADAMS Accession No. ML090700378), TVA submitted a marked-up (redline) version of Appendix C to the TVA REP in support of the Unit 2 OL application pursuant to 10 CFR 50.34(b)(6)(v). By letter dated December 3, 2009 (ADAMS Accession No. ML100120493), TVA supplemented its application with the following:

- a copy of the WBN REP (i.e., Revision 89 of the NP-REP and Revision 89 of Appendix C) and the other site-specific appendices and supporting attachments
- a copy of the March 15, 2009, revision of the Tennessee Multi-Jurisdictional Radiological Emergency Response Plan (MJRERP) (pursuant to 10 CFR 50.33(g))
- the Watts Bar 2009 public information calendar

The MJRERP (Appendix E to the NP-REP) comprises the radiological emergency plans of the State of Tennessee and local governmental entities that are within the WBN site plume exposure pathway emergency planning zone (EPZ), as referred to in 10 CFR 50.33(g). The NRC staff provided these plans to FEMA for review.

Annex H of the MJRERP contains the analysis of the time required to evacuate and to take other protective actions for transient and permanent populations within the WBN site plume exposure pathway EPZ as required by 10 CFR Part 50, Appendix E, Section IV.

By letters dated December 14, 2009 (ADAMS Accession No. ML093500617), April 27, 2010 (ADAMS Accession No. ML101230264), July 15, 2010 (ADAMS Accession No. ML102070552), August 13, 2010 (ADAMS Accession No. ML02250233), and August 26, 2010 (ADAMS Accession No. ML102380335), TVA supplemented its application by responding to staff requests for additional information (RAIs). On October 11, 2010, TVA supplemented its application by responding (ADAMS Accession No. ML102930286, non-publicly available) to staff requests for additional information and by submitting the following:

- a copy of Revision 92 of the NP-REP
- a copy of Revision 92 of the WBN REP, Appendix C, specific to Unit 1
- a copy of Revision 92xx of the draft WBN Unit 1/2 REP, Appendix C

The latter document was produced from Revision 92 of the WBN REP for Unit 1, Appendix C, and contains information specific to Unit 2 to identify the changes that will become effective upon the licensing of Unit 2, at which time there will be just one Appendix C for both units. On November 17, 2010 (ADAMS Accession No. ML103210645, non-publicly available), TVA further supplemented its application by correcting an error in the October 11, 2010, submittal.

13.3.1.3 Regulatory Basis

Based on the regulatory requirements of 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50, the NRC staff reviewed the application. In accordance with 10 CFR 50.47(a)(1)(i), no initial operating licensee for a nuclear power reactor will be issued under 10 CFR Part 50 unless the NRC finds that there is a reasonable assurance that adequate protective measures can and will be taken in a radiological emergency. As provided by 10 CFR 50.47(a)(2), the NRC will base its finding on a review of the FEMA findings and determinations as to whether State and local offsite emergency plans are adequate, and whether there is reasonable assurance that they can be implemented, and on the NRC assessment, documented herein, as to whether the applicant's onsite emergency plans are adequate and whether there is reasonable assurance that they can be implemented.

Section 13.3, "Emergency Planning," of the SRP provides guidance to the NRC staff for the conduct of the review and evaluation of emergency planning information submitted in an OL application and the determination of compliance with the applicable regulations. The SRP identifies NUREG-0654/FEMA-REP-1, Revision 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," issued November 1980; NUREG-0696, "Functional Criteria for Emergency Response Facilities," issued February 1981; and other related guidance that staff should consider during the review. SRP Section 13.3.II, "Acceptance Criteria," identifies the applicable acceptance criteria.

In accordance with 44 CFR Part 353, Appendix A, FEMA is responsible for the findings and determinations as to whether offsite emergency plans are adequate and can be implemented. FEMA regulations for this review appear in 44 CFR Part 350, "Review and Approval of State and Local Radiological Emergency Plans and Preparedness." These regulations require that 10 CFR 50.47, "Emergency Plans," Appendix E to 10 CFR Part 50, and NUREG-0654/FEMA REP-1, Revision 1, be used in evaluating State and local emergency plans and preparedness. FEMA radiological offsite emergency preparedness program documents provide guidance on various topics for use by State and local organizations responsible for radiological emergency preparedness and response.

13.3.2 Technical Evaluation

This section provides the results of the NRC staff's review of the WBN REP, which TVA characterizes as a modification of the WBN REP as approved for Unit 1 to reflect Unit 2. In accordance with the guidance in Section 13.3 of the SRP, the staff focused its review on the changes identified in the modified WBN REP and applied the following guidance from the SRP:

> In general, if an application is for an additional reactor at an operating reactor site, and the application proposes to incorporate and extend elements of the existing emergency planning program to the new reactor (included by reference), those existing elements should be considered acceptable and adequate. The reviewer should generally focus the review on the extension of the existing program to the new reactor, and should determine whether the incorporated emergency planning program information from the existing reactor site (1) is applicable to the proposed reactor, (2) is up-to-date when the application is submitted, and (3) reflects use of the site for the construction of a new reactor (or reactors) and appropriately incorporates the new reactor(s) into the existing plan.

Applicability of the WBN REP to Unit 2

The WBN site was originally permitted for construction of two reactor units, and TVA submitted an OL application including an onsite emergency plan for both. In its April 27, 2010, response to RAI 4, TVA stated that the WBN site was originally designed for two-unit operation and the existing WBN REP was written to provide adequate margin for the addition of a second unit. TVA also identified that differences in design have been minimized to ensure that the existing staff's knowledge and skills are adequate to ensure a strong ERO. The staff acknowledges its earlier approval of the WBN REP for both units.

Construction continued on both units until work on Unit 2 was suspended in 1985, at which time approximately 90 percent of the construction on Unit 2 was complete. When construction was suspended on Unit 2, the licensing and design bases of Unit 1 and Unit 2 were essentially identical. In its letter of August 3, 2007, TVA stated its intent to align the Unit 1 and Unit 2 licensing and design bases to ensure that there is design and operational fidelity between the units while demonstrating compliance with applicable NRC regulatory requirements. Because the WBN EPZ was established in support of both Unit 1 and Unit 2, and given the design and operational fidelity between Unit 1 and Unit 2, no change in the shape and size of the EPZ is necessary because of the licensing of Unit 2.

The onsite emergency preparedness program at Unit 1 has been subject, since the issuance of the Unit 1 OL, to routine NRC inspection oversight, and TVA has conducted the biennial exercises required by 10 CFR 50.47(b)(14) and Section IV.F.2.b of Appendix E to 10 CFR Part 50. The NRC routinely evaluates TVA's performance in these exercises and finds TVA's ability to critique and correct identified weaknesses acceptable, as documented in inspection reports.[1] In the most recent Unit 1 emergency exercise, conducted June 10, 2009, the NRC identified no findings of significance (ADAMS Accession No. ML091980360). TVA's emergency preparedness cornerstone performance indicators for Unit 1 are in the licensee response band of the Reactor Oversight Process action matrix. (See http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/.) This demonstrated performance confirms the staff findings made at the time of Unit 1 licensing that the WBN REP was adequate and that there is reasonable assurance that TVA could and would implement it.

Currentness of the WBN REP at Time of Submittal

At the start of its review, the NRC staff was aware that TVA could modify the WBN REP during the period of this review in the interest of Unit 1 and/or the other two nuclear sites that TVA operates. As authorized by 10 CFR 50.54(q), TVA could make these changes without prior NRC approval if it met certain conditions. Thus, the staff could not know whether the version it was reviewing was current. The staff issued RAI 1 to raise this concern with TVA. TVA's April 27, 2010, response to this RAI did not fully address the staff's concern. The staff subsequently became aware that the staffing commitments made in the redline version of Appendix C to the WBN REP, submitted on March 4, 2009, and marked as Revision 88, were not shown in Revision 89, which was submitted in December 2009. The staff subsequently issued RAI 23 specifically to question this discrepancy. In a teleconference, TVA advised the staff that the March 4, 2009 submittal should not have been identified as Revision 88, as it was TVA's intent to maintain separate versions - one for WBN Unit 2 and one for the operating

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[1] The NRC oversight of the emergency plans and exercises at the Sequoyah and Browns Ferry sites provides additional insight into TVA's potential emergency preparedness performance at Unit 2, as the site-independent aspects of the NP-REP apply to all TVA sites.

plants. In its July 15, 2010, response to RAI 23, TVA committed to providing the NRC with updates of all changes made to the WBN REP in the future to ensure that the NRC staff has accurate and complete information. In the October 11, 2010, submittal, TVA provided copies of Revision 92 to the NP-REP and Appendix C, and a new WBN Unit 1/2, Appendix C, designated as Revision 92xx.

The NRC staff review documented in this SSER used Revision 92 of the NP-REP and the WBN Unit 1/2 version of Appendix C, designated as Revision 92xx. Unless otherwise stated, "WBN REP," as used in this supplement, refers to Revision 92 of the NP-REP and the WBN Unit 1/2 version of Appendix C, Revision 92xx. The staff has identified an open item to review the combined WBN Appendix C before issuance of the Unit 2 OL to confirm (1) that the proposed Unit 2 changes were incorporated into Appendix C and (2) that changes made to Appendix C for Unit 1 since Revision 92 and the changes made to the NP-REP since Revision 92 do not affect the bases of the staff's findings in this SER supplement. This is Open Item 37 (Appendix HH).

Construction of a New Reactor on the Site Reflected in the WBN REP

In reviewing the WBN REP, the NRC staff did not find explicit information to support a finding on whether the WBN REP reflects provisions for the Unit 2 construction activity. In its April 27, 2010, response to RAI 4.c, TVA stated that both units are (1) within the same protected area (PA) boundary, and (2) all personnel working on Unit 2 are subject to the same training (including general employee actions required during emergency plan implementation) and access controls as the personnel assigned to Unit 1. The staff had previously found these aspects of the WBN REP, as approved for Unit 1, acceptable. The staff notes that these provisions have been successfully used to address the influx of construction personnel associated with the regularly scheduled Unit 1 outages. TVA stated that it revised its procedures for personnel accountability and evacuation to include a provision for early dismissal, which was successfully demonstrated in a drill conducted in June 2008. In addition, TVA has assigned trailer wardens for construction trailers located inside and outside of the PA to assist in evacuation of construction personnel. These trailer wardens receive the same pager signal that the site ERO teams receive during activation of the WBN REP. Given that the construction personnel are enveloped within the employee protective measures arrangements in the existing WBN REP, as approved for Unit 1, the staff finds these provisions acceptable.

In its April 27, 2010, response to RAI 2 regarding changes that TVA has made under 10 CFR 50.54(q) since the WBN REP was approved by the NRC, TVA provided a listing of all such changes with a brief summary of each. The staff screened these changes to identify any changes that could potentially affect the bases for the staff's approval of the extension of the WBN REP to Unit 2. The staff performed this screening to ensure the continued validity of the staff's discussions in SER Supplement Nos. 13 and 20. With the exception of an issue regarding ERO augmentation time, which is addressed in Section 13.3.2.2 of this SER supplement, the staff found no issues.

Accordingly, the WBN REP, as approved for Unit 1, is a candidate for treatment under the SRP provision excerpted above, and the remainder of the NRC staff's review is focused on the adequacy of TVA's actions to extend the WBN REP, as approved for Unit 1, to include Unit 2. Sections 13.3.2.1 through 13.3.2.16 below, which correspond directly with the identically numbered sections in SSER 13, discuss the staff's review against the planning standards of 10 CFR 50.47(b)(1) through (b)(16), respectively.

13.3.2.1 Assignment of Responsibility (Organizational Control)

Regulatory Basis: Planning standard 10 CFR 50.47(b)(1) requires that primary responsibilities for emergency response by the nuclear facility licensee and by State and local organizations within the EPZs have been assigned, the emergency responsibilities of the various supporting organizations have been specifically established, and each principal response organization has staff to respond and to augment its initial response on a continuous basis. Section IV.A of Appendix E to 10 CFR Part 50 provides supporting requirements.

Technical Evaluation: In Section 13.3.2.1 of SSER 13, the staff described its evaluation of the assignment of responsibilities for emergency response in the WBN REP and its finding that TVA had met the planning standard 10 CFR 50.47(b)(1) for Unit 1.[2] TVA's subsequent performance in actual emergency events, periodic emergency exercises, and periodic routine and supplemental inspections continue to support the findings made by the staff regarding the WBN REP, as approved for Unit 1. Having identified that no changes are needed to the WBN REP to reflect Unit 2, TVA proposes to use the same assignment of responsibilities for emergency response for Unit 2 as it has used for Unit 1. The staff finds that no dual-unit or unit-specific changes are necessary to reflect Unit 2 because the assignments of responsibilities are not unit specific, given the design and operational fidelity between the units.

Conclusion: The staff concludes that the WBN REP, extended to incorporate Unit 2, meets the requirements of 10 CFR 50.47(b)(1) and Section IV.A of Appendix E to 10 CFR Part 50 with regard to assignment of responsibilities for emergency response. The staff bases its conclusion on (1) the approval of the WBN REP for Unit 1 as described in Supplement No. 13 to the SER, (2) TVA's demonstrated performance in maintaining emergency preparedness at the existing Unit 1, (3) the design and operational fidelity between the two units, and (4) the WBN REP's applicability to, and incorporation of, Unit 2.

13.3.2.2 Onsite Emergency Organization

Regulatory Basis: Planning standard 10 CFR 50.47 (b)(2) requires that on-shift facility licensee responsibilities for emergency response are unambiguously defined, adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, timely augmentation of response capabilities is available, and the interfaces among various onsite response activities and offsite support and response activities are specified. Section IV.A of Appendix E to 10 CFR Part 50 provides supporting requirements.

Technical Evaluation: In Section 13.3.2.2 of SSER 13, the staff described its evaluation of the WBN REP onsite emergency organization and its finding that TVA had met the planning standard 10 CFR 50.47(b)(2) for Unit 1. TVA's subsequent performance in actual emergency events, periodic emergency exercises, and periodic routine and supplemental inspections continue to support the findings made by the staff regarding the WBN REP. By letter dated May 22, 2003 (ADAMS Accession No. ML031480076, non-publicly available), TVA proposed a change in the Unit 1 staffing levels and the augmentation time for the Technical Support Center (TSC) and Operations Support Center (OSC). By letter dated June 24, 2004 (ADAMS Accession No. ML041810056), the staff issued its safety evaluation, which found these changes acceptable. The staff's acceptance of that staffing proposal was based, in part, on the

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[2] In Section 13.3.2 of Supplement No. 13 to the SER, the staff did not provide a conclusion for each discussion of a planning standard, but instead presented an overall conclusion that the plan met the planning standards of 10 CFR 50.47(b) and the requirements of Appendix E to 10 CFR Part 50.

availability of 6–10 advanced radiation workers to perform various radiation protection activities and 5 dedicated shift fire operations personnel to provide firefighting, rescue, and first aid functions. In its April 27, 2010, response to RAI 5.a and 5.b, TVA stated that these personnel would also be available to Unit 2.

In its March 4, 2009, submittal, TVA proposed to add two additional on-shift positions: a second Unit Supervisor and an additional Reactor Operator to reflect dual-unit operation. These additions are consistent with the guidance in Table B-1 of NUREG-0654/FEMA-REP-1, Revision 1. Although footnote "*" of Table B-1 of NUREG-0654/FEMA-REP-1, Revision 1 also requires an additional Auxiliary Operator, the WBN ERO already provides five Assistant Unit Operators, three more than required by Table B-1. Other than these specific changes, TVA proposes to use the same ERO for Unit 2 as it has used for Unit 1.

However, in performing the current review, the staff determined that TVA's implementation of the revised augmentation time was inconsistent with the language of the staff's June 24, 2004, safety evaluation. In its October 11, 2010, response to RAI 5.c, TVA provided a revision to Section C.5 of the WBN REP to address the staff's concern. The staff found the content of the revision to be consistent with the intent of its earlier safety evaluation and, therefore, acceptable.

Conclusion: The NRC staff concludes that the WBN REP, extended to incorporate Unit 2, meets the requirements of 10 CFR 50.47(b)(2) and Section IV.A of Appendix E to 10 CFR Part 50 with regard to the onsite emergency organization. The staff bases its conclusion on (1) the approval of the WBN REP for Unit 1 as described in Supplement No. 13 to the SER, (2) TVA's demonstrated performance in maintaining emergency preparedness at the existing Unit 1, (3) the design and operational fidelity between the two units, and (4) the WBN REP's applicability to, and incorporation of, Unit 2.

13.3.2.3 Emergency Response Support and Resources

Regulatory Basis: Planning standard 10 CFR 50.47(b)(3) requires that arrangements for requesting assistance and effectively using resources have been made, arrangements to accommodate various State and local staff at the licensee's near-site emergency operations facility (EOF) have been made, and other organizations capable of augmenting the planned response have been identified. Section IV.A of Appendix E to 10 CFR Part 50 provides supporting requirements.

Technical Evaluation: In Section 13.3.2.3 of SSER 13, the NRC staff described its evaluation of the arrangements for emergency response support and resources as described in the WBN REP and its finding that TVA had met the planning standard 10 CFR 50.47(b)(3) for Unit 1. TVA's subsequent performance in actual emergency events, periodic emergency exercises, and periodic routine and supplemental inspections continue to support the findings made by the staff regarding the WBN REP. The staff reviewed TVA's letters of agreement for external support and resources and found them to be complete and current. Having identified no changes needed to the WBN REP to reflect Unit 2, TVA proposes to use the same assignment of responsibilities for emergency response for Unit 2 as it has used for Unit 1. The staff finds that no dual-unit or unit-specific changes are necessary to reflect Unit 2 because arrangements for emergency response support and resources are not unit specific, given the design and operational fidelity between both units.

Conclusion: The NRC staff concludes that the WBN REP, extended to incorporate Unit 2, meets the requirements of 10 CFR 50.47(b)(3) and Section IV.A of Appendix E to 10 CFR Part 50 with regard to arrangements for emergency response support and resources. The staff bases its conclusion on (1) the approval of the WBN REP for Unit 1 as described in Supplement No. 13 to the SER, (2) TVA's demonstrated performance in maintaining emergency preparedness at the existing Unit 1, (3) the design and operational fidelity between the two units, and (4) the WBN REP's applicability to, and incorporation of, Unit 2.

13.3.2.4 Emergency Classification System

Regulatory Basis: Planning standard 10 CFR 50.47(b)(4) requires that a standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by the facility licensees for determinations of minimum initial offsite response measures. Sections IV.B and IV.C of Appendix E to 10 CFR Part 50 provide supporting requirements.

Technical Evaluation: In Section 13.3.2.4 of SSER 13, the NRC staff described its evaluation of the emergency classification scheme in the WBN REP and its finding that TVA had met the planning standard 10 CFR 50.47(b)(4) for Unit 1. The emergency classification scheme under which Unit 1 was licensed, and which is still in use, is based on NUMARC/NESP-007, "Methodology for Development of Emergency Action Levels," Revision 2, issued January 1992. This is an alternative approach endorsed in RG 1.101, "Emergency Planning and Preparedness for Nuclear Power Reactors," Revision 3. This emergency classification scheme appears in Appendix C to the WBN REP. TVA's subsequent performance in actual emergency events, periodic emergency exercises, and periodic routine and supplemental inspections continue to support the findings made by the staff regarding the WBN REP and the emergency classification scheme. By letter dated March 4, 2009, TVA proposed using the WBN REP emergency classification scheme, approved for Unit 1, for Unit 2 with changes limited to reflecting dual-unit applicability and unit-specific differences. Since the submitted changes have been minimal, no additional certification of discussion with, and agreement of, State and local official agreement under Section IV.B of Appendix E to 10 CFR Part 50 is necessary.

The NRC staff reviewed the proposed changes for dual-unit applicability and unit-specific differences and determined that it needed additional information to complete its review. TVA responded to these RAIs by letter dated April 27, 2010. Twelve of the responses contained explanatory information that would not result in changes to any emergency action level (EAL); the staff determined that these responses were acceptable. RAI Responses 3.f, 3.j, 3.o, 3.p, 3.q, 3.s, and 3.t identified specific proposed changes to subject EALs. The staff determined that the proposed changes were consistent with regulatory guidance and, therefore, were acceptable. TVA's responses to RAIs 3.e, 3.n, and 3.r remained as open items. TVA provided supplemental information in its October 11, 2010, response that enabled the staff to resolve these open items:

1. TVA's April 27, 2010, RAI Response 3.e stated that TVA would verify the radiation monitor readings used in Initiating Conditions 1.1.5 and 1.3.5 and would justify the proposed values. TVA subsequently determined that the radiation monitor readings were in error for both Unit 1 and Unit 2 and calculated replacement values. The staff reviewed the calculations submitted as part of the response and determined that TVA's assumptions, methodology, and calculated radiation monitor readings were reasonable

and consistent with the associated EAL basis and, therefore, acceptable. This item is closed.

2. TVA's April 27, 2010, RAI Response 3.n stated that TVA would evaluate the addition of information to the basis for the applicable initiating condition addressing the capability for electrical cross-feeding between the two units. TVA subsequently determined that restoring power by inter-ties between the two units would not be feasible in the context of timely emergency classification and, accordingly, decided against adding this capability to the EAL scheme. The staff found this response acceptable; this item is closed.

3. TVA's April 27, 2010, RAI Response 3.r stated that TVA would provide an updated security classification scheme that reflects Unit 2. In its October 11, 2010, response, as supplemented by letter dated November 11, 2010, TVA stated that revised security EALs were implemented in Revision 91 of Appendix C for Unit 1 under the provisions of 10 CFR 50.54(q). The Unit 1 changes were sent to the NRC in a letter dated April 28, 2010 (ADAMS Accession No. ML101241133, non-publicly available), and identical changes were included in Revision 92xx of WBN Unit 1/2, Appendix C. In the April 28, 2010, letter, TVA stated that it had implemented the security EALs in the NRC-endorsed Nuclear Energy Institute (NEI) 99-01, Revision 5, "Methodology for the Development of Emergency Action Levels," issued February 2008 (the NRC's review and endorsement is available under ADAMS Accession No. ML080430552) in a manner consistent with the guidance provided in EAL Frequently Asked Question Report No. 2009-48 (ADAMS Accession No. ML100710723). Because the 10 CFR 50.54(q) change process does apply to Unit 2 until its OL is issued, the staff compared the revised security EALs against the endorsed EALs and identified no deviations from the guidance. The staff also determined that because the revised security EALs are not unit specific, there are no unit differences or dual-unit issues raised by this revision to the WBN REP. Accordingly, the staff finds these EALs acceptable; this issue is closed.

Conclusion: The NRC staff concludes that the WBN REP, extended to incorporate Unit 2, meets the requirements of 10 CFR 50.47(b)(4) and Sections IV.B and IV.C of Appendix E to 10 CFR Part 50 with regard to the emergency classification and action level scheme. The staff bases its conclusion on (1) the approval of the WBN REP for Unit 1 as described in Supplement No. 13 to the SER, (2) TVA's demonstrated performance in maintaining emergency preparedness at the existing Unit 1, (3) the design and operational fidelity between the two units, and (4) the WBN REP's applicability to, and incorporation of, Unit 2.

13.3.2.5 Notification Methods and Procedures

Regulatory Basis: Planning standard 10 CFR 50.47(b)(5) requires that procedures are established for notification by the licensee of State and local response organizations and for notification of emergency personnel by all response organizations; the content of initial and followup messages to response organizations and the public has been established; and means to provide early notification and clear instructions to the populace within the plume exposure pathway EPZ have been established. Section IV.D of Appendix E to 10 CFR Part 50 provides supporting requirements.

Technical Evaluation: In Section 13.3.2.5 of SSER 13, the NRC staff described its evaluation of the notification methods and procedures in the WBN REP and its finding that TVA had met the planning standard 10 CFR 50.47(b)(5) for Unit 1. Section 13.3.2.17 of SSER 20 addressed the

FEMA review of the alerting and notification system (ANS). TVA's subsequent performance in actual emergency events, periodic emergency exercises, and periodic routine and supplemental inspections continue to support the findings made by the staff regarding the WBN REP. Having identified that no changes are needed to the WBN REP to reflect Unit 2, TVA proposes to use the same notification methods and procedures for Unit 2 as it has used for Unit 1. The staff finds that no dual-unit or unit-specific changes are necessary to reflect Unit 2 in that, once initiated by the Shift Manager in the affected unit, all external notifications (e.g., to Federal, State, and local officials) and ERO activation callouts are performed by the TVA Operations Duty Specialist in accordance with the previously approved WBN REP Section 5.0. Additionally, the size and shape of the WBN EPZ and, therefore, the distribution of ANS devices in the EPZ, are not affected by the licensing of Unit 2 because of the design and operational fidelity between the two units.

On March 24, 2010, FEMA issued its "Interim Finding Report (IFR) for Reasonable Assurance (RA) of the Offsite Emergency Response Plans for the Watts Bar Nuclear Site" (ADAMS Accession Nos. ML100840569 and ML100840557, both non-publicly available). Section II.E.6 of the IFR confirmed FEMA's earlier findings on the adequacy of the ANS for Unit 1. Based on its review of the FEMA determinations, the NRC staff finds that TVA has met its obligation to demonstrate under Section IV.D.3 of Appendix E to 10 CFR Part 50 that administrative and physical means have been established for alerting and informing the public.

Conclusion: The NRC staff concludes that the WBN REP, extended to incorporate Unit 2, meets the requirements of 10 CFR 50.47(b)(5) and Section IV.D of Appendix E to 10 CFR Part 50 with regard to notification means and procedures. The staff bases its conclusion on (1) the approval of the WBN REP for Unit 1 as described in SSERs 13 and 20, (2) TVA's demonstrated performance in maintaining emergency preparedness at the existing Unit 1, (3) the staff review of FEMA's March 24, 2010, findings regarding the ANS, (4) the design and operational fidelity between the two units, and (5) the WBN REP's applicability to, and incorporation of, Unit 2.

13.3.2.6 Emergency Communications

Regulatory Basis: Planning standard 10 CFR 50.47(b)(6) requires that provisions exist for prompt communications among principal response organization organizations to emergency personnel and to the public. Sections IV.E and VI of Appendix E to 10 CFR Part 50 provide supporting requirements.

Technical Evaluation: In Section 13.3.2.6 of SSER 13, the NRC staff described its evaluation of the emergency communications capabilities in the WBN REP and its finding that TVA had met the planning standard 10 CFR 50.47(b)(6) for Unit 1. TVA's subsequent performance in actual emergency events, periodic emergency exercises, and periodic routine and supplemental inspections continue to support the findings made by the staff regarding the communications capabilities described in the WBN REP. TVA proposes to use the same communication arrangements for Unit 2 as it has used for Unit 1.

In its April 27, 2010, response to RAI 6, TVA stated that Unit 1 and Unit 2 will share a common control room, TSC, and OSC and that the Unit 1 communication facilities were originally designed and built to serve both units.[3] As described in SSER 13, TVA's Central Emergency

[3] The staff reviewed the control room, TSC, and OSC, and their systems and equipment (including communications, monitoring, and assessment capabilities), in the context of both reactor units and

Control Center (CECC), located in Chattanooga, TN, serves as the EOF for Unit 1 and will serve as the EOF for Unit 2. TVA also stated in its response to RAI 7 that the Unit 2 design would provide plant data to the NRC's Emergency Response Data System (ERDS) as is currently done for Unit 1. The staff finds that no dual-unit or unit-specific changes are necessary to reflect Unit 2, in that the communication capabilities were originally designed and built to serve both units. The staff plans to confirm the availability and operability of the ERDS for Unit 2 before issuance of the Unit 2 OL. This is Open Item 38 (Appendix HH). The staff also plans to confirm the adequacy of the communications capability to support dual-unit operations before issuance of the Unit 2 OL. This is Open Item 39 (Appendix HH).

Conclusion: The NRC staff concludes that the WBN REP, extended to incorporate Unit 2, and upon satisfactory completion of the open items identified above, meets the requirements of 10 CFR 50.47(b)(6) and Sections IV.E and VI of Appendix E to 10 CFR Part 50 with regard to emergency communication capabilities and the ERDS. The staff bases its conclusion on (1) the approval of the WBN REP for Unit 1 as described in SSER 13, (2) TVA's demonstrated performance in maintaining emergency preparedness at the existing Unit 1, (3) the design and operational fidelity between the two units, and (4) the WBN REP's applicability to, and incorporation of, Unit 2.

13.3.2.7 Public Education and Information

Regulatory Basis: Planning standard 10 CFR 50.47(b)(7) requires that information is periodically made available to members of the public as to how they will be notified and what their initial actions should be in an emergency (e.g., listening to a local broadcast station and remaining indoors), the principal points of contact with the news media for dissemination of information during an emergency (including the physical location or locations) are established in advance, and procedures for coordinated dissemination of information to the public are established. Section IV.D.2 of Appendix E to 10 CFR Part 50 provides supporting requirements.

Technical Evaluation: In Section 13.3.2.7 of SSER 13, the NRC staff described its evaluation of the public education and information arrangements in the WBN REP and its finding that TVA had met the planning standard 10 CFR 50.47(b)(7) for Unit 1. TVA's subsequent performance in actual emergency events, periodic emergency exercises, and periodic routine and supplemental inspections continue to support the findings made by the staff regarding the WBN REP. Having identified that no changes are needed to the WBN REP to reflect Unit 2, TVA proposes to use the same public education and information arrangements for Unit 2 as it has used for Unit 1. The staff finds that no dual-unit or unit-specific changes are necessary to reflect Unit 2, in that public education and information arrangements are administered by the TVA corporate organization for all TVA reactor units in accordance with the previously approved WBN REP Section 7.0.

Conclusion: The NRC staff concludes that the WBN REP, extended to incorporate Unit 2, meets the requirements of 10 CFR 50.47(b)(7) and Section IV.D.2 of Appendix E to 10 CFR Part 50 with regard to public education and information arrangements. The staff bases

documented its review in NUREG-0847 issued June 1982. The staff approved the CECC on March 19, 1981, for all TVA nuclear plants. Although the sitewide WBN REP was subsequently withdrawn due to TVA's decision to delay completion of Unit 2, its earlier approval validates TVA's statement that the facilities were designed and built to serve two units. Although the capabilities were found acceptable in March 1981 and in June 1982, the staff plans to confirm the continued acceptability for dual-unit operation now that Unit 2 will be licensed.

its conclusion on (1) the approval of the WBN REP for Unit 1 as described in SSER 13, (2) TVA's demonstrated performance in maintaining emergency preparedness at the existing Unit 1, (3) the design and operational fidelity between the two units, and (4) the WBN REP's applicability to, and incorporation of, Unit 2.

13.3.2.8 Emergency Facilities and Equipment

Regulatory Basis: Planning standard 10 CFR 50.47(b)(8) requires that adequate emergency facilities and equipment to support the emergency response are provided and maintained. Section IV.E of Appendix E to 10 CFR Part 50 provides supporting requirements.

Technical Evaluation: In Section 13.3.2.8 of SSER 13,[4] the NRC staff described its evaluation of the emergency facilities and equipment in the WBN REP and its finding that TVA had met the planning standard 10 CFR 50.47(b)(8) for Unit 1. TVA's subsequent performance in actual emergency events, periodic emergency exercises, and periodic routine and supplemental inspections continue to support the findings made by the staff regarding the emergency facilities and equipment described in the WBN REP. TVA proposes to use the same emergency facilities and equipment for Unit 2 as it has used for Unit 1.

In its April 27, 2010, response to RAI 8, TVA stated that Unit 1 and Unit 2 will share a common control room, TSC, OSC, and a local recovery center and that the Unit 1 facilities in place were originally designed and built to serve both units. As described in SSER 13, TVA's CECC, located in Chattanooga, TN, serves as the EOF for Unit 1 and will serve as the EOF for Unit 2. No dual-unit or unit-specific changes are necessary to reflect Unit 2, because that the facilities and equipment were originally designed and built to serve both units. The staff should confirm the adequacy of the emergency facilities and equipment to support dual-unit operations before issuance of the Unit 2 OL. This is Open Item 40 (Appendix HH).

Conclusion: The NRC staff concludes that the WBN REP, extended to incorporate Unit 2, and upon satisfactory completion of the open items identified above, meets the requirements of 10 CFR 50.47(b)(8) and Section IV.E of Appendix E to 10 CFR Part 50 with regard to emergency facilities and equipment. The staff bases its conclusion on (1) the approval of the WBN REP for Unit 1 as described in SSER 13, (2) TVA's demonstrated performance in maintaining emergency preparedness at the existing Unit 1, (3) the design and operational fidelity between the two units, and (4) the WBN REP's applicability to, and incorporation of, Unit 2.

13.3.2.9 Accident Assessment

Regulatory Basis: Planning standard 10 CFR 50.47(b)(9) requires that adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition be provided for use. Section IV.E of Appendix E to 10 CFR Part 50 provides supporting requirements.

Technical Evaluation: In Section 13.3.2.9 of SSER 13, the NRC staff described its evaluation of the accident assessment capabilities in the WBN REP and its finding that TVA had met the

[4] TVA proposed to extend the activation time for the TSC and OSC from the 60 minutes identified in Section 13.3.2.8 of SSER 13 to 90 minutes. The activation time for the CECC was unchanged. The staff approved this proposal on June 24, 2004 (ADAMS Accession No. ML041770169). See Section 13.3.2.2 of this SSER.

planning standard 10 CFR 50.47(b)(9) for Unit 1. TVA's subsequent performance in actual emergency events, periodic emergency exercises, and periodic routine and supplemental inspections continue to support the findings made by the staff regarding the accident assessment capabilities described in the WBN REP. TVA proposes to use the same accident assessment capabilities for Unit 2 as it has used for Unit 1.

In its April 27, 2010, response to RAI 9, TVA stated that Unit 1 and Unit 2 will share a common control room, TSC, and OSC and that the Unit 1 facilities in place were originally designed and built to serve both units. As described in SSER 13, TVA's CECC, located in Chattanooga, TN, serves as the EOF for Unit 1 and will serve as the EOF for Unit 2. TVA also committed to (1) updating plant data displays as necessary to include Unit 2 and (2) updating dose assessment models to provide capabilities for assessing releases from both WBN units. The staff plans to confirm the adequacy of these items before issuance of the Unit 2 OL. This is Open Item 41 (Appendix HH).

The staff requested additional information in RAI 9 regarding the ability of the plant data displays and other assessment capabilities in the TSC, OSC, and CECC to support the response to an event simultaneously affecting both units (e.g., earthquake, tornado, flooding, loss of offsite power, hostile actions).[5] TVA's response to RAI 9 did not explain its basis for concluding that the assessment capabilities could address a simultaneous response. TVA did commit to reviewing the assessment capabilities relative to hostile actions to ensure that the strategies and guidance reflect the addition of Unit 2, but provided no proposed language for staff review. The capability for a simultaneous response is an issue because of the proposed licensing of Unit 2.

In its October 11, 2010, response, TVA stated that the WBN Integrated Computer System (ICS) will have added components specific to Unit 2 and the Unit 1 graphic display screens will be modified to provide Unit 2 and common data displays. Although there is only one ICS, each WBN unit will have displays specific to that unit. Although specific to each unit, the displays will provide that unit's data, common data, and any of the other unit's data deemed necessary. Display of data for both units will be available in the main control room, TSC, OSC, and CECC. New ICS hardware, including workstations, monitors, and large screen displays, will be provided in the TSC. This equipment will be able to access and display ICS data for both units. TVA stated that the computerized dose models in the CECC provide capability for assessing releases from both WBN units and that manual dose assessment methods described in the WBN REP are being updated to provide a capability for dose assessments for both units. TVA also stated that strategies and guidance related to hostile actions are being updated to reflect the addition of Unit 2.

Since these actions are currently in progress, the staff will confirm the adequacy of these assessment capabilities to support dual unit operations before issuance of the Unit 2 OL. This is Open Item 42 (Appendix HH).

Conclusion: The NRC staff concludes that the WBN REP, extended to incorporate Unit 2, and upon satisfactory completion of the open items identified above, meets the requirements of 10 CFR 50.47(b)(9) and Section IV.E of Appendix E to 10 CFR Part 50 with regard to accident assessment. The staff bases its conclusion on (1) the approval of the WBN REP for Unit 1 as described in Supplement No. 13 to the SER, (2) TVA's demonstrated performance in

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5 The staff's request is not postulating an initiating event occurring simultaneously at each of the two units, but rather, a single initiating event that affects both units.

maintaining emergency preparedness at the existing Unit 1, (3) the design and operational fidelity between the two units, and (4) the WBN REP's applicability to, and incorporation of, Unit 2.

13.3.2.10 Protective Response

Regulatory Basis: Planning standard 10 CFR 50.47(b)(10) requires that a range of protective actions be developed for the plume exposure pathway EPZ for emergency workers and the public. In the development of this range of actions, consideration has been given to evacuation, sheltering, and, as a supplement to these, the prophylactic use of potassium iodine (KI) as appropriate. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are in place, and protective actions for the ingestion exposure pathway EPZ appropriate to the locale have been developed. Section II.C of Appendix E to 10 CFR Part 50 provides supporting requirements.

Technical Evaluation: In Section 13.3.2.10 of SSER 13, the NRC staff described its evaluation of the protective action arrangements presented in the WBN REP and its finding that TVA had met the planning standard 10 CFR 50.47(b)(10) for Unit 1. TVA's subsequent performance in actual emergency events, periodic emergency exercises, and periodic routine and supplemental inspections continue to support the findings made by the staff regarding the protective action arrangements described in the WBN REP. TVA proposes to use the same protective action protocols for Unit 2 as it has used for Unit 1 and TVA's other licensed reactor units. The staff finds that no dual-unit or unit-specific changes are necessary to reflect Unit 2 because the decision criteria, methods, and implementation resources for protective actions are not unit specific, given the design and operational fidelity between the two units.

Conclusion: The NRC staff concludes that the WBN REP, extended to incorporate Unit 2, meets the requirements of 10 CFR 50.47(b)(10) with regard to protective response. The staff bases its conclusion on (1) the approval of the WBN REP for Unit 1 as described in Supplement No. 13 to the SER, (2) TVA's demonstrated performance in maintaining emergency preparedness at the existing Unit 1, (3) the design and operational fidelity between the two units, and (4) the WBN REP's applicability to, and incorporation of, Unit 2.

13.3.2.11 Radiological Exposure Control

Regulatory Basis: Planning standard 10 CFR 50.47(b)(11) requires that means for controlling radiological exposures, in an emergency, are established for emergency workers. The means for controlling radiological exposures shall include exposure guidelines consistent with U.S. Environmental Protection Agency (EPA) Emergency Worker and Lifesaving Activity Protective Action Guides.[6]

Technical Evaluation: In Section 13.3.2.11 of SSER 13, the NRC staff described its evaluation of the radiological exposure control arrangements described in the WBN REP and its finding that TVA had met the planning standard 10 CFR 50.47(b)(11) for Unit 1. TVA's subsequent performance in actual emergency events, periodic emergency exercises, and periodic routine and supplemental inspections continue to support the findings made by the staff regarding the radiological exposure control arrangements described in the WBN REP. TVA proposes to use

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6 These guides now appear in Section 2.5, "EPA Guidance for Controlling Doses to Workers under Emergency Conditions," of EPA 400-R-92-001, ""Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," issued May 1992.

the same arrangements for Unit 2 as it has used for Unit 1 and TVA's other licensed reactor units. The staff finds that no dual-unit or unit-specific changes are necessary to reflect Unit 2 because the decision criteria, methods, and implementation resources for radiological exposure control are not inherently unit specific, given the design and operational fidelity between the two units.

Conclusion: The NRC staff concludes that the WBN REP, extended to incorporate Unit 2, meets the requirements of 10 CFR 50.47(b)(11) with regard to radiological exposure control arrangements. The staff bases its conclusion on (1) the approval of the WBN REP for Unit 1 as described in Supplement No. 13 to the SER, (2) TVA's demonstrated performance in maintaining emergency preparedness at the existing Unit 1, (3) the design and operational fidelity between the two units, and (4) the WBN REP's applicability to, and incorporation of, Unit 2.

13.3.2.12 Medical and Public Health Support

Regulatory Basis: Planning standard 10 CFR 50.47(b)(12) requires that arrangements be made for medical services for contaminated injured individuals.

Technical Evaluation: In Section 13.3.2.12 of SSER 13, the NRC staff described its evaluation of the medical support arrangements presented in the WBN REP and its finding that TVA had met the planning standard 10 CFR 50.47(b)(12) for Unit 1. TVA's subsequent performance in actual emergency events, periodic emergency exercises, and periodic routine and supplemental inspections continue to support the findings made by the staff regarding the medical support arrangements described in the WBN REP. In its April 27, 2010, response to RAI 10, TVA described the makeup and responsibilities of the Medical Emergency Response Team, an onsite team drawn from operations, fire operations, radiation protection, medical services, and nuclear security. TVA proposes to use the same arrangements for Unit 2 as it has used for Unit 1 and TVA's other licensed reactor units. The staff finds that no dual-unit or unit-specific changes are necessary to reflect Unit 2 because the arrangements for medical support are not inherently unit specific, given the design and operational fidelity between the two units.

Conclusion: The NRC staff concludes that the WBN REP, extended to incorporate Unit 2, is acceptable for Unit 2 and that it meets the requirements of 10 CFR 50.47(b)(12) with regard to medical support arrangements. The staff bases its conclusion on (1) the approval of the WBN REP for Unit 1 as described in Supplement No. 13 to the SER, (2) TVA's demonstrated performance in maintaining emergency preparedness at the existing Unit 1, (3) the design and operational fidelity between the two units, and (4) the WBN REP's applicability to, and incorporation of, Unit 2.

13.3.2.13 Recovery and Reentry Planning and Postaccident Operations

Regulatory Basis: Planning standard 10 CFR 50.47(b)(13) requires that general plans for recovery and reentry are developed.

Technical Evaluation: In Section 13.3.2.13 of SSER 13, the NRC staff described its evaluation of the recovery and reentry arrangements presented in the WBN REP and its finding that TVA had met the planning standard 10 CFR 50.47(b)(13) for Unit 1. TVA's subsequent performance in actual emergency events, periodic emergency exercises, and periodic routine and supplemental inspections continue to support the findings made by the staff regarding the recovery and reentry arrangements described in the WBN REP. TVA proposes to use the same arrangements for Unit 2 as it has used for Unit 1 and TVA's other licensed reactor units. The staff finds that no dual-unit or unit-specific changes are necessary to reflect Unit 2 because general plans for recovery and reentry and associated decision criteria, methods, and implementation resources are not inherently unit specific, given the design and operational fidelity between both units.

Conclusion: The NRC staff concludes that the WBN REP, extended to incorporate Unit 2, meets the requirements of 10 CFR 50.47(b)(13) with regard to recovery and reentry arrangements. The staff bases its conclusion on (1) the approval of the WBN REP for Unit 1 as described in Supplement No. 13 to the SER, (2) TVA's demonstrated performance in maintaining emergency preparedness at the existing Unit 1, (3) the design and operational fidelity between the two units, and (4) the WBN REP's applicability to, and incorporation of, Unit 2.

13.3.2.14 Exercises and Drills

Regulatory Basis: Planning standard 10 CFR 50.47(b)(14) requires that periodic exercises be conducted to evaluate major portions of emergency response capabilities, that periodic drills be conducted to develop and maintain key skills, and that deficiencies identified as a result of exercises or drills be corrected. Section IV.F of Appendix E to 10 CFR Part 50 provides supporting requirements.

Technical Evaluation: In Section 13.3.2.14 of SSER 13, the NRC staff described its evaluation of the exercises and drill program described in the WBN REP and its finding that TVA had met the planning standard 10 CFR 50.47(b)(14) and the requirements of 10 CFR Part 50, Appendix E, Section IV.F, for Unit 1. TVA's subsequent performance in actual emergency events, periodic emergency exercises, and periodic routine and supplemental inspections continue to support the findings made by the staff regarding the exercise and drill program described in the WBN REP. The most recent Unit 1 emergency exercise was conducted on June 10, 2009, and the NRC identified no findings of significance (ADAMS Accession No. ML091980360). TVA proposes to use the same program for Unit 2 as it has used for Unit 1 and TVA's other licensed reactor units. In its April 27, 2010, response to RAI 11, TVA stated that the addition of WBN Unit 2 adds no new skill sets or knowledge requirements for the ERO that would warrant or require additional drills or exercises. The staff finds that no dual-unit or unit-specific changes are necessary for the drills and exercise program to reflect Unit 2 because of the design and operational fidelity between Unit 1 and Unit 2.

Section IV.F.2.a of Appendix E to 10 CFR Part 50 requires that a full-participation exercise be conducted within a certain time window before the issuance of the first OL for full power of the first reactor on a site. TVA complied with this requirement by conducting evaluated exercises at Unit 1 on November 15, 1995 (documented in NRC Inspection Report 50-390/95-78); October 26, 1995; and October 6, 1993. The October 6, 1993, and November 15, 1995, exercises involved the participation of State and local response organizations and were evaluated by FEMA as the qualifying exercises. The staff finds no significant rationale for requiring such an exercise specifically for Unit 2. The staff bases its conclusion on (1) the design and operational fidelity between the two units and (2) TVA's demonstrated performance in biennial exercises conducted since the licensing of Unit 1.

Conclusion: The NRC staff concludes that the WBN REP, extended to incorporate Unit 2, meets the requirements of 10 CFR 50.47(b)(14) with regard to the exercises and drill program. The staff bases its conclusion on (1) the approval of the WBN REP for Unit 1 as described in

Supplement No. 13 to the SER, (2) the successful execution of the full-participation exercise required by Appendix E, Section IV.F.2.a, (3) TVA's demonstrated performance in maintaining emergency preparedness at the existing Unit 1, (4) the design and operational fidelity between the two units, and (5) the WBN REP's applicability to, and incorporation of, Unit 2.

13.3.2.15 Radiological Emergency Response Training

Regulatory Basis: Planning standard 10 CFR 50.47(b)(15) requires that radiological emergency response training is provided to those who may be called upon to assist in an emergency. Section IV.F of Appendix E to 10 CFR Part 50 provides supporting requirements.

Technical Evaluation: In Section 13.3.2.15 of SSER 13, the NRC staff described its evaluation of the radiological emergency response training program presented in the WBN REP and its finding that TVA had met the planning standard 10 CFR 50.47(b)(15) for Unit 1. TVA's subsequent performance in actual emergency events, periodic emergency exercises, and periodic routine and supplemental inspections continue to support the findings made by the staff regarding the radiological emergency response training program described in the WBN REP. TVA proposes to use the same arrangements for Unit 2 as it has used for Unit 1 and TVA's other licensed reactor units. The staff finds that no dual-unit or unit-specific changes are necessary to reflect Unit 2 because the knowledge and skill sets for the ERO will be essentially the same for both units, given the design and operational fidelity between the two units.

Conclusion: The NRC staff concludes that the WBN REP, extended to incorporate Unit 2, meets the requirements of 10 CFR 50.47(b)(15) with regard to the radiological emergency response training program. The staff bases its conclusion on (1) the approval of the WBN REP for Unit 1 as described in SSER 13, (2) TVA's demonstrated performance in maintaining emergency preparedness at the existing Unit 1, (3) the design and operational fidelity between the two units, and (4) the WBN REP's applicability to, and incorporation of, Unit 2.

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

Regulatory Basis: Planning standard 10 CFR 50.47(b)(16) requires that responsibilities for plan development and review and for distribution of emergency plans are established and that planners are properly trained.

Technical Evaluation: In Section 13.3.2.16 of SSER 13, the NRC staff described its evaluation of the responsibilities for the planning effort and the qualification of planners presented in the WBN REP and its finding that TVA had met the planning standard 10 CFR 50.47(b)(16) for Unit 1. TVA's subsequent performance in actual emergency events, periodic emergency exercises, and periodic routine and supplemental inspections continue to support the findings made by the staff regarding the planning effort described in the WBN REP. TVA proposes to use the same arrangements for Unit 2 as it has used for Unit 1 and TVA's other licensed reactor units. The staff finds that no dual-unit or unit-specific changes are necessary to reflect Unit 2 because responsibilities for the planning effort and the qualification of the planners are not inherently unit specific since the TVA corporate radiological emergency plan organization is responsible for the planning effort.

Conclusion: The NRC staff concludes that the WBN REP, extended to incorporate Unit 2, meets the requirements of 10 CFR 50.47(b)(16) with regard to the responsibilities for the planning effort and the qualification of planners. The staff bases its conclusion on (1) the approval of the WBN REP for Unit 1 as described in SSER 13, (2) TVA's demonstrated performance in maintaining emergency preparedness at the existing Unit 1, (3) the design and operational fidelity between the two units, and (4) the WBN REP's applicability to, and incorporation of, Unit 2.

13.3.2.17 Evaluation of Offsite Emergency Preparedness

Regulatory Basis: See the first and last paragraphs of Section 13.3.1.3 of this SER supplement.

Technical Evaluation: During 2009 and 2010, FEMA conducted a review of the offsite emergency response plans for the WBN site (i.e., MJRERP) and prepared an IFR documenting the results of its review (ADAMS Accession Nos. ML100840569 and ML100840557, dated March 24, 2010, non-publicly available). FEMA performed this review in accordance with its regulations, its applicable planning standards, and the evaluation criteria established in NUREG-0654/FEMA-REP-1. FEMA noted in its IFR that the licensing and design bases of WBN Unit 1 and Unit 2 are essentially the same and that Unit 1 had successfully completed six biennial plume exposure exercises and two ingestion phase exercises, with State and county participation, since it was licensed in 1996. FEMA noted that an offsite response utilizing the current State and county plans and procedures would be the same no matter which unit at the WBN site was involved. The IFR dated October 6, 2009, concluded the following:

> After a thorough review of currently available Offsite Plans and Procedures, based upon the standards and criteria of NUREG-0654 FEMA-REP-1, Rev. 1, a determination has been made that the plans are adequate and there is Reasonable Assurance that the plans can be implemented with no corrections needed. A qualifying REPEX will be demonstrated by the Watts Bar Nuclear Plant Site in conjunction with Offsite State and County's participation.

The NRC staff requested clarification from FEMA regarding the statement that a qualifying REP exercise (REPEX) would be required for the licensing of WBN Unit 2. Section IV.F.2.a of Appendix E to 10 CFR Part 50 requires that a full-participation exercise be conducted within a certain time window before the issuance of the first OL for full power of the first reactor on a site. TVA complied with this requirement by conducting evaluated exercises at Unit 1 on November 15, 1995 (documented in NRC Inspection Report 50-390/95-78), October 26, 1995, and October 6, 1993. The October 6, 1993, and November 15, 1995, exercises involved the participation of State and local response organizations and were evaluated by FEMA as the qualifying exercises.

By letter dated August 19, 2010 (ADAMS Accession No. ML102350530) and supplemented in an e-mail dated September 23, 2010 (ADAMS Accession No. ML103130537, non-publicly available), FEMA advised the staff that if TVA developed a WBN REP that is essentially identical regardless of reactor unit, the June 10, 2009, full-participation exercise for WBN Unit 1 is adequate for FEMA's purpose of determining reasonable assurance for the Watts Bar site. In its letter dated October 6, 2009 (ADAMS Accession No. ML102600549, non-publicly available), FEMA provided the NRC with a copy of its final exercise report and advised the staff as follows:

> Based on the results of the June 10, 2009 exercise and FEMA's review of Tennessee's Annual Letters of Certification for 2008 and 2009, the offsite radiological emergency response plans and preparedness for the State of Tennessee and the affected local jurisdictions, site-specific to the Watts Bar Nuclear Plant, can be implemented and are adequate to provide reasonable

assurance that appropriate measures can be taken offsite to protect the health and safety of the public in the event of a radiological emergency at the site.

No findings of significance were identified on site (NRC Inspection Report 05000390/200950, dated July 17, 2009 (ADAMS Accession No. ML091980360)). The staff reviewed the FEMA IFR, the exercise report, and the correspondence as required by 10 CFR 50.47(a)(2) and found the bases of the FEMA findings and determinations to be understandable and clear. The staff review did not identify any discrepancies in the integration of the licensee's plan with those of State and local authorities. As discussed in this SSER, TVA has demonstrated the design and operational fidelity of the two units and has presented an acceptable emergency plan that is equally applicable to Unit 1 and Unit 2.

In Section 13.3.2.17 of SSER 20, the NRC staff described its evaluation of the evacuation time estimate (ETE). The requirement for an analysis of the time required to evacuate is in Section IV of Appendix E to 10 CFR Part 50. Guidance is provided in NUREG-0654/FEMA-REP-1, Revision 1, Section II.J.10.l and Appendix 4. The WBN ETE (located in Annex H to the Tennessee MJRERP) was last updated with the 2000 census and was reviewed by the State in 2009. In SSER 20, the staff found the placement of the ETE in the MJRERP, rather than in the WBN REP, to be acceptable.

Under contract from the NRC staff, technical experts from Sandia National Laboratories reviewed the ETE and prepared a technical evaluation report, "JCN No: 4201, (TAC No. ME0853) Q4011 Task Order Number 9, Tennessee Valley Authority, Watts Bar Unit 2, Evacuation Time Estimate (ETE) Review," dated September 27, 2010 (ADAMS Accession No. ML102730565, non-publicly available), containing the results of the review. During the review, the NRC contractor determined that additional information was needed to complete the review. TVA provided additional information by letter dated July 15, 2010, as supplemented by letters dated August 13, 2010, and August 26, 2010. The contractor reviewed the ETE and supporting annexes and appendices; performed an independent evaluation of demographics; reviewed publicly available information on schools and special facilities; reviewed aerial photography of the road network; reviewed the analysis methodology; and reviewed assumptions regarding automobile occupancy and roadway capacity factors, travel speeds, and preparation times. The NRC staff reviewed the contractor's assessment of the content of the ETE and its conformance with regulatory guidance. The staff concluded that the WBN ETE is consistent with the guidance in Appendix 4 to NUREG–0654/FEMA-REP-1 and meets the applicable requirements of 10 CFR Part 50, Appendix E.IV, and is, therefore, acceptable.

Conclusion: Based on its review of the (1) FEMA findings on the offsite plans documented in the IFR, (2) FEMA findings on the June 10, 2009, WBN Unit 1 full-participation exercise for State and local agencies, and (3) FEMA-NRC correspondence discussed above, the staff finds, pursuant to 10 CFR 50.47(a)(2), that the State and local emergency plans for the WBN site provide an adequate planning basis and that there is reasonable assurance that the State and local emergency plans can be implemented.

13.3.2.18 Other Matters

On March 11, 2010, the NRC staff asked TVA for additional information regarding why various actions performed in response to NRC Bulletin 2005-02, "Emergency Preparedness and Response Actions for Security-Based Events," dated July 18, 2005, as identified in TVA's letter dated August 16, 2005 (ADAMS Accession No. ML052310130, non-publicly available), did not appear to be documented in the WBN REP. In its April 27, 2010, response to RAI 12, TVA stated that these actions were documented in various EPIPs and provided specific citations. In the absence of explicit requirements that these actions be described in the WBN REP, the NRC staff concluded that locating this information in EPIPs is acceptable in regard to NRC Bulletin 2005-02.

The NRC has issued numerous generic communications addressing emergency preparedness issues. By letters dated September 7, 2007, March 20, 2008, and July 29, 2008, TVA provided a summary of its responses to NRC generic communications issued before 1995 for Unit 2 and those issued after Unit 1 was licensed. The NRC staff identified those generic communications that remained open in SSER 21. No generic communications related to emergency preparedness, including NUREG-0737 and NUREG-0696, are open. Any such generic communication would have been addressed, in the context of Unit 1, in the WBN REP, which TVA is now proposing to extend to Unit 2. Based on the design and operational fidelity between Unit 1 and Unit 2, the NRC staff concludes that TVA has adequately addressed these generic communications with regard to Unit 2.

In October 1985, TVA identified and proposed 18 corrective action programs and 11 special programs to address concerns regarding the management of its nuclear program. None of these programs involved emergency preparedness issues or systems.

By letter dated October 11, 2007, TVA stated that Unit 2 did not require any exemptions from regulation that had previously been approved for Unit 1.

Section V of Appendix E to 10 CFR Part 50 requires TVA to submit its detailed implementing procedures for its emergency plan no less than 180 days before the scheduled issuance of an OL. Completion of this requirement is an open item that must be resolved before the issuance of an OL. This is Open Item 43 (Appendix HH).

On May 18, 2009, the NRC announced a proposed rulemaking that would revise certain aspects of the Commission's emergency preparedness regulations. The agency has prepared the final rule and expects to issue it in the third quarter of 2011. Should the final rule be issued in its current form before the issuance of the Unit 2 OL, TVA would have the same periods for implementing the various amended requirements for Unit 2 as for Unit 1. This protocol reflects NRC support of a licensing review approach that employs the current licensing basis for Unit 1 as the reference basis for review and licensing of Unit 2, as stated in SRM-SECY-07-0096, "Possible Reactivation of Construction and Licensing Activities for the Watts Bar Nuclear Plant Unit 2," dated July 25, 2007.

13.3.3 Conclusion

On the basis of its review of the WBN REP, the results of onsite inspections, and its evaluation of TVA's performance in exercises, the NRC staff concluded in SSER 20 that the WBN REP provided an adequate basis for an acceptable state of onsite emergency preparedness and that the WBN REP met the requirements of 10 CFR 50.47(b), Appendix E to 10 CFR Part 50, with regard to Unit 1.

In its review, the NRC staff used the guidance of Section 13.3 of the SRP to verify that the WBN REP, as approved for Unit 1, (1) is applicable to Unit 2, (2) is up to date when the application is submitted, and (3) reflects use of the site for the construction of a new reactor and appropriately incorporates the new reactor into the existing WBN REP. The staff has not performed a complete re-review of the WBN REP, since the staff previously established its acceptability in SSER 20. Instead, the staff focused its review on whether the WBN REP adequately encompasses any differences between Unit 1 and Unit 2 and that any dual-unit issues raised by the licensing of Unit 2 have been addressed.

On the basis of its review, the NRC staff concludes that the WBN REP (i.e., Revision 92 of the site-independent TVA REP and the site-specific WBN Unit 1/2, Appendix C, Revision 92xx), upon satisfactory completion of the confirmatory items identified above, will provide an adequate planning basis for an acceptable state of emergency preparedness for WBN Units 1 and 2. The staff also finds that the WBN REP complies with the planning standards of 10 CFR 50.47(b) and the requirements of Appendix E to 10 CFR Part 50 and that there is reasonable assurance that the WBN REP can be implemented.

FEMA has provided its findings and determinations on the adequacy of offsite emergency planning and preparedness, based on its plan reviews, exercise observations, and analyses. On the basis of its review of the FEMA findings, the NRC staff concludes, pursuant to 10 CFR 50.47(a)(2), that the WBN offsite emergency plans (MJRERP) provide an adequate planning basis and that there is reasonable assurance that they can be implemented.

Accordingly, the NRC staff concludes that, pursuant to 10 CFR 50.47(a)(1)(i), and subject to the satisfactory completion of the confirmatory items identified above, there is reasonable assurance that adequate protective measures can and will be taken in a radiological emergency at either WBN Unit 1 or Unit 2.

13.4 Review and Audit

As stated in 10 CFR 50.36(c)(5), the technical specifications (TS) will include administrative controls. The administrative controls are the provisions relating to organization and management, procedures, recordkeeping, and reporting that are necessary to ensure safe operation of the facility. In this regard, TVA's corporate quality assurance (QA) program, described in TVA-NQA-PLN89-A, "[TVA] Nuclear Quality Assurance Program" (NQA Plan), addresses review and audit requirements. The NRC had previously reviewed and approved the TVA NQA Plan and plant-specific appendices for use at TVA's facilities, including WBN Unit 1.

Section 9.9 of the NQA Plan describes the WBN plant review program. As part of the review of the OL application for WBN Unit 1, the NRC staff evaluated the program in accordance with SRP Section 13.4, "Operational Review." SRP Section 5.2.15 describes an acceptable method for complying with 10 CFR Part 50, Appendix B, Criterion VI, "Document Control," including review and approval of procedures and changes to procedures. SRP Section 4.4 describes an acceptable method for complying with the Commission's regulations related to onsite review activities. Based on these evaluations, the staff determined that TVA's NQA Plan was acceptable for use at WBN Unit 1.

Pursuant to 10 CFR 50.54(a)(3), each licensee may make a change to a previously accepted QA program description included or referenced in the safety analysis report without prior NRC approval, provided that the change does not reduce the commitments in the program description as accepted by the NRC. Further, the requirements in 10 CFR 50.54(a)(4) state that changes to the QA program description that reduce the commitments must be submitted to the NRC and receive NRC approval prior to implementation.

By letters dated March 22, 2000, as supplemented on April 25 and June 13, 2000, TVA proposed to revise the NQA Plan to standardize the requirements for plant reviews and the Plant Operations Review Committee (PORC). TVA submitted these changes as a reduction in commitment for NRC review and approval in accordance with 10 CFR 50.54(a)(4). TVA stated that the standardization was consistent with its current regulatory commitment to American National Standards Institute (ANSI) Standard N18.7, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants" [ANSI N18.7/ANS-3.2], as endorsed by RG 1.33, "Quality Assurance Program Requirements." By letter dated July 25, 2000, the NRC staff concluded that TVA's proposed reduction in commitment conformed to the applicable guidelines of ANSI N18.7-1976, as endorsed by Regulatory Guide 1.33, Revision 2, and continued to satisfy the criteria of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 and applicable regulatory administrative control requirements and was, therefore, acceptable.

By letter dated January 11, 2010, TVA submitted WBN Unit 2 FSAR Amendment 97, which, in part, included the onsite review and the independent review and audit. The description of the organizations and performance of these review functions was provided by reference to the NQA Plan in TVA-NQA-PLN89-A. The NRC staff reviewed changes to the FSAR along with TVA's NQA Plan, Revision 23, and related correspondence. Specifically, the staff reviewed the following documents in the process of this evaluation:

1. letter from Mark J. Burzynski (TVA) to the NRC, dated March 22, 2000—proposed revision to PORC and plant reviews standardization

2. letter from Mark J. Burzynski (TVA) to the NRC, dated April 25, 2000—superseded letter dated March 22, 2000, on revision to PORC and plant reviews standardization

3. letter from Mark J. Burzynski (TVA) to the NRC, dated June 13, 2000—in response to questions concerning proposed revision to PORC and plant reviews standardization

4. letter from Richard P. Correia (NRC) to TVA, dated July 25, 2000—safety evaluation of QA program with finding of reduction in commitment continuing to meet regulatory requirements

The principal changes proposed by TVA were the elimination of plant reviews that are unique to individual nuclear sites, a reduction in the threshold of items requiring PORC review, and a change of PORC composition requirements. As discussed above, the NRC staff previously approved these changes and, in its letter dated July 25, 2000, concluded that TVA's proposed reduction in commitment continued to satisfy the criteria of Appendix B to 10 CFR Part 50 and applicable regulatory administrative control requirements and was therefore acceptable. The staff reviewed TVA-NQA-PLN89-A and determined that it continues to conform to the applicable guidelines of ANSI N18.7-1976, as endorsed by RG 1.33, Revision 2.

Summary and Conclusion

TVA's review and audit administrative requirements conform to the applicable guidelines of ANSI N18.7-1976, as endorsed by RG 1.33, Revision 2. The plant review process is consistent with the applicable regulatory guidelines. The NRC staff concludes that the plant review process described in FSAR Section 13.4 and the TVA NQA Plan is consistent with applicable regulatory guidelines, will continue to satisfy the criteria of Appendix B to 10 CFR Part 50, and therefore is acceptable.

13.5 Plant Procedures

In its letter dated January 11, 2010, TVA submitted FSAR Amendment 97, which proposed changes to FSAR Sections 13.5.1, "Administrative Procedures," and 13.5.2, "Operating Maintenance Procedures."

In 10 CFR 50.34, "Contents of Applications; Technical Information," the NRC provides regulations that address the required content of the FSAR for applicants. Specifically relevant to this evaluation, 10 CFR 50.34(f)(2)(ii) provides requirements for applicants to establish programs to integrate and improve plant procedures. This section requires that the scope of the programs include emergency procedures and human factors engineering considerations. The regulation in 10 CFR 50.34(f)(2)(iii) also requires a control room design that reflects state-of-the-art human factor principles. In 10 CFR 50.34(b)(6)(ii), the NRC states that administrative controls are to be used to ensure safe operation. Appendix B to 10 CFR Part 50 sets forth the requirements for controls for nuclear power plants.

RG 1.33 provides guidance on acceptable applications. This guidance describes a method acceptable to the NRC staff for complying with regulations related to overall QA program requirements for the operation phase of nuclear power plants as stated in Appendix B to 10 CFR Part 50. This guide endorses ANSI N18.7-1976/American Nuclear Society (ANS)-3.2, with modifications. ANSI N18.7-1976/ANS 3.2 requires the preparation of many procedures to carry out an effective QA program. This document provides guidance to ensure written procedural coverage for plant operating activities, including related maintenance activities. It also provides guidance on the preparation of plant procedures.

Following the accident at TMI Unit 2 in 1979, the NRC staff developed the TMI Task Action Plan, to provide a comprehensive and integrated plan to improve safety at power reactors. In NUREG-0737, Supplement 1, issued January 1983, these specific items constitute a single document, which includes additional information about schedules, applicability, method of implementation review, submittal dates, and clarification of technical positions. The specific items relevant to this evaluation include administrative procedures, normal operating procedures, emergency operating procedures, maintenance activities, and control room design reviews. NUREG-0737 provides information on the procedures that should be developed to provide administrative controls that could affect plant safety. NUREG-0737 also provides guidance on the evaluation and development of operating and emergency procedures. Additionally, NUREG-0737 provides guidance on the requirement that all licensees and applicants for OLs conduct a detailed control room design review to identify and correct design deficiencies.

NUREG-0899, "Guidelines for Preparation of Emergency Operating Procedures—Resolution of Comments on NUREG-0799," dated September 3, 1982, identifies the elements necessary for utilities to prepare and implement a program of emergency operating procedures for use by control room personnel to assist in mitigating the consequences of a broad range of accidents and multiple equipment failures. The NRC issued this document to assist applicants in meeting, and to ensure that licensees continue to meet, the requirements of 10 CFR 50.34. The document explains the process by which applicants and licensees should develop, implement, and maintain emergency operating procedures.

SRP Section 13.5.2.1 contains specific review criteria. Section 13.5.2.1 provides guidance regarding the operating procedures that will be used by the operating organization.

In the SER, the NRC staff reviewed the following:

- TVA's description of the program and procedures that provide administrative controls over activities important to safety
- the plan for development and implementation of operating and maintenance procedures
- control room design via an onsite review from October 6 through October 10, 1980

In this review, the NRC identified many deficiencies shortly before the WBN Unit 1 OL was expected to be issued. By letter dated September 17, 1985, the NRC, pursuant to 10 CFR 50.54(f), requested that TVA submit information on its plans for correcting problems concerning the overall management of its nuclear program and for correcting plant-specific problems. Subsequently, TVA developed a corporate nuclear performance plan (NPP) that identified and proposed corrections to problems concerning the overall management of its nuclear program and a site-specific plan for WBN entitled, "Watts Bar Nuclear Performance Plan." TVA submitted its NPP to the NRC by letter dated November 1, 1985. The NRC staff issued its safety evaluation of the revised NPP as NUREG-1232, Volume 1, "Safety Evaluation Report on Tennessee Valley Authority: Revised Corporate Nuclear Performance Plan."

SSER 21 addressed the topics in the SER and SSERs, as well as the issues discussed in NUREG-1232 and the WBN FSAR through Amendment 91. SSER 21 lists the Administrative Procedures, Operating and Maintenance Procedures, and Control Room Design sections as open items.

13.5.1 Administrative Procedures

TVA submitted FSAR Amendment 44 for both WBN units in 1981 with a description in Section 13.5 of the administrative procedures. This section described the conformance with regulatory guides and preparation of procedures and listed the activities categorized as administrative procedures. TVA stated that it would use ANSI Standard N18.7-1976 [ANSI N18.7-1976/ANS-3.2] as guidance for the preparation of administrative procedures. The section described the preparation of procedures, the PORC, and the position of the Plant Superintendent. The submittal stated that procedures covering plant operations, maintenance work, tests, equipment changes, and other activities that might adversely affect safety are put into effect only after being reviewed by the PORC and with written authorization from the Plant Superintendent. This section also stated that the Plant Superintendent is responsible for ensuring that required reviews and approvals are completed before authorizations are issued. The submittal described the procedures categorized as administrative, including those for changes to the plant, for changes to procedures, and for defining the responsibilities for Reactor

Operators and Senior Reactor Operators. The submittal also described the responsibilities of key personnel, including the Shift Technical Advisor and the Shift Supervisor.

In the SER, the NRC staff reviewed TVA's description of the program and procedures that provide administrative controls over activities important to safety. The staff concluded that the administrative procedures information presented in the Watts Bar safety analysis report is acceptable and contributes to meeting the requirements of the relevant sections of 10 CFR 50.34. The staff found that the provisions met the requirements in the guidance described in Section 5.2 of ANSI/ANS 3.2, which is endorsed by RG 1.33. The staff also found that the provisions met the applicable parts of the TMI Action Plan Requirements, items I.A.1.2, I.A.1.3, I.C.2, I.C.3, I.C.4, I.C.4, I.C.5, and I.C.6.

In 2010, TVA submitted FSAR Amendment 97 for WBN Unit 2. The structure of the section of the report pertaining to administrative procedures has been updated subsequent to the NRC SER, which determined that administrative procedures were acceptable for Unit 1. The portion of the report pertaining to issuance of procedures continues to follow the guidance of RG 1.33, with the updated language referring directly to the guide instead of to the ANSI standard it endorses. The NRC staff concludes that the administrative procedures information presented in FSAR Amendment 97 continues to be in compliance with the requirements of 10 CFR 50.34. The staff also finds that the changes meet the applicable parts of the NUREG-0737, TMI Action Plan Requirements by including administrative procedural provisions in FSAR Section 13.5.1.3. Based on its review of FSAR Amendment 97, and the previous staff evaluation documented in the SER and its supplements, the NRC staff concludes that the administrative procedures meet the relevant requirements of NUREG-0737 and 10 CFR 50.34 and the guidance of the relevant regulatory guides and is therefore acceptable.

13.5.2 Operating and Maintenance Procedures

TVA submitted FSAR Amendment 44 (for both WBN units) in 1981 with a description in Section 13.5.2 of the operating and maintenance procedures. This section described the general, system, abnormal, emergency, and refueling procedures.

In the SER, the NRC staff's review of Section 13.5.2 of the FSAR identified that TVA's plans for the preparation, review, approval, and use of written procedures at WBN are essentially identical to those for the Sequoyah Nuclear Plant (Sequoyah). Therefore, the discussion and conclusions regarding the initial operating and maintenance procedures in Section 13.5 of the Sequoyah SER (NUREG-0011) apply to WBN. Sequoyah SSER 2, Section 15.2, provides further discussion. The procedures noted some necessary differences resulting from differences in the design of the plants.

In 2010, TVA submitted FSAR Amendment 97 for WBN Unit 2. The section of the report pertaining to operating and maintenance procedures has been updated in structure. The content of this section satisfies the relevant portions of RG 1.33 and the TMI Action Plan Requirements. This section of the FSAR describes the different classifications of procedures that the operators will use in the control room and locally in the plant for plant operations. As with the administrative procedures, the FSAR describes TVA's program for developing the operating and emergency procedures in the section of the FSAR that follows the guidance of RG 1.33. The FSAR identified the individuals responsible for maintaining the procedures and the general format and content of the operating and maintenance procedures including emergency operating procedures. The different classifications of procedures and maintenance activities were also described. The FSAR addressed the following categories of procedures:

- general
- system
- operating
- abnormal
- emergency
- fuel handling
- maintenance
- modification

The identification of the individuals responsible and the descriptions of the content of the operating and maintenance procedures were in accordance with NUREG-0800. Based on this and the previous staff evaluation documented in the SER and its supplements, the NRC staff concludes that the operating and maintenance procedures are acceptable for WBN Unit 2.

13.5.3 NUREG-0737 Items

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

In SSER 16, the NRC staff concluded in Section 13.5.3 that TVA's commitment (1) to promptly report failures to the primary power-operated relief valves and safety valves and (2) to report all challenges to these valves in the time requirements specified by the Watts Bar TS, Section 5.9.4, is consistent with NUREG-0737 and was, therefore, acceptable.

By letter dated April 29, 2010, TVA stated that Amendment 57 to the Unit 1 TS removed Section 5.9.4 relating to monthly operating reports. The NRC staff approved this amendment by letter dated March 21, 2005. TVA further stated that the Unit 2 TS will also contain no such requirement and listed this item as "submitted," based on its March 4, 2009, submittal of Developmental Revision A of the WBN Unit 2 TS. The staff's safety evaluation for Amendment 57 states the following:

> The reporting requirements for the monthly operating report include challenges to the pressurizer power operated relief valves or pressurizer safety valves. The reporting of challenges to the pressurizer power operated relief valves or pressurizer safety valves was included in TSs based on the guidance in NUREG-0694, "[Three Mile Island] TMI-Related Requirements for New Operating Licensees." The industry proposed and the NRC accepted the elimination of the reporting requirements in TS for challenges to the pressurizer power operated relief valves or pressurizer safety valves in Revision 4 to TSTF-258, "Changes to Section 5.0, Administrative Controls." The staff's acceptance of TSTF-258 and subsequent approval of plant-specific adoptions of TSTF-258 is based on the fact that the information on challenges to relief and safety valves is not used in the evaluation of the monthly operating report data, and that the information needed by the NRC is adequately addressed by the reporting requirements in 10 CFR 50.73, "Licensee event reports."

The NRC staff concluded that the above evaluation may also be applied to WBN Unit 2. The staff concludes that TVA's statement that the Unit 2 TS will not contain the requirement for a monthly operating report is acceptable, and item II.K.3.3 is resolved.

Report on Outages of Emergency Core Cooling System (II.K.3.17)

In SSER 3, the NRC staff concluded in Section 13.5.3 that TVA's position that active participation in the nuclear power reliability data system and compliance with the requirements of 10 CFR 50.73 satisfy the current requirements of item II.K.3.17 of NUREG-0737. Therefore, the staff considers this item closed.

The NRC staff determined that its conclusion in SSER 3 applies to both Unit 1 and Unit 2, and this item is resolved.

Conclusion

In SSER 21, the NRC staff listed Section 13.5.3 as "Open (Inspection)." Based on the above evaluations, the staff concludes that no inspection is required for items II.K.3.3 and II.K.17, and Section 13.5.3 is resolved.

13.6 Physical Security

13.6.1 Introduction

The OL application for WBN Unit 2 describes TVA's physical protection program, which is intended to meet NRC regulations for protection against the design-basis threat (DBT) of radiological sabotage as stated in 10 CFR 73.1, "Purpose and Scope," and to provide a high assurance that activities involving special nuclear material are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety.

In its letter dated August 3, 2007, TVA stated that it believes that, from regulatory, safety, and plant operational perspectives, significant benefit would be gained from aligning the licensing and design bases of WBN Units 1 and 2 to the fullest extent practicable. TVA stated that it will complete WBN Unit 2 in compliance with applicable regulations promulgated before and after the issuance of the WBN Unit 1 OL. In addition, TVA will incorporate modifications made to WBN Unit 1, and those modifications currently captured in the WBN Unit 1 5-year plan, into the WBN Unit 2 licensing and design bases. By this approach, TVA believes that this alignment of the WBN Unit 1 and 2 licensing and design bases will ensure that there is operational fidelity between the units and, at the same time, demonstrate that WBN Unit 2 complies with applicable NRC regulatory requirements.

In its letter dated August 3, 2007, TVA also stated that it anticipated making no changes to the Site Security Plan or the Site Emergency Plan for purposes of WBN Unit 2 construction reactivation. If needed, TVA will submit changes to the Site Security Plan or the Site Emergency Plan to the NRC as required by applicable regulations. Before resuming construction activities on quality- or safety-related structures, systems, or components (SSCs), the QA program and procedures will be in place.

In Section 13.7 of FSAR Amendment 101, TVA states that "the physical security program for the plant site [includes] physical barriers and means of detecting unauthorized intrusions; provisions for monitoring access to vital equipment and access control; provisions for selection and training of personnel for security purposes; communication systems for security; provisions for maintenance and testing of security systems; arrangements for law enforcement assistance;

provisions for responding to security threats; and required organizational charts and drawings that depict the site layout." The physical protection system includes the capabilities to detect, assess, interdict, and neutralize threats of radiological sabotage.

The OL application, consisting of the WBN Unit 2 Physical Security Plan (PSP), Training and Qualification Plan (T&QP), and Safeguards Contingency Plan (SCP), is referenced in Section 13.6 of the FSAR to describe the physical protection program and physical protection systems. The PSP, T&QP, and SCP are located in the NRC's Secure Local Area Network as document No. ES10016221 (nonpublicly available Safeguards Information (SGI)).

13.6.2 Summary of Application

TVA provided design and program descriptions and information related to physical protection systems, features, and elements of physical security programs in the following sections of the combined license application and referenced technical report:

<u>FSAR Amendment 98</u>: FSAR, Chapter 13, "Conduct of Operations," Section 13.1.1.1 describes the design responsibilities of Westinghouse Electric Corporation, TVA, and Bechtel Corporation for WBN Unit 2.

<u>FSAR, Chapter 13.7.1, "Physical Security and Contingency Plan"</u>: The Security Plan consists of three parts (the PSP, T&QP, and SCP).

By letter dated March 4, 2009, TVA submitted a PSP, Revision 7, prepared February 2008, to the NRC as part of the OL application for the proposed WBN Unit 2. By letter dated August 28, 2009, TVA submitted Revision 8 to the PSP. By letter dated October 16, 2009, TVA submitted Revision 9 to its PSP. By letter dated March 31, 2010, TVA submitted Revision 10 to its PSP. By letter dated July 23, 2010, TVA submitted Revision 11 to its PSP.

13.6.3 Regulatory Basis

The applicable regulatory requirements for physical protection are as follows:

- In 10 CFR 50.34(b), the NRC states that each application for an OL shall include an FSAR, which contains information that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the SSCs and the facility as a whole. Regarding facility operation, the FSAR must include information on the applicant's organizational structure and qualifications, managerial and administrative controls to ensure safe operation, plans for preoperational testing and initial operations, plans for normal operations including maintenance, surveillance, and testing of SSCs, emergency plans, and the proposed TS.

- The provisions of 10 CFR 50.34(c)(2) require, in part, that applicants for an OL for a utilization facility that will be subject to the requirements of 10 CFR 73.55, "Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors against Radiological Sabotage," must include a PSP and T&QP in accordance with the criteria in Appendix B, "General Criteria for Security Personnel," to 10 CFR Part 73, "Physical Protection of Plants and Materials."

- The provisions of 10 CFR 50.34(d)(2) require, in part, that applicants for an OL for a utilization facility that will be subject to the requirements of 10 CFR 73.55 must include an SCP in accordance with the criteria in Section II of Appendix C, "Nuclear Power Plant Safeguards Contingency Plans," to 10 CFR Part 73.

- The provisions of 10 CFR 50.34(e) require that each applicant for an OL for a utilization facility that prepares a PSP, a T&QP, and an SCP shall protect the plans and other SGI against unauthorized disclosure in accordance with the requirements of 10 CFR 73.21, "Protection of Safeguards Information: Performance Requirements."

- Under the provisions of 10 CFR 73.55(a)(5), TVA, holding a current construction permit under the provisions of 10 CFR Part 50 for WBN Unit 2, shall meet the revised requirements in paragraphs (a) through (r) of this section as applicable to operating nuclear power reactor facilities.

- The provisions of 10 CFR 73.55(a)(6) state, in part, that applicants for an OL under the provisions of 10 CFR Part 50, or holders of a combined license under the provisions of 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," that do not reference a standard design certification or reference a standard design certification issued after May 26, 2009, shall meet the requirement of 10 CFR 73.55(i)(4)(iii).

- The provisions of 10 CFR Part 73 include performance-based and prescriptive regulatory requirements that, when adequately met and implemented, provide high assurance that activities involving special nuclear material are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety. An OL applicant must describe how it will meet the regulatory requirements of 10 CFR Part 73 that apply to nuclear power plants.

- The provisions of 10 CFR Part 11, "Criteria and Procedures for Determining Eligibility for Access to or Control over Special Nuclear Material," include the requirements for access to or control over special nuclear material.

An OL applicant is required to identify and describe design features, analytical techniques, and technical bases for its design and how it will meet provisions of physical protection system requirements in the NRC regulations, using applicable regulatory guides and the SRP. However, the NRC regulatory guides and the SRP are not regulatory requirements and are not a substitute for compliance with established regulations. Where alternative methods are chosen or differences exist, the OL applicant is required to describe how the proposed alternatives to guidance or acceptance criteria provide acceptable methods of compliance with the NRC regulations.

The NRC staff used SRP Section 13.6.1, Revision 1, dated June 15, 2010 (ADAMS Accession No. ML100350158), as guidance in conducting its review of physical security.

Regulatory guidance documents, technical reports, and accepted industry codes and standards that an applicant may apply to meet regulatory requirements include, but are not limited to, the following:

- RG 5.7, "Entry/Exit Control for Protected Areas, Vital Areas, and Material Access Areas," Revision 1, May 1980

- RG 5.12, "General Use of Locks in the Protection and Control of Facilities and Special Nuclear Materials," November 1973

- RG 5.44, "Perimeter Intrusion Alarm Systems," Revision 3, October 1997

- RG 5.59, "Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate or Low Strategic Significance," Revision 1, February 1983

- RG 5.62, "Reporting of Safeguards Events," Revision 1, November 1987

- RG 5.65, "Vital Area Access Controls, Protection of Physical Protection System Equipment and Key and Lock Controls," September 1986

- RG 5.66, "Access Authorization Programs for Nuclear Power Plants," Revision 1, July 2009

- RG 5.68, "Protection Against Malevolent Use of Vehicles at Nuclear Power Plants," August 1994

- RG 5.74, "Managing the Safety/Security Interface," March 2009

- RG 5.75, "Training and Qualification of Security Personnel at Nuclear Power Reactor Facilities," June 2009

- RG 5.77, "Insider Mitigation Program," March 2009

- NRC letter dated April 9, 2009, NRC Staff Review of Nuclear Energy Institute (NEI) 03-12, "Template for Security Plan, Training and Qualification, Safeguards Contingency Plan, [and Independent Spent Fuel Storage Installation Security Program]," Revision 6 (ADAMS Accession No. ML090920528)

- NUREG-0847, "Safety Evaluation Report Related to the Operation of WBN Nuclear Plant, Units 1 and 2," June 1982, and SSERs including SSER 1, September 1982; SSER 2, September 1982; SSER 3, January 1984; SSER 4, March 1985; SSER 10, October 1992; SSER 15, January 1995; and SSER 20, February 1996

- SECY-07-0096, "Staff Requirements—Possible Reactivation of Construction and Licensing Activities for the Watts Bar Nuclear Plant Unit 2," July 25, 2007

The following documents contain security-related information or SGI and are not publicly available:

- RG 5.69, "Guidance for the Application of Radiological Sabotage Design Basis Threat in the Design, Development, and Implementation of a Physical Security Protection Program that Meets 10 CFR 73.55 Requirements," June 2006

- RG 5.76, "Physical Protection Programs at Nuclear Power Reactors," July 2009

- Nuclear Energy Institute (NEI) 03-12, "Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Installation Security Program," Revision 6, March 2009

- NUREG/CR-6190, "NUREG/CR-6190 Material to Reflect Postulated Threat Requirements," March 27, 2003

13.6.4 Technical Evaluation

Section 13.6.4.1 of this SER documents the NRC staff's evaluation of the PSP. Section 13.6.4.2 documents the staff's evaluation of the T&QP. Section 13.6.4.3 documents the staff's evaluation of the SCP. Section 13.6.5 contains the NRC staff's overall conclusions regarding each of the plan submissions.

13.6.4.1 Physical Security Plan

Section 13.7.1 of the WBN Unit 2 FSAR states the following:

> TVA's plan for protection of Watts Bar Nuclear Plant is contained in separate controlled documents. These documents require submission as separate submittals to ensure compliance with 10 CFR 73.21 and paragraph 2.390(d) of 10 CFR Part 2.... These documents are identified as:
>
> (1) The Physical Security Plan/Contingency Plan as specified by 10 CFR Parts 50.34(c), 50.34(d), and 73.55(a).
> (2) The Security Personnel Training and Qualification Plan as specified by 10 CFR 73.55(b).

Section 13.7 of the FSAR references the WBN Unit 2 PSP, T&QP, and SCP in describing the licensing basis for establishing a physical protection program, design of a physical protection system, and security organization which will have as its objective to provide high assurance that activities involving special nuclear material are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety.

Security plans must describe how TVA will implement Commission requirements and those site-specific conditions that affect implementation as required by 10 CFR 73.55(c)(1)(i).

The provisions of 10 CFR 73.55(c) and (d) require TVA to establish, maintain, and implement a PSP to meet the requirements of 10 CFR 73.55 and 10 CFR Part 73, Appendix B and Appendix C, "Nuclear Power Plant Safeguards Contingency Plans." TVA must show the establishment and maintenance of a security organization, the use of security equipment and technology, the training and qualification of security personnel, the implementation of predetermined response plans and strategies, and the protection of digital computer and communication systems and networks. TVA must have a management system for development, implementation, revision, and oversight of security implementing procedures. The approval process for implementing security procedures will be documented.

The NRC staff reviewed TVA's description in PSP Section 1 of the implementation of the site-specific PSP in accordance with Commission regulations and the NUREG-0800 acceptance criteria. Because TVA's description in the PSP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description provided in the PSP meets the requirements of 10 CFR 73.55(c) and (d) and is, therefore, acceptable.

13.6.4.1.1 Introduction and Physical Facility Layout

The provisions of 10 CFR 50.34(c)(2) require, in part, that applicants for an OL for a utilization facility that will be subject to the requirements of 10 CFR 73.55 must include a PSP and T&QP in accordance with the criteria in Appendix B to 10 CFR Part 73.

The provisions of 10 CFR 50.34(d)(2) require, in part, that applicants for an OL for a utilization facility that will be subject to the requirements of 10 CFR 73.55 must include an SCP in accordance with the criteria in Section II of Appendix C to 10 CFR Part 73.

In 10 CFR 73.55(c)(2), the NRC establishes requirements to ensure protection of SGI against unauthorized disclosure in accordance with 10 CFR 73.21. TVA's submittal acknowledges that the PSP, the T&QP, and the SCP discuss specific features of the physical security system or response procedures and are SGI.

Section 1 of the PSP describes TVA's commitment to satisfying 10 CFR 50.34(c) and (d) and 10 CFR Part 73 by submitting a PSP and to controlling the PSP and appendices as SGI in accordance with 10 CFR 73.21.

The provisions of 10 CFR Part 73, Appendix C, Section II(B)(3)(b), require a description of the physical layout of the site.

Section 1.1 of the PSP describes the location, site layout, and facility configuration. The PSP describes the physical structures and their locations on the site, the PA, and the site in relation to nearby towns, roads, and other environmental features important to the coordination of response operations. The plant layout includes identification of main and alternate entry routes for law enforcement assistance forces and the location of control points for marshaling and coordinating response activities.

Section 2.1.1 of the WBN FSAR provides general plant descriptions that include details of the geographical area within a radius of 10 to 50 miles of the WBN Unit 2 site, a site area map, and general plant and site descriptions. Chapter 1 of the FSAR references a four-loop PWR nuclear steam supply system furnished by Westinghouse Electric Corporation for the principal design and operating characteristics for the design and construction of WBN Unit 2. Chapter 1 of the FSAR gives the name of TVA and its principal business locations.

The NRC staff has reviewed the facility physical layout provided in Section 1.1 of the PSP and in the FSAR. The staff determined that TVA included site-specific conditions that affect TVA's capability to satisfy the requirements of a comprehensive PSP. TVA has adequately described the physical structures and their locations on site and the site in relation to nearby towns, roads, and other environmental features important to the effective coordination of response operations. The staff concludes that TVA's security plans have met the requirements for content of a PSP as stated above. Therefore, the NRC staff finds that the facility layout described in the PSP and OL application FSAR is adequate.

13.6.4.1.2 Performance Objectives

The provisions of 10 CFR 73.55(b)(1) require, in part, that the licensee shall establish and maintain a physical protection program with an objective to provide high assurance that activities involving special nuclear material are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety. In part, 10 CFR 73.55(b)(2) establishes the requirement to protect a nuclear power reactor against the DBT of radiological sabotage as described in 10 CFR 73.1. The provisions of 10 CFR 73.55(b)(3)(i) and 10 CFR 73.55(b)(3)(ii) require TVA to establish a physical protection program designed to ensure that the capabilities to detect, assess, interdict, and neutralize threats up to and including the DBT of radiological sabotage, as stated in 10 CFR 73.1, are maintained at all times and provide defense in depth, supporting processes, and implementing procedures that ensure the effectiveness of the physical protection program.

Section 2 of the PSP outlines the requirements for the establishment and maintenance of an onsite physical protection system, security organization, and integrated response capability. As part of the objective, the security program design shall incorporate supporting processes such that no single event can disable the security response capability because of defense-in-depth principles, including diversity and redundancy. The physical protection systems and programs described in Section 2 are designed to protect against the DBT of radiological sabotage in accordance with the requirements of 10 CFR 73.55(a) through (r) or equivalent measures that meet the same high assurance objectives provided by paragraphs (a) through (r).

At WBN, TVA uses the corrective action program to track, trend, correct, and prevent recurrence of failures and deficiencies in the physical protection program.

The NRC staff reviewed TVA's description in PSP Section 2, in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the PSP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the PSP meets the requirements of 10 CFR 73.55(b) and is, therefore, acceptable.

13.6.4.1.3 Performance Evaluation Program

In 10 CFR 73.55(b)(4) through (b)(11), the NRC establishes requirements for the applicant to analyze and identify site-specific conditions and establish programs, plans, and procedures that address performance evaluations, access authorization, cyber security, insider mitigation, fitness for duty (FFD), corrective actions, and operating procedures. The regulation in 10 CFR 73.55(b)(6) prescribes specific requirements to establish, maintain, and implement a performance evaluation program in accordance with 10 CFR Part 73, Appendix B, Section VI, for implementation of the plant protective strategy.

Section 3.0 of the PSP states that drills and exercises, as discussed in the T&QP, will be used to assess the effectiveness of the contingency response plan and the effectiveness of TVA's response strategy. Other assessment methods include formal and informal exercises or drills, self-assessments, and internal and external audits and evaluations.

The performance evaluation processes and criteria for assessing the effectiveness of the security program, including adequate protection against radiological sabotage, will be established in facility procedures. The corrective action program will manage any deficiencies identified.

Section 3.0 of the PSP references Section 4.0 of the T&QP, which provides additional details related to the performance evaluation of security personnel in accordance with 10 CFR Part 73, Appendix B, Section VI. Section 4.0 of the T&QP includes the requirements to conduct security force tactical drills and force-on-force exercises to evaluate the effectiveness of security systems and the response performances of security personnel. In addition, Section 17 of the PSP describes additional details of TVA's processes for reviews, evaluations, and audits that will complement the performance evaluation program.

The NRC staff reviewed TVA's description in PSP Section 3 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the PSP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the PSP meets the requirements of 10 CFR 73.55(b)(4) through (b)(11) and is, therefore, acceptable.

13.6.4.1.4 Establishment of Security Organization

The provisions of 10 CFR 73.55(d) establish requirements to describe a security organization, including the management system for oversight of the physical protection program. The security organization must be designed, staffed, trained, qualified, requalified, and equipped to implement the physical protection program as required by 10 CFR 73.55(b) and Appendices B and C to 10 CFR Part 73.

Section 4.0 of the PSP describes how TVA meets the requirements of 10 CFR 73.55(d)(1).

Security Organization Management

Section 4.1 of the PSP describes the organization's management structure. The PSP establishes that the security organization is a critical component of the physical protection program and is responsible for the effective application of engineered systems, technologies, programs, equipment, procedures, and personnel necessary to detect, assess, interdict, and neutralize threats up to and including the DBT of radiological sabotage. The security organization may be proprietary or employ contract or other qualified personnel.

The PSP states that the organization will be staffed with appropriately trained and equipped personnel, in a command structure with administrative controls and procedures, to provide a comprehensive response. Section 4.1 of the PSP also describes the roles and responsibilities of the security organization. The PSP provides that at least one full-time, dedicated Nuclear Security Lieutenant, who has the authority for command and control of all security operations, is on site at all times.

The regulation in 10 CFR 73.55(d)(3) states the following:

> The licensee may not permit any individual to implement any part of the physical protection program unless the individual has been trained, equipped, and qualified to perform their assigned duties and responsibilities in accordance with appendix B to this part and the Training and Qualification Plan. Non-security

personnel may be assigned duties and responsibilities required to implement the physical protection program and shall:
(i) Be trained through established licensee training programs to ensure each individual is trained, qualified, and periodically re-qualified to perform assigned duties.
(ii) Be properly equipped to perform assigned duties.
(iii) Possess the knowledge, skills, and abilities, to include physical attributes such as sight and hearing, required to perform their assigned duties and responsibilities.

The WBN PSP, Revision 10, follows NEI 03-12, which is endorsed by the NRC. Appendix C to NEI 03-12 gives guidance as to which position, including lieutenant, fills the role of Response Team Leader. The template is prescriptive in terms of the management positions that it describes, and Appendix C does not specifically describe the role of sergeant. Appendix A to the PSP does discuss the role of sergeant as part of facility supervision.

In RAI 13.6-9, the NRC staff asked TVA to clarify the duty positions of sergeant and lieutenant. Section 4.1 of the PSP does not identify these positions. The staff requested clarification regarding the position filled by sergeants and lieutenants (e.g., Armed Responder, Armed Security Officer, or Response Team Leader). By letter dated April 4, 2010, TVA responded to the RAI. On the basis of its review, the NRC staff concluded that TVA's response to the RAI, and the WBN PSP, Revision 10, dated March 31, 2010, did not meet the requirements of 10 CFR 73.55(d)(3). The NRC and TVA staffs discussed the issue during a secure telephone call on June 29, 2010, and TVA stated that it would revise the WBN PSP to clarify the duties, responsibilities, and training requirements of the lieutenant and sergeant. The NRC staff subsequently reviewed the WBN PSP, Revision 11, dated July 23, 2010, and concluded that, with the revision described, it meets the requirements of 10 CFR 73.55(d)(3).

The security force implementing security functions as described in this section of the plan will be either a proprietary force or a contractor force or will consist of other qualified personnel.

The NRC staff has reviewed TVA's description in PSP Sections 4 and 4.1 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the PSP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff concludes that the description meets the requirements of 10 CFR 73.55(d) and is, therefore, acceptable.

13.6.4.1.5 Qualification for Employment in Security

The requirements of 10 CFR 73.55(d)(3) state, in part, that the licensee may not permit any individual to implement any part of the physical protection program unless the individual has been trained, equipped, and qualified to perform assigned duties and responsibilities in accordance with Appendix B to 10 CFR Part 73 and the licensee's T&QP.

Section 5 of the PSP states that the T&QP delineates the employment qualifications for members of the security force.

The NRC staff has reviewed TVA's description in PSP Section 5 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the PSP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the PSP meets the requirements of 10 CFR 73.55(d)(3) and is, therefore, acceptable.

13.6.4.1.6 Training of Facility Personnel

Consistent with requirements of 10 CFR 73.55(d)(3),10 CFR 73.56 ("Personnel Access Authorization Requirements for Nuclear Power Plants"), and Section VI.C.1 of Appendix B to 10 CFR Part 73, all personnel who are authorized unescorted access to the licensee's PA must receive training, in part to ensure that they understand their role in security and their responsibilities in the event of a security incident. Individuals assigned to perform security-related duties or responsibilities, such as, but not limited to, material searches and vehicle escort must be trained and qualified in accordance with the T&QP to perform these duties and responsibilities and to ensure that each individual has the minimum knowledge, skills, and abilities required for effective performance of assigned duties and responsibilities.

Section 6 of the PSP describes the training provided for all personnel who have been granted unescorted access to TVA's PA.

The NRC staff has reviewed TVA's description in PSP Section 6 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the PSP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the PSP meets the requirements of 10 CFR 73.56 and 10 CFR Part 73, Appendix B, and is, therefore, acceptable.

13.6.4.1.7 Security Personnel Training

The provisions of 10 CFR 73.55(d) require that all security personnel are trained and qualified in accordance with 10 CFR Part 73, Appendix B, Section VI, before performing their duties.

Section 7 of the PSP states that all security personnel are trained and qualified and perform tasks at levels specific for their assignments in accordance with the licensee's T&QP.

The NRC staff has reviewed TVA's description in PSP Section 7 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the PSP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the PSP meets the requirements of 10 CFR 73.55(d) and is, therefore, acceptable. The NRC staff's review of the licensee T&QP is located in Section 13.6.4.2 of this SSER.

13.6.4.1.8 Local Law Enforcement Liaison

The regulation at 10 CFR 73.55(k)(9) states, "To the extent practicable, licensees shall document and maintain current agreements with applicable law enforcement agencies to include estimated response times and capabilities." In addition, 10 CFR 73.55(m)(2) requires, in part, that an evaluation of the effectiveness of the physical protection system include an audit of response commitments by local, State, and Federal law enforcement authorities.

Section 8 of the PSP provides a detailed discussion of TVA's ongoing relationship with local law enforcement agencies (LLEAs). The plans addressing response, communication methodologies and protocols, command and control structures, and marshaling locations are located in the operations procedures, emergency plan procedures, and the site-specific law enforcement response plan. The law enforcement response plan is reviewed biennially, concurrent with the PSP effectiveness review.

The NRC staff has reviewed TVA's description in PSP Section 8 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the PSP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the PSP meets the requirements of 10 CFR 73.55(k)(9) and 10 CFR 73.55(m)(2) and is, therefore, acceptable.

13.6.4.1.9 Security Personnel Equipment

The requirements of 10 CFR 73.55(d)(3) state, in part, that the licensee may not permit any individual to implement any part of the physical protection program unless the individual has been trained, equipped, and qualified in accordance with Appendix B to 10 CFR Part 73 and the T&QP. In 10 CFR Part 73, Appendix B, Section VI.G.2(a), states, in part, that the applicant must ensure that each individual is equipped or has ready access to all personal equipment or devices required for the effective implementation of the NRC-approved security plans, the applicant's protective strategy, and implementing procedures. Sections VI.G.2(b) and (c) of Appendix B to 10 CFR Part 73 delineate the minimum equipment requirements for security personnel and armed response personnel.

Section 9 of the PSP describes the equipment, including armament, ammunition, and communications equipment that is provided to security personnel to ensure that they are capable of performing the function stated in the Commission-approved security plans, applicant's protective strategy, and implementing procedures.

The NRC staff has reviewed TVA's description in PSP Section 9 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the PSP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the PSP meets the requirements of 10 CFR 73.55(d)(3) and Appendix B, Section VI.G.2, and is, therefore, acceptable.

13.6.4.1.10 Work-Hour Controls

The provisions of 10 CFR Part 26, "Fitness for Duty Programs," Subpart I, "Managing Fatigue," establish the requirements for managing fatigue. The provisions of 10 CFR 26.205, "Work Hours," establish the requirements. The provisions of 10 CFR 26.205(a) require that any individual who performs duties identified in 10 CFR 26.4(a)(1) through (a)(5) shall be subject to the requirements of this section [10 CFR 26.205].

Section 10 of the PSP states that the site will implement work-hour controls consistent with 10 CFR Part 26, Subpart I, and that site procedures shall describe performance objectives and implementing procedures.

The NRC staff's review of the fitness-for-duty program is found in Section 13.6.4.1.14 of this SSER.

13.6.4.1.11 Physical Barriers

In part, 10 CFR 73.55(e) states the following:

> Each licensee shall identify and analyze site-specific conditions to determine the specific use, type, function, and placement of physical barriers needed to satisfy the physical protection program design requirements of §73.55(b).
> (1) The licensee shall:
> (i) Design, construct, install and maintain physical barriers as necessary to control access into facility areas for which access must be controlled or denied to satisfy the physical protection program design requirements of paragraph (b) of this section.

The provisions of 10 CFR 73.55(b)(3)(ii) require that the physical protection program "Provide defense-in-depth through the integrations of systems, technologies, programs, equipment, supporting processes, and implementing procedures as needed to ensure the effectiveness of the physical protection program."

Section 11 of the PSP provides a general description of how TVA has implemented its program for physical barriers and states that this implementation is in accordance with the performance objectives and requirements of 10 CFR 73.55(b).

<u>Owner-Controlled Area Barriers</u>

Section 11.1 of the PSP states that WBN uses owner-controlled area (OCA) barriers for nuisance control and as an enhancement of the protective strategy.

<u>Vehicle Barriers</u>

PSP Sections 11.2.1 and 11.2.2 describe the requirement to establish and maintain vehicle control measures, as necessary, to protect against the DBT of radiological sabotage, consistent with the physical protection program design requirements of 10 CFR 73.55(b)(3)(ii) and 10 CFR 73.55(e)(10)(i), and in accordance with site-specific analysis. The PSP identifies measures taken to provide high assurance that such an event can be defended against. The PSP also states that facility procedures include the inspection, monitoring, and maintenance of the vehicle barrier system.

<u>Waterborne Threat Measures</u>

The regulation at 10 CFR 73.55(e)(10)(ii) requires the following:

> Licensees shall: (A) Identify areas from which a waterborne vehicle must be restricted, and where possible, in coordination with local, State, and Federal agencies having jurisdiction over waterway approaches, deploy buoys, markers, or other equipment. (B) In accordance with the site-specific analysis, provide periodic surveillance and observation of waterway approaches and adjacent areas.

Section 11.2.3 of the PSP provides a general description of how TVA implemented its protection measures for the intake structure and adjacent areas. The implementation of these protection measures is in accordance with the performance objectives and requirements of 10 CFR 73.55(e)(10)(ii).

Protected Area Barriers

The regulation at 10 CFR 73.55(e)(8)(i) requires the following:

> The protected area perimeter must be protected by physical barriers that are designed and constructed to: (A) Limit access into the protected area to only those personnel, vehicles, and materials required to perform official duties; (B) Channel personnel, vehicles, and materials to designated access control portals; and (C) Be separated from any other barrier designated as a vital area physical barrier, unless otherwise identified in the [PSP].

PSP Section 11.3 describes the PA barrier. These descriptions are consistent with the definitions of physical barrier and PA contained in 10 CFR 73.2, "Definitions," and the requirements of 10 CFR 73.55(e)(8).

Section 11.3 of the PSP describes the extent to which the PA barrier at the perimeter is separated from a vital area/island barrier. The security plan identifies where the PA barrier is not separated from a vital area barrier, consistent with 10 CFR 73.55(e)(8)(i)(c).

Section 11.3 of the PSP describes isolation zones. As required in 10 CFR 73.55(e)(7), the isolation zone is maintained in outdoor areas adjacent to the PA perimeter barrier and is designed to ensure the ability to observe and assess activities on either side of the PA perimeter.

The regulation at 10 CFR 73.55(e)(9(i) states that "Vital equipment must be located only within a vital area, which must be located within a protected area such that access to vital equipment requires passage through at least two physical barriers, except as otherwise approved by the Commission and identified in the security plans." In RAI 13.6-7, the NRC staff requested that TVA provide additional details explaining how the two-barrier concept referred to in Section 6.3, page 17, paragraph 5, of the PSP meets the vital area barrier requirements. By letter dated April 5, 2010, TVA responded.

The NRC staff approved exceptions to the "two-barrier requirement" in SSER 15, issued June 1995. Specifically, Section 13.6.2.1 states the following:

> The appendix (the section of the SER protected under 10 CFR 73.21) identifies those areas determined to be vital for protection purposes. Vital equipment is located within vital areas which are located within the protected area and which require passage through at least two barriers as defined in 10 CFR 73.2, definitions for "Physical Barriers," with certain exceptions, to gain access to vital equipment. The NRC staff has reviewed those exceptions and has determined that the barriers are sufficiently substantial to meet the intent of the two barrier requirement.

On the basis of its previous review, as documented in SSER 15, and Commission-approved exceptions, and its review of TVA's response to RAI 13.06-7, the NRC staff concludes that the WBN PSP, Revision 10, dated March 31, 2010, meets the intent of the two-barrier requirement of 10 CFR 73.55(e)(9(i) and is, therefore, acceptable.

Vital Area Barriers

The regulation at 10 CFR 73.55(e)(9) states, "Vital equipment must be located only within vital areas, which must be located within a protected area so that access to vital equipment requires passage through at least two physical barriers, except as otherwise approved by the Commission and identified in the security plans." In addition, 10 CFR 73.55(e)(5) requires that certain vital areas must be bullet resisting.

Section 11.4 of the PSP states that vital areas are restricted access areas surrounded by physical barriers with the capability to restrict access to only authorized individuals. All vital areas are constructed in accordance with established regulatory requirements. Section 11.4 also states that the reactor control room, central alarm station (CAS), and the location within which the last access control function for access to the PA is performed must be bullet resisting. Based on its review, the NRC staff concludes that the requirements of 10 CFR 73.55(e)(9) and (e)(5) are met, and, therefore, Section 11.4 of the PSP is acceptable.

Target Set Equipment

The regulation at 10 CFR 73.55(f) states the following:

> (1) The licensee shall document and maintain the process used to develop and identify target sets, to include the site-specific analyses and methodologies used to determine and group the target set equipment or elements.
> (2) The licensee shall consider cyber attacks in the development and identification of target sets.
> (3) Target set equipment or elements that are not contained within a protected or vital area must be identified and documented consistent with the requirements in §73.55(f)(1) and be accounted for in the licensee's protective strategy.
> (4) The licensee shall implement a process for the oversight of target set equipment and systems to ensure that changes to the configuration of the identified equipment and systems are considered in the licensee's protective strategy. Where appropriate, changes must be made to documented target sets.

The regulation at 10 CFR 73.55(b)(4) states, "The licensee shall analyze and identify site-specific conditions, including target sets, that may affect the specific measures needed to implement the requirements of this section and shall account for these conditions in the design of the physical protection program."

The regulation at 10 CFR 73.55(e) states, in part, the following:

> (1) The licensee shall: (i) Design, construct, install and maintain physical barriers as necessary to control access into facility areas for which access must be controlled or denied to satisfy the physical protection program design requirements of paragraph (b) of this section. (ii) Describe in the security plan,

physical barriers, barrier systems, and their functions within the physical protection program.
(2) The licensee shall retain, in accordance with § 73.70, all analyses and descriptions of the physical barriers and barrier systems used to satisfy the requirements of this section, and shall protect these records in accordance with the requirements of § 73.21.
(3) Physical barriers must: (i) Be designed and constructed to: (A) Protect against the design basis threat of radiological sabotage; (B) Account for site-specific conditions; and (C) Perform their required function in support of the licensee physical protection program.

The regulation at 10 CFR 73.55(e)(9)(i) states, "Vital equipment must be located only within vital areas, which must be located within a protected area so that access to vital equipment requires passage through at least two physical barriers, except as otherwise approved by the Commission and identified in the security plans."

In RAI 13.6-6, the NRC staff requested that TVA describe the process used to evaluate vital areas and equipment to support Unit 2 target sets, including the analysis data for the vehicle barrier system standoff distance against the vehicle bomb threat.

In its response dated April 5, 2010, TVA stated that it will utilize the expert panel methodology, as described in Appendix C to the NEI 03-12, Revision 6, template, when evaluating vital areas and equipment in support of Unit 2 target sets. Target sets for WBN Unit 2 will be developed by August 20, 2010. Section 11.2 of Revision 11 to the WBN PSP describes the vehicle barrier system requirements.

On the basis of its review of (1) the information provided by TVA in its letter dated April 5, 2010, (2) WBN PSP, Revision 11, dated July 23, 2010, and (3) TVA's letter, "Response to Request for Additional Information Regarding Target Set Development," dated November 18, 2010, the NRC staff concludes that the PSP meets the requirements of 10 CFR 73.55(f) for WBN Unit 2.

Section 11.5 of the PSP states that target set equipment or elements that are not contained within a protected or vital area are identified and accounted for in the site protective strategy, as required by 10 CFR 73.55(f)(3).

The NRC staff reviewed TVA's description in Sections 11.5 and 14.5 of the PSP, and Section 7 of the SCP, of the implementation of the site-specific physical protection program, in accordance with Commission regulations and NUREG-0800 acceptance criteria. Since TVA's description in Sections 11.5 and 14.5 of the PSP, and in Section 7 of the SCP, are consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the PSP and SCP meet the requirements of 10 CFR 73.55(f)(1), (3) and (4), and are, therefore, acceptable. The target sets, target set analysis, and site protective strategy are in the facility implementing procedures, which are not subject to NRC staff review as part of this SSER. These are subject to future NRC inspection in accordance with 10 CFR 73.55(c)(7)(iv) and 10 CFR Part 73, Appendix C, Section II.B.5(iii).

Delay Barriers

The regulation at 10 CFR 73.55(e)(3)(C)(ii) requires that physical barriers must "provide deterrence, delay, or support access control" to perform the required function in support of the licensee physical protection program. The PSP describes the use of delay barriers at WBN Unit 2.

Section 11.6 of the PSP includes a description of the use of delay barriers to meet the requirement of 10 CFR 73.55(e).

The NRC staff has reviewed TVA's description in PSP Sections 11, 11.1, 11.2, 11.2.1, 11.2.2, 11.2.3, and Sections 11.3 through 11.6, for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the PSP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the PSP meets the requirements of 10 CFR 73.55(e) and is, therefore, acceptable.

13.6.4.1.12 Security Posts and Structures

The regulation at 10 CFR 73.55(e)(5) states that "the reactor control room, the central alarm station, and the location within which the last access control function for access to the protected area is performed, must be bullet-resisting."

Section 12 of the PSP states that security posts and structures are qualified to a level commensurate with their application within the site protective strategy and that these positions are constructed of bullet-resisting materials.

The NRC staff has reviewed TVA's description in PSP Section 12 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the PSP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the PSP meets the requirements of 10 CFR 73.55(e)(5) and is, therefore, acceptable.

13.6.4.1.13 Access Control Devices

The regulation at 10 CFR 73.55(g)(1) states that "consistent with the function of each barrier or barrier system, the licensee shall control personnel, vehicle, and material access, as applicable, at each access control point in accordance with the physical protection program design requirements of §73.55(b)."

The regulation at 10 CFR 73.55(g)(6)(i) states, in part, "The licensee shall control all keys, locks, combinations, passwords and related access control devices used to control access to protected areas, vital areas and security systems to reduce the probability of compromise."

Types of Security-Related Access Control Devices

Section 13.1 of the PSP states, in part, that TVA uses security-related access control devices to control access to protected and vital areas and security systems.

Control and Accountability

Section 13.2.1 of the PSP describes the control of security-related locks. Section 13.2.2 of the PSP describes the controls associated with the changes to and replacements of access control devices, the accountability and inventory control process, and the circumstances that require changes in security-related locks. TVA uses facility procedures that reduce the probability of compromise to produce, control, and recover keys, locks, and combinations for all areas and equipment. Issue of access control devices is limited to individuals who have unescorted access authorization and require access to perform official duties and responsibilities. Keys and locks are accounted for through a key inventory control process, as described in facility procedures.

The NRC staff has reviewed TVA's description in PSP Sections 13, 13.1, 13.2, 13.2.1, and 13.2.2 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the PSP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the PSP meets the requirements of 10 CFR 73.55(g)(1) and (6) and is, therefore, acceptable.

13.6.4.1.14 Access Requirements

Access Authorization and Fitness for Duty

The regulation at 10 CFR 73.55(b)(7) states that "The licensee shall establish, maintain, and implement an access authorization program in accordance with §73.56 and shall describe the program in the [PSP]." The provisions of 10 CFR Part 26 require the licensee to establish and maintain an FFD program.

Section 14.1 of the PSP states that the access authorization program implements regulatory requirements utilizing the provisions in RG 5.66, Revision 1. RG 5.66 is an acceptable method for meeting the requirements of 10 CFR 73.55(b)(7).

The NRC staff has reviewed TVA's description in PSP Section 14.1 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the PSP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the PSP meets the requirements of 10 CFR 73.55(b)(7), 10 CFR 73.56, and 10 CFR Part 26 and is, therefore, acceptable.

Insider Mitigation Program

The regulation at 10 CFR 73.55(b)(9) states the following:

> The licensee shall establish, maintain, and implement an insider mitigation program and shall describe the program in the [PSP].
> (i) The insider mitigation program must monitor the initial and continuing trustworthiness and reliability of individuals granted or retaining unescorted access authorization to a protected or vital area, and implement defense-in-depth methodologies to minimize the potential for an insider to adversely affect, either directly or indirectly, the licensee's capability to prevent significant core damage and spent fuel sabotage.
> (ii) The insider mitigation program must contain elements from: [the access authorization program, the fitness for duty program, the cyber security program, and the physical protection program.]

Section 14.2 of the PSP describes how TVA will establish, maintain, and implement an insider mitigation program utilizing the guidance in RG 5.77. The insider mitigation program requires elements from the access authorization program described in 10 CFR 73.56; the FFD program described in 10 CFR Part 26; the cyber security program described in 10 CFR 73.54, "Protection of Digital Computer and Communication Systems and Networks"; and the physical security program described in 10 CFR 73.55. Section 14.2 describes the integration of the programs mentioned above to form a cohesive and effective insider mitigation program. In addition, TVA addresses the observations established for the detection of tampering.

The NRC staff has reviewed TVA's description in PSP Section 14.2 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the PSP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the PSP meets the requirements of 10 CFR 73.55(b)(9) and is, therefore, acceptable.

<u>Picture Badge Systems</u>

The regulation at 10 CFR 73.55(g)(6)(ii) states that "The licensee shall implement a numbered photo identification badge system for all individuals authorized unescorted access to the protected area and vital areas." In addition, the regulation requires that identification badges may be removed from the PA under limited conditions and only by authorized personnel and that records of all badges shall be retained and shall include the name and areas to which persons are granted unescorted access.

The regulation at 10 CFR 73.55(g)(7)(ii) states the following:

> Individuals not employed by the licensee but who require frequent or extended unescorted access to the protected area and/or vital areas to perform duties and responsibilities required by the licensee at irregular or intermittent intervals, shall satisfy the access authorization requirements of §73.56 and part 26 of this chapter, and shall be issued a non-employee photo identification badge that is easily distinguished from other identification badges before being allowed unescorted access to the protected and vital areas. Non-employee photo identification badges must visually reflect that the individual is a non-employee and that no escort is required.

Section 14.3 of the PSP describes the site picture badge system. Identification badges will be displayed while individuals are inside the PA or vital areas. When not in use, badges may be removed from the PA by authorized holders, provided that a process exists to deactivate the badge upon exit and positively confirm the individual's true identity and authorization for unescorted access prior to entry into the PA. Records are maintained to include the name and areas to which unescorted access is granted of all individuals to whom photo identification badges have been issued.

The NRC staff has reviewed TVA's description in PSP Section 14.3 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the PSP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the PSP meets the requirements of 10 CFR 73.55(g)(6) and (7) and is, therefore, acceptable.

Searches

The regulation at 10 CFR 73.55(h) states the following:

> (1) The objective of the search program is to detect, deter, and prevent the introduction of firearms, explosives, incendiary devices, or other items which could be used to commit radiological sabotage. To accomplish this licensee's shall search individuals, vehicles, and materials consistent with the physical protection program design requirements in paragraph (b) of this section, and the function to be performed at each access control point or portal before granting access.

Section 14.4 of the PSP describes an overview of the search process for vehicles, personnel, and materials. The search process is conducted using security personnel, specifically trained nonsecurity personnel, and technology. Detailed discussions of actions to be taken if unauthorized materials are discovered appear in the implementing procedures.

Vehicle Barrier System Access Control Point

The regulations at 10 CFR 73.55(h)(2)(ii) through (v) require licensees to search vehicles at the OCA. The provisions of 10 CFR 73.55(h)(3) require searches of personnel, vehicles, and materials before they enter the PA.

Section 14.4.1 of the PSP describes the process for the search of personnel, vehicles, and materials at predetermined locations before they are granted access to designated facility areas identified by TVA, as needed to satisfy the physical protection program. TVA states that it has developed specific implementing procedures to address vehicle and materials searches at these locations.

Protected Area Packages and Materials Search

Section 14.4.2 of the PSP describes the process for conducting searches of packages and materials for firearms, explosives, incendiary devices, or other items that could be used to commit radiological sabotage using equipment capable of detecting these items, or through visual and physical searches, or both, to ensure that all items are clearly identified before they can enter the WBN PA. Detailed requirements for conducting these searches appear in applicant implementing procedures and include the search and control of bulk materials and products. TVA's implementing procedures also discuss the control of packages and materials previously searched and tamper sealed by personnel trained in accordance with the T&QP.

Protected Area Vehicle Search

Section 14.4.3 of the PSP describes the process for the search of vehicles for firearms, explosives, incendiary devices, or other items that could be used to commit radiological sabotage using equipment capable of detecting these items, or through visual and physical searches, or both, to ensure that all items are clearly identified at the PA. TVA's implementing procedures contain detailed requirements for conducting these searches. TVA's implementing procedures also address the search methodologies for vehicles that must enter the PA under emergency conditions.

Protected Area Personnel Searches

Section 14.4.4 of the PSP describes the process for searches of all personnel requesting access into PAs. The PSP describes the search procedures for firearms, explosives, incendiary devices, or other items that could be used to commit radiological sabotage using equipment capable of detecting these items, or through visual and physical searches, or both, to ensure that all items are clearly identified before they are allowed access into the PA. All persons except official Federal, State, and LLEA personnel on official duty are subject to these searches upon entry to the PA. The implementing procedures contain detailed discussions of observation and control measures.

Protected Area Access Controls

Section 14.4.5 of the PSP describes the process for controlling access at all points where personnel or vehicles could gain access into TVA's PA. The plan states that all points of personnel access are through lockable portals. Multiple security personnel normally monitor the entry process. Personnel are normally allowed access through means that verify identity and authorization following the search process. Vehicles are controlled through positive control methods as described in facility procedures.

Escort and Visitor Requirements

The regulation at 10 CFR 73.55(g)(7) states, in part, that "the licensee may permit escorted access to protected and vital areas to individuals who have not been granted unescorted access in accordance with the requirements of §73.56 and part 26 of this chapter." Escort requirements for visitors are addressed in 10 CFR 73.55(g)(8).

Section 14.4.6 of the PSP describes the process for control of visitors. The PSP affirms that procedures address the identification, processing, and escorting of visitors and the maintenance of a visitor control register. Training requirements for escorting visitors include responsibilities, communications, and escort ratios. All escorts are trained to perform escort duties in accordance with site requirements. All visitors wear a badge that clearly indicates that an escort is required.

The NRC staff has reviewed TVA's description in PSP Sections 14.4, and 14.4.1 through 14.4.6, for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the PSP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the PSP meets the requirements of 10 CFR 73.55(h)(2), (h)(3), (g)(7), and (g)(8) and is, therefore, acceptable.

Vital Area Access Controls

The regulation at 10 CFR 73.55(g)(4) states that "(i) Licensees shall control access into vital areas consistent with established access authorization lists. (ii) In response to a site-specific credible threat or other credible information, implement a two-person (line-of-sight) rule for all personnel in vital areas so that no one individual is permitted access to a vital area."

The regulation at 10 CFR 73.56(j) states the following:

> Licensees or applicants shall establish, implement, and maintain a list of individuals who are authorized to have unescorted access to specific nuclear power plant vital areas during non-emergency conditions. The list must include only those individuals who have a continued need for access to those specific vital areas in order to perform their duties and responsibilities. The list must be approved by a cognizant licensee manager or supervisor who is responsible for directing the work activities of the individual who is granted unescorted access to each vital area, and updated and re-approved no less frequently than every 31 days.

Section 14.5 of the PSP describes vital areas and states that TVA maintains vital areas locked and protected by an active intrusion alarm system. An access authorization system is established to limit unescorted access that is controlled by an access authorization list, which is reassessed and reapproved at least once every 31 days. Facility procedures describe additional access control measures.

The NRC staff has reviewed TVA's description in PSP Section 14.5 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the PSP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the PSP meets the requirements of 10 CFR 73.55(g)(4) and is, therefore, acceptable.

13.6.4.1.15 Surveillance Observation and Monitoring

The regulation at 10 CFR 73.55(i)(1) states that "the licensee shall establish and maintain intrusion detection and assessment systems that satisfy the design requirements of §73.55(b) and provide, at all times, the capability to detect and assess unauthorized persons and facilitate the effective implementation of the licensee's protective strategy."

Illumination

The regulation at 10 CFR 73.55(i)(6) states, in part, that "all areas of the facility are provided with illumination necessary to satisfy the design requirements of §73.55(b) and implement the protective strategy." Specific requirements include providing a minimum illumination level of 0.2 foot-candles, measured horizontally at ground level, in the isolation zones and appropriate exterior areas within the PA. Alternatively, the licensee may augment the facility illumination system by means of low-light technology to meet the requirements of this regulation or otherwise implement the protective strategy. The licensee shall describe in the security plans how it will meet the lighting requirements of this regulation and the type(s) and application of low-light technology, if used.

Section 15.1 of the PSP describes that all isolation zones and appropriate exterior areas within the PA have lighting capabilities that provide illumination sufficient for the initiation of an adequate response to an attempted intrusion of the isolation zone, a PA, or a vital area. The section discusses the implementation of technology using fixed and nonfixed low-light-level cameras or alternative technological means. TVA has addressed the potential for loss of lighting and the compensatory actions that would be taken if lighting were lost.

Surveillance Systems

The regulation at 10 CFR 73.55(i)(5) states, in part, that "the physical protection program must include surveillance, observation, and monitoring as needed to satisfy the design requirements of §73.55(b)...."

Section 15.2 of the PSP states that surveillance is accomplished by human observation and technology. Surveillance systems include a variety of cameras, video display, and annunciation systems designed to assist the security organization in observing, detecting, and assessing alarms or unauthorized activities. Certain systems provide real-time and recorded playback of recorded video images. The facility implementing procedures describe the specifics of the surveillance systems.

Intrusion Detection Equipment

Section 15.3 of the PSP describes the perimeter intrusion detection system and the PA and vital area intrusion detection systems. These systems are capable of detecting attempted penetration of the PA perimeter barrier, are monitored with assessment equipment designed to satisfy the requirements of 10 CFR 73.55(i), and provide real-time and playback/recorded video images of the detected activities before and after each alarm annunciation. The PSP describes how TVA will meet regulatory requirements for redundancy, tamper indication, and uninterruptible power supply.

Central Alarm Station and Secondary Alarm Station Operation

The regulation at 10 CFR 73.55(i)(4)(i) states the following:

> Both alarm stations...must be designed and equipped to ensure that a single act, in accordance with the [DBT] of radiological sabotage defined in §73.1(a)(1), cannot disable both alarm stations. The licensee shall ensure the survivability of at least one alarm station to maintain the ability to perform the following functions: (A) Detect and assess alarms; (B) Initiate and coordinate an adequate response to an alarm; (C) Summon offsite assistance; and (D) Provide command and control.

Section 15.4 of the PSP describes the functional operations of the CAS and the secondary alarm station (SAS). The PSP states that the alarm stations are equipped such that no single act will disable both alarm stations. TVA's PSP also states that each alarm station is properly manned and that no activities are permitted that would interfere with the operator's ability to execute assigned duties and responsibilities.

Owner-Controlled Area Surveillance

The regulation at 10 CFR 73.55(e)(6) states that "the licensee shall establish and maintain physical barriers in the owner controlled area as needed to satisfy the physical protection program design requirements of §73.55(b)."

The regulation at 10 CFR 73.55(i)(5)(ii) states, in part, that "the licensee shall provide continuous surveillance, observation, and monitoring of the owner controlled area...[these]

responsibilities may be performed by security personnel during continuous patrols, through the use of video technology, or by a combination of both."

Section 15.5.1 of the PSP describes the processes used to meet this requirement. The PSP discusses the process to be used and states that details regarding the implementation of OCA surveillance techniques appear in facility procedures. The PSP discusses the implementation of manned and video options for patrolling and surveillance of the OCA.

Protected and Vital Area Patrols

The regulations at 10 CFR 73.55(i)(5)(iii) through (viii) require, in part, that armed patrols check unattended openings that intersect a security boundary, such as an underground pathway; check external areas of the PA and vital area portals; periodically inspect vital areas; conduct random patrols of accessible target set equipment; and be trained to recognize obvious tampering and, if tampering is detected, initiate an appropriate response in accordance with established plans and procedures.

Section 15.5.2 of the PSP describes the process employed by TVA to meet the above requirements. The PSP describes the areas of the facility that will be patrolled and observed, as well as the frequency of these patrols and observations. TVA has addressed the observations for the detection of tampering in Section 14.2 of the PSP and facility procedures.

The NRC staff has reviewed TVA's description in PSP Sections 15, 15.1 through 15.4, 15.5.1, and 15.5.2, for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the PSP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the PSP meets the requirements of 10 CFR 73.55(b) and (i) and is, therefore, acceptable.

13.6.4.1.16 Communications

The regulations at 10 CFR 73.55(j)(1) through (6) describe the requirements for establishment and maintenance of continuous communication capability with both onsite and offsite resources to ensure effective command and control during both normal and emergency situations. Alarm stations must be capable of calling for assistance, on-duty security force personnel must be capable of maintaining continuous communication with each alarm station and vehicle escorts, and personnel escorts must maintain timely communication with security personnel. Continuous communication capabilities must terminate in both alarm stations, between LLEAs and the control room. Nonportable communications must remain operable from independent power sources, and the licensee must identify areas where communications could be interrupted or not maintained.

Notifications (Security Contingency Event Notifications)

Section 16.1 of the PSP describes TVA's process for ensuring that continuous communications are established and maintained between the onsite security force staff and the offsite support agencies.

System Descriptions

Section 16.2 of the PSP describes the establishment and maintenance of the communications system. The facility procedures include detailed descriptions of security systems. WBN has access to both hard-wired and alternate communications systems. Site security personnel are assigned communications devices with which to maintain continuous communications with the CAS and SAS. All personnel and vehicles are assigned communications resources with which to maintain continuous communications. Continuous communications protocols are available between the CAS, SAS, and control room.

The NRC staff has reviewed TVA's description in PSP Sections 16, 16.1, and 16.2 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the PSP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the PSP meets the requirements of 10 CFR 73.55(j)(1) through (6) and is, therefore, acceptable.

13.6.4.1.17 Review, Evaluation, and Audit of the Physical Security Program

The regulation at 10 CFR 73.55(m) requires, in part, that each element of the physical protection program will be reviewed at least every 24 months. An initial review is required within 12 months after implementation of the original plan, or a change in personnel, procedures, equipment, or facilities that could have a potentially adverse affect on security, or as necessary based on site-specific analysis assessments or other performance indicators. Reviews must be conducted by individuals independent of the security program and must include the plans, implementing procedures, and local law enforcement commitments. Results of reviews shall be presented to senior management above the level of the Security Manager and findings must be entered in the site corrective action program.

Section 17 of the PSP states that the physical security program is reviewed 12 months following initial implementation, and at least every 24 months by individuals independent of both the security program management and the personnel who have a direct responsibility for implementation of the security program. The physical security program review includes, but is not limited to, an audit of the effectiveness of the physical security program; cyber security plans; implementing procedures, safety/security interface activities; the testing, maintenance, and calibration program; and response commitments by local, State, and Federal law enforcement authorities.

A review shall be conducted as necessary based on site-specific analyses, assessments, or other performance indicators and as soon as reasonably practical but no longer than 12 months after changes occur in personnel, procedures, equipment, or facilities that could adversely affect safety or security.

The results and recommendations of the physical security program review, management's finding as to the current effectiveness of the physical security program, and any actions taken as a result of recommendations from prior program reviews are documented in a report to plant management and to appropriate corporate management at least one level higher than that having responsibility for the day-to-day plant operation. These reports are maintained in an auditable form and maintained for inspection.

Findings from the onsite physical security program reviews are entered into the facility's corrective action program.

The NRC staff has reviewed TVA's description in PSP Section 17 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the PSP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the PSP meets the requirements of 10 CFR 73.55(m) and is, therefore, acceptable.

13.6.4.1.18 Response Requirements

The regulation at 10 CFR 73.55(k) requires, in part, that the licensee establish and maintain a properly trained, qualified, and equipped security force required to interdict and neutralize threats up to and including the DBT defined in 10 CFR 73.1, to prevent significant core damage and spent fuel sabotage. To meet this objective, the licensee must ensure that necessary equipment is in supply, working, and readily available. The licensee must ensure that training has been provided to all armed members of the security organization who will be available on site to implement the applicant's protective strategy, as described in facility procedures and 10 CFR Part 73, Appendix C. The licensee must have facility procedures to reconstitute armed response personnel and have established working agreement(s) with an LLEA. The licensee must implement a threat warning system, which identifies specific graduated protective measures and actions to be taken against a heightened security threat.

Section 18 of the PSP describes an armed response team, responsibilities, training, and equipment and requires a number of armed response force personnel to be immediately available at all times to implement the site's protective strategy. TVA ensures that training is conducted in accordance with the requirements of 10 CFR Part 73, Appendix B, which will ensure implementation of the site protective strategy in accordance with 10 CFR Part 73, Appendix C. Procedures are in place to reconstitute the armed response personnel, as are agreements with an LLEA. Procedures are in place to manage the threat warning system.

The regulation at 10 CFR 73.55(k) states the following:

> (1) The licensee shall establish and maintain, at all times, properly trained, qualified and equipped personnel required to interdict and neutralize threats up to and including the design basis threat of radiological sabotage as defined in § 73.1, to prevent significant core damage and spent fuel sabotage.
> (2) The licensee shall ensure that all firearms, ammunition, and equipment necessary to implement the site security plans and protective strategy are in sufficient supply, are in working condition, and are readily available for use.
> (3) The licensee shall train each armed member of the security organization to prevent or impede attempted acts of radiological sabotage by using force sufficient to counter the force directed at that person, including the use of deadly force when the armed member of the security organization has a reasonable belief that the use of deadly force is necessary in self-defense or in the defense of others, or any other circumstances as authorized by applicable State or Federal law.
> (4) The licensee shall provide armed response personnel consisting of armed responders which may be augmented with armed security officers to carry out

armed response duties within predetermined time lines specified by the site protective strategy.

In RAI 13.6-10, the NRC staff requested that TVA provide more information about the methodology used in determining the staffing requirements for armed responders and armed security officers based on the development of Unit 2 vital areas, target sets, and timelines. The staff requested that TVA confirm or describe the types of armed personnel that will make up the armed response force for the additional Unit 2 vital areas and equipment, including how this process was used as a basis for establishing the Unit 2 target sets and calculating responders' timelines.

By letter dated April 5, 2010, TVA responded that target sets for WBN Unit 2 have not been developed at this time. TVA will utilize the expert panel methodology in determining the staffing requirements for armed responders and armed security officers based on the development of Unit 2 vital areas, target sets, and timelines. Appendix B to Revision 10 of the PSP contains the types of armed responders and their training and qualification. Target sets for WBN Unit 2 will be developed by August 20, 2010.

On the basis of its review of (1) the information provided by TVA in its letter dated April 5, 2010, (2) the WBN PSP, Revision 11, dated July 23, 2010, and (3) the TVA letter, "Response to Request for Additional Information Regarding Target Set Development," dated November 18, 2010, the NRC staff concludes that the TVA PSP meets the requirements of 10 CFR 73.55(f) for WBN Unit 2.

The NRC staff has reviewed TVA's description in PSP Section 18 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the PSP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the PSP meets the requirements of 10 CFR 73.55(k) and is, therefore, acceptable.

13.6.4.1.19 Special Situations Affecting Security

The regulation at 10 CFR 73.58, "Safety/Security Interface Requirements for Nuclear Power Reactors," states, in part, the following:

> (b) The licensee shall assess and manage the potential for adverse effects on safety and security, including the site emergency plan, before implementing changes to plant configurations, facility conditions, or security.
> (c) The scope of changes to be assessed and managed must include planned and emergent activities (such as, but not limited to, physical modifications, procedural changes, changes to operator actions or security assignments, maintenance activities, system reconfiguration, access modification or restrictions, and changes to the security plan and its implementation).
> (d) Where potential conflicts are identified, the licensee shall communicate them to appropriate licensee personnel and take compensatory and/or mitigative actions to maintain safety and security under applicable Commission regulations, requirements, and license conditions.

Section 19 of the PSP includes requirements for assessment to manage increased risk of special situations affecting security.

Refueling/Major Maintenance

Section 19.1 of the PSP states that security procedures identify measures for implementation of actions before refueling or major maintenance activities. These measures include controls to ensure that a search is conducted before revitalizing an area, that protective barriers and alarms are fully operational, and that postmaintenance performance testing is done to ensure the operational readiness of equipment per 10 CFR 73.55(n)(8).

Construction and Maintenance

Section 19.2 of the PSP explains that, during periods of construction and maintenance when temporary modifications are necessary, TVA will implement actions that provide for equivalency in the physical protection measures and features impacted by the activities such that physical protection measures are not degraded. Facility procedures describe the process for making such changes or modifications.

Section 19.2 of the PSP also contains the description of the security support for construction activities for the Unit 2 reactor building.

The NRC staff has reviewed TVA's description in PSP Sections 19, 19.1, and 19.2 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the PSP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the PSP meets the requirements of 10 CFR 73.55(n)(8) and 10 CFR 73.58 and is, therefore, acceptable.

13.6.4.1.20 Maintenance, Testing, and Calibration

The regulation at 10 CFR 73.55(n) provides requirements for maintenance, testing, and calibration.

Sections 20.1 through 20.6 of the PSP describe the maintenance, testing, and calibration program for security-related equipment. Section 20.1 states that TVA shall conduct intrusion detection testing in accordance with RG 5.44. Each operational component required for the implementation of the security program is, at a minimum, tested in accordance with 10 CFR 73.55(n), the PSP, and implementing procedures.

The regulation at 10 CFR 73.55(n) states, in part, the following:

> (1) The licensee shall:
> (i) Establish, maintain, and implement a maintenance, testing and calibration program to ensure that security systems and equipment, including secondary and uninterruptible power supplies, are tested for operability and performance at predetermined intervals, maintained in operable condition, and are capable of performing their intended functions.
> (ii) Describe the maintenance, testing and calibration program in the physical security plan. Implementing procedures must specify operational and technical details required to perform maintenance, testing, and calibration activities to

include, but not limited to, purpose of activity, actions to be taken, acceptance criteria, and the intervals or frequency at which the activity will be performed.
(iii) Identify in procedures the criteria for determining when problems, failures, deficiencies, and other findings are documented in the site corrective action program for resolution.
(iv) Ensure that information documented in the site corrective action program is written in a manner that does not constitute safeguards information as defined in 10 CFR 73.21.
(v) Implement compensatory measures that ensure the effectiveness of the onsite physical protection program when there is a failure or degraded operation of security-related component or equipment.

In RAI 13.6-12, the NRC staff asked TVA to clarify how the current requirements for tamper and line supervision alarm testing meet the acceptance criteria of RG 5.44. By letter dated April 5, 2010, TVA responded that it developed the site procedure (SSI 15.1, Testing and Maintenance) for current tamper and alarm supervision testing using the criteria in RG 5.44, Sections 1.4 and 1.5. No exceptions were taken from these sections. The procedure is available at WBN for audit.

On December 5, 2000, the NRC issued License Amendment No. 29 (ADAMS Accession No. ML003774549) to WBN Unit 1, regarding testing frequency for tamper switch/line supervision alarms.

On the basis of its review of License Amendment No. 29 and the information provided by TVA, the NRC staff concluded that Section 20.1 of the WBN PSP, Revision 10, dated March 31, 2010, is inadequate, because Revision 10 of the PSP omitted the approved tamper and alarm supervision testing process.

The NRC and TVA staffs discussed the issue during a secure telephone call on June 29, 2010. As a result, TVA revised the WBN PSP to include the reference to the site procedure (SSI 15.1, "Testing and Maintenance") for tamper and alarm supervision testing. Based on its review of WBN PSP, Revision 11, dated July 23, 2010, the NRC staff concluded that the PSP meets the requirements of 10 CFR 73.55(n)(1).

The NRC staff has reviewed TVA's description in PSP Sections 20 and 20.1 through 20.6 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the PSP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the PSP meets the requirements of 10 CFR 73.55(n) and is, therefore, acceptable.

13.6.4.1.21 Compensatory Measures

The regulation at 10 CFR 73.55(o) states, in part, the following:

(1) The licensee shall identify criteria and measures to compensate for degraded or inoperable equipment, systems, and components to meet the requirements of this section. (2) Compensatory measures must provide a level of protection that is equivalent to the protection that was provided by the degraded or inoperable,

equipment, system, or components. (3) Compensatory measures must be implemented within specific time frames necessary to meet the requirements....

Section 21 of the PSP identifies measures and criteria required to compensate for degraded or inoperable equipment, systems, and components in accordance with 10 CFR 73.55(o) to ensure that the effectiveness of the physical protection system is not reduced by failure or other contingencies affecting the operation of the security-related equipment or structures. Sections 21.1 through 21.12 of the PSP address PA and vital area barriers, intrusion detection and alarm systems, lighting, alarm systems, fixed and nonfixed closed circuit television, playback and recorded video systems, computer systems, access control devices, vehicle barrier systems, channeling barrier systems, and other security-related equipment.

The NRC staff has reviewed TVA's description in PSP Sections 21 and 21.1 through 21.12 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the PSP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the PSP meets the requirements of 10 CFR 73.55(o) and is, therefore, acceptable.

13.6.4.1.22 Records

Requirements for recordkeeping appear in 10 CFR Part 26; 10 CFR 73.55(q); 10 CFR 73.56(k) and (o); Appendix B, Section VI.H, and Appendix C, Section II.C, to 10 CFR Part 73; and 10 CFR 73.70, "Records." These regulations state that the licensee must retain and maintain all records required to be kept by Commission regulations, orders, or license conditions until the Commission terminates the license for which the records were developed, and shall maintain superseded portions of these records for at least 3 years after the record is superseded, unless otherwise specified by the Commission. The licensee is required to keep records of contracts with any contracted security force that implements any portion of the onsite physical protection program for the duration of the contract. The licensee must make all records required to be kept by the Commission available to the Commission, and the Commission may inspect, copy, retain, and remove all such records, reports, and documents whether kept by the licensee or a contractor. Review and audit reports must be maintained and available for inspection for a period of 3 years.

Section 22.0 of the PSP addresses the requirements to maintain records. Sections 22.1 through 22.13 address each kind of record that the applicant will maintain and the duration of retention for each record. The following types of records are maintained in accordance with the above regulations: Access Authorization Records; Suitability, Physical, and Psychological Qualification Records for Security Personnel; PA and Vital Area Access Control Records; PA Visitor Access Records; PA Vehicle Access; VA Access Transaction Records; Vitalization and Devitalization Records; VA Access List Reviews; Security Plans and Procedures; Security Patrols, Inspections, and Tests; Maintenance; CAS and SAS Alarm Annunciation and Security Response Records; LLEA Records; Records of Audits and Reviews; Access Control Devices; Security Training and Qualification Records; Firearms Testing and Maintenance Records; and Engineering Analysis for the Vehicle Barrier System.

The NRC staff has reviewed TVA's description in PSP Sections 22 and 22.1 through 22.13 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the PSP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the PSP meets the requirements of 10 CFR 73.55(q), 10 CFR 73.55(o), and 10 CFR 73.70 and is, therefore, acceptable.

13.6.4.1.23 Digital Systems Security

Section 23 of the PSP addresses digital systems security. TVA stated in its PSP that it has implemented the requirements of 10 CFR 73.54 and maintains a cyber security plan that describes how it has provided high assurance that safety, security, and emergency preparedness functions are protected against the DBT.

Section 13.6.4.1.14 of this SSER addresses the NRC staff's review of the cyber security plan.

13.6.4.1.24 Temporary Suspension of Security Measures

The regulation at 10 CFR 73.55(p) describes the conditions under which a licensee may suspend security measures.

Suspension of Security Measures in Accordance with 10 CFR 50.54(x) and (y)

Section 24.1 of the PSP addresses suspension of security measures in accordance with 10 CFR 50.54(x) and 10 CFR 50.54(y). Specifically, the plan describes the conditions under which suspension is permissible, the authority for suspension, and the requirements for reporting such a suspension.

Suspension of Security Measures During Severe Weather or Other Hazardous Conditions

As required in 10 CFR 73.55(p), suspension of security measures is reported and documented in accordance with the provisions of 10 CFR 73.71, "Reporting of Safeguards Events." Before such action, the suspension of security measures must be approved, as a minimum, by a licensed senior operator, with input from the security supervisor or manager. Suspended security measures must be reinstated as soon as conditions permit.

Section 24.2 of the PSP provides that certain security measures may be temporarily suspended during circumstances such as imminent severe or hazardous weather conditions, but only when such action is immediately needed to protect the personal health and safety of security force personnel and no other immediately apparent action consistent with the security measures can provide adequate or equivalent protection. Under the PSP, suspended security measures shall be restored as soon as practical.

The NRC staff has reviewed TVA's description in PSP Sections 24, 24.1, and 24.2 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the PSP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the PSP meets the requirements of 10 CFR 73.55(p) and is, therefore, acceptable.

13.6.4.1.25 Appendix A, Glossary of Terms and Acronyms

The NRC staff reviewed WBN PSP, Appendix A, "Glossary of Terms and Acronyms," and found it to be consistent with the template of NRC-endorsed NEI 03-12, Revision 6.

13.6.4.1.26 Conclusions on the Physical Security Plan

On the basis of the NRC staff's review described in Sections 13.6.4.1.1 through 13.6.4.1.25 of this SSER, the NRC staff concludes that the PSP meets the requirements of 10 CFR 73.55(a) through (r). The NRC staff also concludes that complete and procedurally correct implementation of the PSP will provide high assurance that activities involving special nuclear material are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety.

The target sets, site-specific target set analysis, and site protective strategy are in the facility implementing procedures, which were not subject to NRC staff review as part of this application but are subject to future NRC audit and inspection in accordance with 10 CFR 73.55(c)(7)(iv) and 10 CFR Part 73, Appendix C, Section II.B.5(iii).

13.6.4.2 <u>Appendix B, Training and Qualification Plan</u>

13.6.4.2.1 Introduction

The regulation at 10 CFR 73.55(c)(4) states that the licensee shall establish, maintain, implement, and follow a T&QP that describes how the licensee will implement the criteria in 10 CFR Part 73, Appendix B.

The regulation at 10 CFR 73.55(d)(3) states the following:

> The licensee may not permit any individual to implement any part of the physical protection program unless the individual has been trained, equipped, and qualified to perform their assigned duties and responsibilities in accordance with appendix B to this part and the [T&QP]. Non-security personnel may be assigned duties and responsibilities required to implement the physical protection program and shall:
> (i) Be trained through established licensee training programs to ensure each individual is trained, qualified, and periodically re-qualified to perform assigned duties.
> (ii) Be properly equipped to perform assigned duties.
> (iii) Possess the knowledge, skills, and abilities, to include physical attributes such as sight and hearing, required to perform their assigned duties and responsibilities.

In addition, 10 CFR Part 73, Appendix B, Section VI.D.2(a), states that armed and unarmed individuals shall be requalified at least annually in accordance with the requirements of the Commission-approved T&QP.

The WBN Unit 2 T&QP states that it is written to address the requirements of 10 CFR Part 73, Appendix B, Section VI. The objective of the plan is to provide a mechanism to ensure that members of the security organization, and all others who have duties and responsibilities in implementing the security requirements and protective strategy, are properly trained, equipped, and qualified. Deficiencies identified during the administration of T&QP requirements are documented in the site corrective action program.

The NRC staff has reviewed the introduction section in the T&QP and has determined that it includes all of the programmatic elements necessary to satisfy the requirements of 10 CFR 73.55 and 10 CFR Part 73, Appendix B, Section VI, that are applicable to the T&QP. Additional section-by-section evaluation and discussion appear in the following paragraphs.

13.6.4.2.2 Employment Suitability and Qualification

The requirements for mental qualifications, documentation, and physical requalification for security personnel (applicant employee and contractor) are described in the following T&QP sections.

Suitability

The regulation at 10 CFR Part 73, Appendix B, Section VI.B.1(a), states the following:

> Before employment, or assignment to the security organization, an individual shall:
> (1) Possess a high school diploma or pass an equivalent performance examination designed to measure basic mathematical, language, and reasoning skills, abilities, and knowledge required to perform security duties and responsibilities;
> (2) Have attained the age of 21 for an armed capacity or the age of 18 for an unarmed capacity; and
> (3) Not have any felony convictions that reflect on the individual's reliability.
> (4) Individuals in an armed capacity, would not be disqualified from possessing or using firearms or ammunition in accordance with applicable State or Federal law, to include 18 U.S.C. 922. Licensees shall use information that has been obtained during the completion of the individual's background investigation for unescorted access to determine suitability. Satisfactory completion of a firearms background check for the individual under 10 CFR 73.19 of this part will also fulfill this requirement.

The provisions of 10 CFR Part 73, Appendix B, Section VI.B.1(b), require that the qualification of each individual to perform assigned duties and responsibilities must be documented by a qualified training instructor and attested to by a security supervisor.

Section 2.1 of the T&QP details the requirements of qualifications for employment in the security organization that follow the regulation in 10 CFR Part 73, Appendix B, Section VI.B.1.

Physical Qualifications

The regulation at 10 CFR Part 73, Appendix B, Section VI.B.2, requires, in part, that individuals whose duties and responsibilities are directly associated with the effective implementation of the Commission-approved security plans, licensee protective strategy, and implementing procedures, may not have any physical conditions that would adversely affect their performance of assigned security duties and responsibilities.

Section 2.2 of the T&QP states that individuals who are directly associated with implementation of the security plans, protective strategy, and procedures may not have any physical conditions that would adversely affect their performance of assigned security duties and responsibilities. All individuals found on the Critical Task Matrix shall demonstrate the necessary physical qualifications before duty.

Physical Examination

The regulation at 10 CFR Part 73, Appendix B, Section VI.B.2(a)(2), states that armed and unarmed individuals assigned security duties and responsibilities shall be subject to a physical examination designed to measure the individual's physical ability to perform assigned duties and responsibilities as identified in the Commission-approved security plans, licensee protective strategy, and implementing procedures.

The regulation at 10 CFR Part 73, Appendix B, Section VI.B.2(a)(3), states that the physical examination must be administered by a licensed health professional. A licensed physician must make the final determination to verify the individual's physical capability to perform assigned duties and responsibilities.

The regulations at 10 CFR Part 73, Appendix B, Section VI.B.2(a)(4)(b) through (e), provide the minimum requirements that individuals must meet, including requirements for vision, hearing, review of existing medical conditions, and examination for potential addictions.

The provisions of 10 CFR Part 73, Appendix B, Section VI.B.2(f) address the medical examination of individuals before their return to assigned duties following any incapacitation.

Section 2.3 of the T&QP describes the physical examinations for armed and unarmed individuals assigned security duties, as well as other individuals who implement parts of the physical protection program. Minimum requirements exist for physical examinations of vision, hearing, existing medical conditions, addiction, or other physical requirements.

The NRC staff has reviewed TVA's description in T&QP Sections 2.1, 2.2, and 2.3 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the T&QP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the T&QP meets the requirements of 10 CFR Part 73, Appendix B, Sections VI.B.1 and VI.B.2, and is, therefore, acceptable.

Medical Examinations and Physical Fitness Qualifications

The regulation at 10 CFR Part 73, Appendix B, Section VI.B.4(a), requires, in part, that armed members of the security organization shall be subject to a medical examination by a licensed physician, to determine the individual's fitness to participate in physical fitness tests. The regulation also states that the licensee shall obtain and retain a written certification from the licensed physician that no medical conditions were disclosed by the medical examination that would preclude the individual's ability to participate in the physical fitness tests or meet the physical fitness attributes or objectives associated with assigned duties.

The regulation at 10 CFR Part 73, Appendix B, Section VI.B.4(b), requires, in part, that before assignment, armed members of the security organization shall demonstrate physical fitness for assigned duties and responsibilities by performing a practical physical fitness test. The physical fitness test must consider physical conditions such as strenuous activity, physical exertion, levels of stress, and exposure to the elements as they pertain to each individual's assigned security duties. The physical fitness qualification of each armed member of the security organization must be documented by a qualified training instructor and attested to by a security supervisor.

Section 2.4 of the T&QP is explicit in its requirements for medical examinations and physical qualifications.

The NRC staff has reviewed TVA's description in T&QP Section 2.4 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the T&QP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the T&QP meets the requirements of 10 CFR Part 73, Appendix B, Sections VI.B.4(a) and VI.B.4(b), and is, therefore, acceptable.

Psychological Qualifications

General Psychological Qualifications

The regulation at 10 CFR Part 73, Appendix B, Section VI.B.3(a), requires, in part, that armed and unarmed individuals shall demonstrate the ability to apply good judgment, show mental alertness, have the capability to implement instructions and assigned tasks, and possess the acuity of senses and ability of expression sufficient to permit accurate communication by written, spoken, audible, visible, or other signals required by assigned duties and responsibilities.

Section 2.5.1 of the T&QP states in detail that individuals who have security tasks and jobs directly associated with the effective implementation of the security plan and protective strategy shall demonstrate the qualities in 10 CFR Part 73, Appendix B, Section VI.B.3(a).

Professional Psychological Examination

Among the requirements of 10 CFR Part 73, Appendix B, Section VI.B.3(b), is that a licensed psychologist, psychiatrist, or physician trained in part to identify emotional instability shall determine whether armed members of the security organization and alarm station operators, in addition to meeting the requirement stated in paragraph (a) of Section VI.B.3, have no emotional instability that would interfere with the effective performance of assigned duties and responsibilities.

The regulation at 10 CFR Part 73, Appendix B, Section VI.B.3(c), requires that a person professionally trained to identify emotional instability shall determine whether unarmed individuals, in addition to meeting the requirement stated in paragraph (a) of this section, have no emotional instability that would interfere with the effective performance of assigned duties and responsibilities.

Section 2.5.2 of the T&QP provides for the determination of psychological and emotional state by appropriately licensed and trained individuals.

The NRC staff has reviewed TVA's description in T&QP Sections 2.5.1 and 2.5.2 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the T&QP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the T&QP meets the requirements of 10 CFR Part 73, Appendix B, Section VI.B.3(a), (b), and (c), and is, therefore, acceptable.

Documentation

The regulation at 10 CFR Part 73, Appendix B, Section VI.H.1, requires, in part, the retention of all reports, records, or other documentation required by Appendix B and 10 CFR 75.55(q).

Section 2.6 of the T&QP states that qualified training instructors document training activities and that security supervisors attest to these records as required. Records are retained in accordance with Section 22 of the PSP.

The NRC staff has reviewed TVA's description in T&QP Section 2.6 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the T&QP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the T&QP meets the requirements of 10 CFR Part 73, Appendix B, Section VI.H.1, and is, therefore, acceptable.

Physical Requalification

The regulation at 10 CFR Part 73, Appendix B, Section VI.B.5, requires that (a) at least annually, armed and unarmed individuals shall be required to demonstrate the capability to meet the physical requirements of Appendix B and the licensee T&QP and (b) the physical requalification of each armed and unarmed individual must be documented by a qualified training instructor and attested to by a security supervisor.

Section 2.7 of the T&QP states that physical requalification is conducted at least annually and documented as described in the PSP.

The NRC staff has reviewed TVA's description in T&QP Section 2.7 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the T&QP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description provided in the T&QP meets the requirements of 10 CFR Part 73, Appendix B, Section VI.B.5, and is, therefore, acceptable.

13.6.4.2.3 Individual Training and Qualification

Duty Training

The regulation at 10 CFR Part 73, Appendix B, Section VI.C.1, specifies duty training and qualification requirements. The regulation states, in part, that all personnel who are assigned to perform any security-related duty or responsibility shall be trained and qualified to perform assigned duties and responsibilities to ensure that each individual possesses the minimum knowledge, skills, and abilities required to effectively carry out those assigned duties and responsibilities. These areas of training include performing assigned duties and responsibilities in accordance with the requirements of the T&QP and the PSP and training and qualification in the use of all equipment or devices required to effectively perform all assigned duties and responsibilities.

Section 3.1 of the T&QP details the requirements that individuals assigned duties must be trained in their duties, meet minimum qualifications, and be trained and qualified in all equipment or devices required to perform their duties.

The NRC staff has reviewed TVA's description in T&QP Sections 3.0, and 3.1 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the T&QP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the T&QP meets the requirements of 10 CFR Part 73, Appendix B, Section VI.C.1, and is, therefore, acceptable.

On-the-Job Training

The regulations at 10 CFR Part 73, Appendix B, Section VI.C.2(a) through (c), specify requirements for on-the-job training. On-the-job training must include individual demonstration of the knowledge, skills and abilities provided during the training process. Individuals who are assigned contingency duties must complete a minimum of 40 hours of on-the-job training.

On-the-job training for contingency activities and drills must include, but is not limited to, hands-on application of knowledge, skills, and abilities related to (1) response team duties, (2) use of force, (3) tactical movement, (4) cover and concealment, (5) defensive positions, (6) fields of fire, (7) redeployment, (8) communications (primary and alternate), (9) use of assigned equipment, (10) target sets, (11) tabletop drills, (12) command and control duties, (13) licensee protective strategy.

The T&QP provides a comprehensive discussion of TVA's approach to meeting the requirements for on-the-job training.

The NRC staff has reviewed TVA's description in T&QP Section 3.2 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the T&QP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the T&QP meets the requirements of 10 CFR Part 73, Appendix B, Section VI.C.2(a) through (c), and is, therefore, acceptable.

Critical Task Matrix

The regulation at 10 CFR Part 73, Appendix B, Section VI.C.2(b), requires, in part, that each individual who is assigned duties and responsibilities identified in the Commission-approved security plans, licensee protective strategy, and implementing procedures shall, before assignment, demonstrate proficiencies in implementing the knowledge, skills, and abilities to perform assigned duties.

The T&QP contains a Critical Task Matrix as Table 1. This matrix addresses the means through which each individual will demonstrate the required proficiencies. RG 5.75 lists tasks that individuals must perform.

The NRC staff has reviewed TVA's description in T&QP Section 3.3 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the T&QP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the T&QP meets the requirements of 10 CFR Part 73, Appendix B, Section VI.C.2(b), and is, therefore, acceptable.

<u>Initial Training and Qualification Requirements</u>

The regulation at 10 CFR Part 73, Appendix B, Section VI.C, specifies the requirements for duty training.

The regulation at 10 CFR Part 73, Appendix B, Section VI.D.1, specifies the requirements for demonstration of qualification.

Section 3.4 of the T&QP states that individuals are trained and qualified before performing security-related duties within the security organization and must meet the minimum qualifying standards in Sections 3.4.1 and 3.4.2.

<u>Written Examination</u>

The regulation at 10 CFR Part 73, Appendix B, Section VI.D.1(b)(1), requires that written exams must include those elements listed in the Commission-approved T&QP to demonstrate an acceptable understanding of assigned duties and responsibilities, to include the recognition of potential tampering involving both safety and security equipment and systems.

<u>Hands-On Performance Demonstration</u>

The regulation at 10 CFR Part 73, Appendix B, Section VI.D.1(b)(2), requires that armed and unarmed individuals shall demonstrate hands-on performance for assigned duties and responsibilities by performing a practical hands-on demonstration for required tasks. The hands-on demonstration must ensure that theory and associated learning objectives for each required task are considered and that each individual demonstrates the knowledge, skills, and abilities required to effectively perform the task.

Sections 3.4.1 and 3.4.2 of the T&QP describe the measures that are implemented by TVA to meet the requirements stated above.

The NRC staff has reviewed TVA's description in T&QP Sections 3.4, 3.4.1, and 3.4.2 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the T&QP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the T&QP meets the requirements of 10 CFR Part 73, Appendix B, Sections VI.C.1 and D.1, and is, therefore, acceptable.

Continuing Training and Qualification

The regulation at 10 CFR Part 73, Appendix B, Section VI.D.2, states, in part, that armed and unarmed individuals shall be requalified at least annually in accordance with the requirements of this appendix and the Commission-approved T&QP. The results of requalification must be documented by a qualified training instructor and attested to by a security supervisor.

Section 3.5 of the T&QP describes the management of the requalification program to ensure that each individual is trained and qualified. In part, TVA's plan provides that annual requalification may be completed up to 3 months before or 3 months after the scheduled date. However, the next annual training must be scheduled 12 months from the previously scheduled date rather than from the date when the training was actually completed.

The NRC staff has reviewed TVA's description in T&QP Section 3.5 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the T&QP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the T&QP meets the requirements of 10 CFR Part 73, Appendix B, Section VI.D.2, and is, therefore, acceptable.

Annual Written Examination

The regulation at 10 CFR Part 73, Appendix B, Section VI.D.1(3), requires that armed individuals shall be administered an annual written exam that demonstrates the required knowledge, skills, and abilities to carry out assigned duties and responsibilities as an armed member of the security organization. The annual written exam must include those elements listed in the Commission-approved T&QP to demonstrate an acceptable understanding of assigned duties and responsibilities.

Section 3.5.1 of the T&QP provides that each individual will be tested, in part, with an annual written exam that, at a minimum, covers the role of security personnel, use of deadly force, the requirements in 10 CFR 73.21, authority of private security personnel, power of arrest, search and seizure, offsite law enforcement response, and tactics and tactical deployment and engagement.

The NRC staff has reviewed TVA's description in T&QP Section 3.5.1 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the T&QP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the T&QP meets the requirements of 10 CFR Part 73, Appendix B, Section VI.D.1(3), and is, therefore, acceptable.

Demonstration of Knowledge, Skills, and Abilities

The regulations at 10 CFR Part 73, Appendix B, Section VI, paragraphs A.4., B.2(c)(2), B.3(a), B.4(b)(1), B.4(b)(3), B.5(a), C.2(a), C.2(b), C.3(a), C.3(b), C.3(d), D.1(a), D.1(b)(1), D.1(b)(2), D.1(b)(3), and D.1(c) state, in part, that an individual must demonstrate required knowledge, skills, and abilities to carry out assigned duties and responsibilities.

Section 3.5.2 of the T&QP provides that all knowledge, skills, and abilities will be demonstrated in accordance with a SAT program, similar to that described in RG 5.75.

The NRC staff has reviewed TVA's description in T&QP Section 3.5.2 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the T&QP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the T&QP meets the requirements of 10 CFR Part 73, Appendix B, Sections VI.A, B, C, and D, and is, therefore, acceptable.

<u>Weapons Training and Qualification</u>

General Firearms Training:

The regulation at 10 CFR Part 73, Appendix B, Section VI.E, requires that armed members of the security organization shall be trained and qualified in accordance with the requirements of Appendix B and the Commission-approved T&QP. Certified firearms instructors, who shall be recertified at least every 3 years, must conduct the training. Licensees shall conduct annual firearms familiarization, and armed members of the security organization must participate in weapons range activities on a nominal 4-month periodicity.

Section 3.6.1 of the T&QP addresses the requirements in 10 CFR Part 73, Appendix B, Section VI.E.1(d)(1) through (11) and includes the requirements for training in the use of deadly force and participation in weapons range activities on a nominal 4-month periodicity.

The NRC staff has reviewed TVA's description in T&QP Section 3.6.1 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the T&QP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the T&QP meets the requirements of 10 CFR Part 73, Appendix B, Section VI.E.1, and is, therefore, acceptable.

General Weapons Qualification:

The regulation at 10 CFR Part 73, Appendix B, Section VI.F.1, requires that qualification firing must be accomplished in accordance with Commission requirements and the Commission-approved T&QP for assigned weapons. The results of weapons qualification and requalification must be documented and retained as a record.

Section 3.6.2 of the T&QP provides that all armed personnel are qualified and requalified with assigned weapons. All weapons qualification and requalification will be documented and retained as a record.

The NRC staff has reviewed TVA's description in T&QP Section 3.6.2 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the T&QP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the T&QP meets the requirements of 10 CFR Part 73, Appendix B, Section VI.F.1, and is, therefore, acceptable.

Tactical Weapons Qualification:

The regulation at 10 CFR Part 73, Appendix B, Section VI.F.2, requires that the licensee conduct tactical weapons qualification. The licensee T&QP must describe the firearms used, the firearms qualification program, and other tactical training required to implement the Commission-approved security plans, licensee protective strategy, and implementing procedures. Licensee-developed tactical qualification and requalification courses must describe the performance criteria needed to include the site-specific conditions (such as lighting, elevation, fields-of-fire) under which assigned personnel shall be required to carry out their assigned duties.

Section 3.6.3 of the T&QP provides that a tactical qualification course of fire is used to assess armed security force personnel in tactical situations to ensure that they are able to demonstrate that their required tactical knowledge, skills, and abilities remain proficient.

The NRC staff has reviewed TVA's description in T&QP Section 3.6.3 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the T&QP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the T&QP meets the requirements of 10 CFR Part 73, Appendix B, Section VI.F.3, and is, therefore, acceptable.

Firearms Qualification Courses:

The regulation at 10 CFR Part 73, Appendix B, Section VI.F.3, states, in part, that the licensee shall conduct the following qualification courses for each weapon used: (a) an annual daylight fire qualification course and (b) an annual night fire qualification course.

Courses of Fire:

The regulation at 10 CFR Part 73, Appendix B, Section VI.F.4, specifies the required courses of fire.

Section 3.6.4 of the T&QP describes the firearms qualification courses used to ensure that armed members of the security organization are properly trained and qualified. Courses of fire are used individually for handguns, semiautomatic rifles, and enhanced weapons.

The NRC staff has reviewed TVA's description in T&QP Section 3.6.4 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the T&QP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the T&QP meets the requirements of 10 CFR Part 73, Appendix B, Sections VI.F.3 and VI.F.4, and is, therefore, acceptable.

Firearms Requalification:

The regulation at 10 CFR Part 73, Appendix B, Section VI.F.5, requires that armed members of the security organization shall be requalified for each assigned weapon at least annually, in accordance with Commission requirements and the Commission-approved T&QP, and the results documented and retained as a record. Firearms requalification must be conducted using the courses of fire outlined in 10 CFR Part 73, Appendix B, Sections VI.F.2, VI.F.3, and VI.F.4.

Section 3.6.5 of the T&QP states that armed members of the security organization requalify at least annually with each weapon assigned, using the courses of fire provided in the T&QP.

The NRC staff has reviewed TVA's description in T&QP Section 3.6.5 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the T&QP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the T&QP meets the requirements of 10 CFR Part 73, Appendix B, Section VI.F.5, and is, therefore, acceptable.

Weapons, Personal Equipment, and Maintenance

The regulation at 10 CFR Part 73, Appendix B, Section VI.G, specifies the requirements for the maintenance of weapons and personal equipment. These requirements state that the licensee shall provide armed personnel with weapons that are capable of performing the function stated in the Commission-approved security plans, licensee protective strategy, and implementing procedures. In addition, the licensee shall ensure that each individual is equipped or has ready access to all personal equipment or devices required for the effective implementation of the Commission-approved security plans, licensee protective strategy, and implementing procedures.

Section 3.7 of the T&QP states that personnel are provided with weapons and personal equipment necessary to meet the plans and the protective strategy. Section 9.0 of the PSP describes the equipment, and Section 20.0 of the PSP describes the performance of maintenance.

The NRC staff has reviewed TVA's description in T&QP Section 3.7 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the T&QP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the T&QP meets the requirements of 10 CFR Part 73, Appendix B, Section VI.G, and is, therefore, acceptable. The staff's review of Sections 9.0 and 20.0 of the PSP is in Sections 13.6.4.1.9 and 13.6.4.1.20 of this SER.

Documentation

The regulation at 10 CFR Part 73, Appendix B, Section VI.H, requires that the licensee shall retain all reports, records, or other documentation required by Appendix B, in accordance with the requirements of 10 CFR 73.55(r). The licensee shall retain each individual's initial qualification record for 3 years after termination of the individual's employment and shall retain each requalification record for 3 years after it is superseded. The licensee shall document data and test results from each individual's suitability, physical, and psychological qualification and shall retain this documentation as a record for 3 years from the date of obtaining and recording these results.

Section 3.8 of the T&QP provides that records are retained in accordance with Section 22 of the PSP.

The NRC staff has reviewed TVA's description in T&QP Section 3.8 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the T&QP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the T&QP meets the requirements of 10 CFR Part 73, Appendix B, Section VI.H, and is, therefore, acceptable.

13.6.4.2.4 Performance Evaluation Program

The regulations at 10 CFR Part 73, Appendix B, Section VI.C.3, specify the requirements for the performance evaluation program.

Section 4 of the T&QP details the performance evaluation program consistent with the requirements of 10 CFR Part 73, Appendix B, Section VI.C.3(a) through (m). Facility procedures offer additional details of the performance evaluation program.

The NRC staff has reviewed TVA's description in T&QP Section 4 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the T&QP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the T&QP meets the requirements of 10 CFR Part 73, Appendix B, Section VI.C.3, and is, therefore, acceptable.

13.6.4.2.5 Definitions

The regulation at 10 CFR Part 73, Appendix B, Section VI.J, states, in part, that terms defined in 10 CFR Part 50, 10 CFR Part 70 ("Domestic Licensing of Special Nuclear Material"), and 10 CFR Part 73 have the same meaning when used in this appendix. Definitions are found in the PSP, Appendix A, "Glossary of Terms and Acronyms."

On the basis of its review, the NRC staff finds that the definitions sections of the PSP meet the requirements of 10 CFR 73.2 and are, therefore, acceptable.

Included in this section of the T&QP is the Critical Task Matrix, which is designated as SGI and not included in this SSER.

The NRC staff has reviewed TVA's description in the T&QP of the Critical Task Matrix tasks for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the T&QP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the T&QP meets the requirements of 10 CFR Part 73, Appendix B, and is, therefore, acceptable.

13.6.4.2.6 Conclusion on the Training and Qualification Plan

On the basis of the NRC staff's review described in Sections 13.6.4.2.1 through 13.6.4.2.5 of this SER, the T&QP meets the requirements of 10 CFR Part 73, Appendix B. The NRC staff concludes that complete and procedurally correct implementation will provide high assurance that activities involving special nuclear material are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety.

The target sets, site-specific target set analysis, and site protective strategy are in the facility implementing procedures, which were not subject to NRC staff review as part of this application. These are subject to future NRC inspection in accordance with 10 CFR 73.55(c)(7)(iv) and 10 CFR Part 73, Appendix C, Section II.B.5(iii).

13.6.4.3 10 CFR Part 73, Appendix C, Safeguards Contingency Plan

13.6.4.3.1 Background Information

The SCP background identifies the perceived dangers and incidents that the SCP addresses and generally describes the organization of the licensee's response.

Purpose of the Safeguards Contingency Plan

The regulation at 10 CFR Part 73, Appendix C, Section II.B.1.b, states that the licensee should discuss general goals, objectives, and operational concepts underlying the implementation of the approved SCP.

Section 1.1 of the SCP details the purposes and goals of the SCP, including guidance to security and management for contingency events.

Scope of the Safeguards Contingency Plan

The regulation at 10 CFR Part 73, Appendix C, Section II.B.1.c, specifies the types of incidents that should be covered by the applicant in the SCP, how the onsite response effort is organized and coordinated to effectively respond to a safeguards contingency event and how the onsite response for safeguards contingency events has been integrated in other site emergency response procedures.

Section 1.2 of the SCP details the scope of the plan to analyze and define decisions and actions of security force personnel, as well as facility operations personnel, for achieving and maintaining safe shutdown.

Perceived Danger

The regulation at 10 CFR Part 73, Appendix C, Section II.B.1(a), requires that, consistent with the DBT specified in 10 CFR 73.1(a)(1), the licensee shall identify and describe the perceived dangers, threats, and incidents against which the SCP is designed to protect.

Section 1.3 of the SCP outlines the threats used to design the physical protection systems. TVA adequately addresses perceived danger, states the purpose of the plan, and describes the scope of the plan.

Definitions

Section 1.4 of the SCP explains that a list of terms and their definitions used in describing operational and technical aspects of the approved SCP, as required by 10 CFR Part 73, Appendix C, Section II.B.1.d, appears in Appendix A to the PSP.

The NRC staff has reviewed TVA's description in SCP Sections 1, 1.1, 1.2, 1.3, and 1.4 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the SCP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the SCP meets the requirements of 10 CFR Part 73, Appendix C, Section II.D.3, and is, therefore, acceptable.

13.6.4.3.2 Generic Planning Base

As required in 10 CFR Part 73, Appendix C, Section II.B.2, this section of the plan defines the criteria for initiation and termination of responses to security events, including the specific decisions, actions, and supporting information needed to respond to each type of incident covered by the approved SCP.

Situations Not Covered by the Safeguards Contingency Plan

Section 2.1 of the SCP details the general types of conditions that are not covered in the plan.

Situations Covered by the Safeguards Contingency Plan

The regulation at 10 CFR Part 73, Appendix C, Section II.B.2, requires, in part, that the plan identify those events that will be used for signaling the beginning or aggravation of a safeguards contingency according to how licensee personnel initially perceived them. Licensees shall ensure detection of unauthorized activities and shall respond to all alarms or other indications signaling a security event, such as penetration of a PA, vital area, or unauthorized barrier penetration (vehicle or personnel); tampering; and bomb threats, or other threat warnings—either verbal, such as telephoned threats, or implied, such as escalating civil disturbances.

The regulation at 10 CFR Part 73, Appendix C, Section II.B.2.b, requires, in part, that the plan define the specific objective to be accomplished in each identified safeguards contingency event. The objective may be to obtain a level of awareness about the nature and severity of the safeguards contingency to prepare for further responses, to establish a level of response preparedness, or to successfully nullify or reduce any adverse safeguards consequences arising from the contingency.

The regulation at 10 CFR Part 73, Appendix C, Section II.B.2.c, requires, in part, that the licensee identify the data, criteria, procedures, mechanisms, and logistical support necessary to achieve the objectives identified.

Section 2.2 of the SCP describes in detail the specific situations covered by the SCP, including objectives and information required for each.

The NRC staff has reviewed TVA's description in SCP Sections 2, 2.1, and 2.2 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the SCP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the SCP meets the requirements of 10 CFR Part 73, Appendix C, Section II.B.2, and is, therefore, acceptable.

13.6.4.3.3 Responsibility Matrix

The regulations at 10 CFR Part 73, Appendix C, Section II.B.4, specify the requirements for the Responsibility Matrix. The matrix consists of the detailed identification of responsibilities and specific actions to be taken by licensee organizations and/or personnel in response to safeguards contingency events.

Section 3 of the SCP includes the Responsibility Matrix. The Responsibility Matrix integrates the response capabilities of the security organization (described in Section 4 of the SCP) with the background information relating to decision/actions and organizational structure (described in Section 1 of the SCP). The Responsibility Matrix provides an overall description of the response actions and their interrelationships. Responsibilities and actions have been predetermined to the maximum extent possible and assigned to specific entities to preclude conflicts that would interfere with or prevent the implementation of the SCP or the ability to protect against the DBT of radiological sabotage.

The NRC staff has reviewed TVA's description in SCP Section 3 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the SCP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the SCP meets the requirements of 10 CFR Part 73, Appendix C, Section II.B.4, and is, therefore, acceptable.

13.6.4.3.4 Licensee Planning Base

The regulation at 10 CFR Part 73, Appendix C, Section II.B.3, requires, in part, that the licensee planning base include factors affecting the SCP that are specific for each facility.

Licensee Organization

The regulation at 10 CFR Part 73, Appendix C, Section II.B.3.a, requires, in part, that the SCP describe the organization's chain of command and delegation of authority during safeguards contingency events, to include a general description of how command and control functions will be coordinated and maintained.

Duties and Communication Protocols

Section 4.1.1 of the SCP details the duties and communications protocols of each member of the security organization responsible for implementing any portion of the applicant's protective strategy.

Security Chain of Command and Delegation of Authority

Section 4.1.2 of the SCP, as well as the Responsibility Matrix portions of the SCP, details the chain of command and delegation of authority during contingency events. The PSP discusses the chain of command and delegation of authority during normal operations.

Physical Layout

The regulation at 10 CFR Part 73, Appendix C, Section II.B.3.b, requires, in part, that the SCP include a site map depicting the physical structures located on the site, including onsite independent spent fuel storage installations, and a description of the structures depicted on the map. Plans must also include a description and map of the site in relation to nearby towns, transportation routes (e.g., rail, water, and roads), pipelines, airports, hazardous material facilities, and pertinent environmental features that may affect the coordination of response activities. Descriptions and maps must indicate main and alternate entry routes for law enforcement or other offsite response and support agencies and the location for marshaling and coordinating response activities.

Section 4.2 of the SCP references Section 1.1 of the PSP for layouts of the OCA, PA, and vital areas; site maps; and descriptions of site features.

Safeguards Systems

The regulation at 10 CFR Part 73, Appendix C, Section II.B.3.c, requires, in part, that the SCP include a description of the physical security systems that support and influence the licensee's response to an event in accordance with the DBT described in 10 CFR 73.1(a). The description must begin with onsite physical protection measures implemented at the outermost perimeter and must move inward through those measures implemented to protect target set equipment.

Section 4.3 of the PSP states that safeguards systems are described in PSP Sections 9, 11, 12, 13, 15, and 16 and in facility implementing procedures/documents. Section 8 of the SCP describes how physical security systems will be used to respond to a threat at the site.

Law Enforcement Assistance

The regulation at 10 CFR Part 73, Appendix C, Section II.B.3.d, requires, in part, that the licensee provide a listing of available law enforcement agencies, a general description of their response capabilities and their criteria for response, and a discussion of working agreements or arrangements for communicating with these agencies.

Section 4.4 of the SCP details the role of LLEAs in the site protective strategy. Section 8 of the PSP and Section 5.6 of the SCP include additional details regarding LLEAs.

Policy Constraints and Assumptions

The regulation at 10 CFR Part 73, Appendix C, Section II.B.3.e, requires, in part, that the SCP contain a discussion of State laws, local ordinances, and company policies and practices that govern licensee response to incidents and must include, but is not limited to, the following: (i) use of deadly force, (ii) recall of off-duty employees, (iii) site jurisdictional boundaries, and (iv) use of enhanced weapons, if applicable.

Section 4.5 of the SCP details the site security policies, including the use of deadly force and authority to request offsite assistance.

Administrative and Logistical Considerations

The regulation at 10 CFR Part 73, Appendix C, Section II.B.3.f, requires, in part, that the licensee describe its practices that influence how the security organization responds to a safeguards contingency event. This description should include, but is not limited to, the procedures that will be used for ensuring that equipment needed to facilitate response will be readily accessible, in good working order, and in sufficient supply.

Section 4.6 of the SCP outlines the administrative duties of the Security Manager and Nuclear Security Captain, the facility procedures, and the administrative forms.

The NRC staff has reviewed TVA's description in SCP Sections 4, 4.1, 4.1.1, 4.1.2, and 4.2 through 4.6 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the SCP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the SCP meets the requirements of 10 CFR Part 73, Appendix C, Section II.B.3, and is, therefore, acceptable.

13.6.4.3.5 Response Capabilities

This section of the SSER discusses the NRC staff's review of the response by TVA to threats to the facility. TVA described how the onsite and offsite organizations protect WBN against the DBT, consistent with the regulations at 10 CFR 50.54(p)(1) and (hh); 10 CFR 73.55(k); and 10 CFR Part 73, Appendix B, Section VI, and Appendix C, Section II.B.3. In addition, the "Introduction," in Appendix C to 10 CFR Part 73, states, in part, that "it is important to note that a licensee's [SCP] is intended to be complementary to any emergency plans developed under appendix E to part 50 of this chapter…."

Response to Threats

Section 5.1 of the SCP states that the protective strategy is designed to defend the facility against all aspects of the DBT. Each organization has defined roles and responsibilities.

Armed Response Team

Section 5.2 of the SCP notes individuals from the Responsibility Matrix and their roles in the site protective strategy. This section also notes the minimum number of individuals and their contingency equipment for implementation of the protective strategy. TVA described the armed response team consistent with 10 CFR 73.55(k)(4), (5), (6), and (7); and 10 CFR Part 73, Appendix B, Section VI, and Appendix C, Section II.B.3.

Supplemental Security Officers

Section 5.3 of the SCP details the use of supplemental security officers in the site protective strategy. TVA described the use of supplemental security officers, consistent with the requirements in 10 CFR 73.55(k)(4).

Facility Operations Response

Section 5.4 of the SCP details the role of operations personnel in the site protective strategy, including responsibilities, strategies, and conditions for operator actions, as discussed in 10 CFR 50.54(hh).

Emergency Plan Response

Section 5.5 of the SCP notes the integration of the site emergency plan with the site's protective strategy and gives some examples of how the emergency plan can influence the protective strategy, as discussed in 10 CFR 73.55(b)(11).

Local Law Enforcement Agencies

Section 5.6 of the SCP meets the requirements of 10 CFR 73.55(k)(9) and 10 CFR Part 73, Appendix C, Section II.B.3.d, and lists the LLEAs that will respond to the site as a part of the protective strategy. Descriptions of the LLEA response capabilities appear in Section 8 of the PSP.

State Response Agencies

Section 5.7 of the SCP meets the requirements of 10 CFR 73.55(k)(9) and 10 CFR Part 73, Appendix C, Section II.B.3.d, and lists the State response agencies that will respond to the site as a part of the protective strategy.

Federal Response Agencies

Section 5.8 of the SCP meets the requirements of 10 CFR 73.55(k)(9) and 10 CFR Part 73, Appendix C, Section II.B.3.d, and lists the Federal response agencies that will respond to the site as a part of the protective strategy.

Response to Independent Spent Fuel Storage Installation Events

WBN does not have an independent spent fuel storage installation, so Section 5.9 does not apply.

The NRC staff has reviewed TVA's description in SCP Sections 5.0 through 5.9 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the SCP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the SCP meets the requirements of 10 CFR 50.54(p)(1), 10 CFR 50.54(hh), 10 CFR 73.55(k), Section VI of Appendix B to 10 CFR Part 73, and Section II.B.3 of Appendix C to 10 CFR Part 73, and is, therefore, acceptable.

13.6.4.3.6 Defense in Depth

Section 6 of the SCP lists site physical security characteristics, programs, and the strategy elements that illustrate the defense-in-depth nature of the site protective strategy, as required in 10 CFR 73.55(b)(3).

The NRC staff has reviewed TVA's description in SCP Section 6 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the SCP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the SCP meets the requirements of 10 CFR 73.55(b)(3) and is, therefore, acceptable.

13.6.4.3.7 Primary Security Functions

Section 7 of the SCP details the primary security functions of the site and their roles in the site protective strategy. It also notes the development of target sets and their function in the development of the site's protective strategy.

The NRC staff has reviewed TVA's description in SCP Section 7 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the SCP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the SCP meets the requirements of 10 CFR 73.55(b) and is, therefore, acceptable.

13.6.4.3.8 Protective Strategy

The regulation at 10 CFR Part 73, Appendix C, Section II.B.3.c(v), requires, in part, that licensees develop, implement, and maintain a written protective strategy that shall (1) be designed to meet the performance objectives of 10 CFR 73.55(a) through (k), (2) identify predetermined actions, areas of responsibilities, and timelines for the deployment of armed personnel, (3) contain measures that limit the exposure of security personnel to possible attack, (4) describe the physical security systems and measures that provide defense in depth, (5) describe the specific structure and responsibilities of the armed response organization, and (6) provide a command and control structure.

Section 8 of the SCP describes the site protective strategy.

The NRC staff has reviewed TVA's description in SCP Section 8 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because TVA's description in the SCP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description in the SCP meets the requirements of 10 CFR Part 73, Appendix C, Section II.B.3.c(v), and is, therefore, acceptable.

13.6.4.3.9 Conclusions on the Safeguards Contingency Plan

On the basis of the NRC staff's review described in Sections 13.6.4.3.1 through 13.6.4.3.8 of this SER, the SCP meets the requirements of 10 CFR Part 73, Appendix C. The NRC staff concludes that complete and procedurally correct implementation of the SCP will provide high assurance that activities involving special nuclear material are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety.

The target sets, site-specific target set analysis, and site protective strategy appear in the facility implementing procedures, which were not subject to NRC staff review as part of this application.

These are subject to future NRC inspection in accordance with 10 CFR 73.55(c)(7)(iv) and 10 CFR Part 73, Appendix C, Section II.B.5(iii).

13.6.5 Conclusions

The NRC staff's review of the WBN Unit 2 PSP, T&QP, and SCP, Revision 11, dated July 23, 2010, and TVA's letter, "Response to Request for Additional Information Regarding Target Set Development," dated November 18, 2010, focused on ensuring that these plans contain the programmatic elements necessary to provide high assurance that activities involving special nuclear material are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety.

Based on its review of the information provided by TVA, the NRC staff concludes that these plans include the necessary programmatic elements that, when effectively implemented, will provide the required high assurance demanded by the regulation. The burden to effectively implement these plans remains with TVA. Effective implementation depends on the procedures and practices that TVA develops to satisfy the programmatic elements of its PSP, T&QP, and SCP.

The target sets, site-specific target set analysis, and site protective strategy appear in the facility implementing procedures, which were not subject to NRC staff review as part of this application. These are subject to future NRC inspection in accordance with 10 CFR 73.55(c)(7)(iv) and 10 CFR Part 73, Appendix C, Section II.B.5(iii).

As required by Section 3 of TVA's PSP, TVA will implement a performance evaluation program that periodically tests and evaluates the effectiveness of the overall protective strategy. This program requires that deficiencies be corrected. In addition, NRC inspectors will conduct periodic force-on-force exercises that will test the effectiveness of TVA's protective strategy. Based on the results of TVA's own testing and evaluation and the NRC's baseline inspections and force-on-force exercises, enhancements to TVA's PSP, T&QP, and SCP may be required to ensure that the overall protective strategy can be effectively implemented. As such, the NRC staff approval of TVA's PSP, T&QP, and SCP is limited to the programmatic elements necessary to provide the required high assurance, as stated above. Should deficiencies be identified in the programmatic elements of these plans as a result of TVA's or the NRC's periodic drills or exercises that test the effectiveness of the overall protective strategy, TVA will correct the plans to address these deficiencies promptly and notify the NRC of these plan changes in accordance with the requirements of 10 CFR 50.54(p) or 10 CFR 50.90, "Application for Amendment of License, Construction Permit, or Early Site Permit."

TVA's security plan information is withheld from public disclosure in accordance with the provisions of 10 CFR 73.21.

17 QUALITY ASSURANCE

17.3 Quality Assurance Program

Background:

The Commission's regulatory requirements related to quality assurance (QA) programs are set forth in Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities." The pertinent requirements of Appendix B apply to all activities affecting the safety-related functions of the facility, including designing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying structures, systems, and components.

By letters dated March 30, September 13, November 3, and December 7, 1989, the Tennessee Valley Authority (TVA) submitted its corporate "Tennessee Valley Nuclear Quality Assurance Plan," TVA-NQA-PLN89-A, Revision 0. In Revision 23 of the plan, dated January 25, 2010, TVA subsequently changed the title of TVA-NQA-PLN89-A to "[TVA] Nuclear Quality Assurance Program." The document covers all of TVA's nuclear power plants, including those either operating or under construction. The U.S. Nuclear Regulatory Commission (NRC) reviewed TVA-NQA-PLN89-A, Revision 0, and documented its results in a letter dated January 18, 1990. Based on its review, the NRC concluded that TVA's QA program met the criteria of Appendix B to 10 CFR Part 50 and found the program acceptable for use in the design, construction, and operation of TVA's nuclear power plants. TVA has implemented this corporate QA program at Watts Bar Nuclear Plant (WBN) Unit 2 construction activities since the NRC staff accepted the program in 1990. TVA has revised TVA-NQA-PLN89-A 23 times since the NRC's initial approval in 1990.

In accordance with the regulation at 10 CFR 50.54(a)(3), a licensee "may make a change to a previously accepted quality assurance program description included or referenced in the Safety Analysis Report without prior NRC approval, provided the change does not reduce the commitments in the program description as accepted by the NRC." The regulation also provides examples of changes that do or do not reduce a licensee's commitments in the program description. The regulation at 10 CFR 50.54(a)(4) requires that "changes...that do reduce the commitments must be submitted to the NRC and receive NRC approval prior to implementation." Accordingly, by letter dated August 28, 2003 (ADAMS Accession No. ML032460719), as supplemented on January 16, 2004 (ADAMS Accession No. ML040210458), TVA submitted TVA-NQA-PLN89-A, Revision 13, in accordance with 10 CFR 50.54(a)(4) for NRC review.

By letter dated May 28, 2004 (ADAMS Accession No. ML041550313), the NRC staff concluded that Revision 13 to the QA program continued to satisfy the criteria of Appendix B to 10 CFR Part 50, and concluded, therefore, that it continued to be acceptable. This was the last NRC staff review of TVA's QA program before this safety evaluation. TVA has subsequently notified the NRC of nine additional revisions to the QA program. In these subsequent revisions, TVA stated per 10 CFR 50.54(a)(3) that there were no reductions in the commitments in the program description as accepted by the NRC.

Evaluation:

The following documents include the changes submitted by TVA subsequent to the NRC staff's review of TVA-NQA-PLN89-A, Revision 13, in 2004 and other relevant correspondence related to the TVA QA program:

- Letter from S. Black (NRC) to TVA, dated January 18, 1990 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML082320622; safety evaluation of QA program, Revision 0)

- Letter from M. Comar (NRC) to TVA, dated May 28, 2004 (ADAMS Accession No. ML041550313; safety evaluation of QA program, Revision 13)

- Letter from M. Burzynski (TVA) to NRC, dated August 31, 2004 (ADAMS Accession No. ML042470056; QA program, Revision 14).

- Letter from G. Morris (TVA) to NRC, dated August 31, 2005 (ADAMS Accession No. ML052450178; QA program, Revision 15)

- Letter from G. Morris (TVA) to NRC, dated August 31, 2006 (ADAMS Accession No. ML062480198; QA program, Revision 16)

- Letter from B. Wetzel (TVA) to NRC, dated August 31, 2007 (ADAMS Accession No. ML072490146; QA program, Revision 17)

- Letter from A. Bhatnager (TVA) to NRC, dated December 4, 2007 (ADAMS Accession No. ML073440041; QA program, Revision 18, which informed the NRC of TVA's intention to complete construction at WBN Unit 2)

- Letter from M. Purcell (TVA) to NRC, dated December 5, 2008 (ADAMS Accession No. ML083440224; QA program, Revision 19)

- Letter from M. Purcell (TVA) to NRC, dated March 13, 2009 (ADAMS Accession No. ML090760973; QA program, Revision 20, which addressed TVA activities at Bellefonte Nuclear Station Units 1 and 2 associated with TVA's viability assessment for completing construction of the units)

- Letter from R. Krich (TVA) to NRC, dated September 28, 2009 (ADAMS Accession No. ML092750350; QA program, Revision 21)

- Letter from R. Krich (TVA) to NRC, dated January 15, 2010 (ADAMS Accession No. ML100210972; QA program, Revisions 22 and 23)

TVA-NQA-PLN89-A applies to all of the TVA sites, including Bellefonte Nuclear Plant, Sequoya Nuclear Plant, Browns Ferry Nuclear Plant, and WBN Units 1 and 2. TVA included specific QA requirements for WBN Unit 2 in Appendix F, "Watts Bar Unit 2 Construction Completion Project," to TVA-NQA-PLN89-A. For this operating license application, the NRC staff reviewed the revisions listed above to TVA-NQA-PLN89-A that TVA has made in accordance with 10 CFR 50.54(a)(3), since the NRC staff's last safety evaluation of TVA's corporate nuclear QA plan in 2004, to determine if TVA made any reductions in commitment. The staff did not identify any unreviewed reductions in commitment made by TVA since the staff's previous review in 2004. Since the staff previously approved the TVA corporate nuclear QA plan in 2004, and there have been no unreviewed reductions in commitment since the staff's approval, the staff concluded that TVA's QA program is in compliance with applicable NRC regulations and is acceptable for the design, construction, and operation of WBN Unit 2.

17.3.1 Authorization to Transfer Code Symbol Stamp

By letter dated June 25, 2010 (ADAMS Accession No. ML101760222), TVA submitted a request to use alternatives to certain American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Section III requirements at WBN Unit 2. Specifically, under 10 CFR 50.55a(a)(3)(i), TVA asked to use the provisions of ASME Code Case N-520-3, "Alternative Rules for Renewal of Active or Expired N-Type Certificates for Plants Not in Active Construction, Section III, Division 1," on the basis that it provides an acceptable level of quality and safety. Code Case N-520-3 would allow TVA to request an extension of its temporary Certificate of Authorization from ASME to complete and transfer documentation of the partially completed ASME Section III systems and components to the jurisdiction of a subcontractor that is an ASME Section III N-Certificate holder. In a safety evaluation dated October 2, 2008, the NRC had previously authorized the use of the alternatives in ASME Code Case N-520-2. The use of that code case allowed TVA to obtain the temporary Certificate of Authorization.

The staff reviewed the information in TVA's letter dated June 25, 2010, and the previous safety evaluation that addressed Code Case N-520-2. The staff noted that the only distinction between the two code cases is that Code Case N-520-3 allows ASME, upon receipt of a request submitted by certified mail, to twice extend the term of temporary certificates for an additional period not to exceed 1 year each.

In its safety evaluation dated September 29, 2010 (ADAMS Accession No. ML102700527), the NRC staff concluded that Code Case N-520-3 provided an acceptable level of quality and safety for the requirements of Section NCA-8100, "Authorization to Perform Code Activities," of Section III of the ASME Code. Therefore, under 10 CFR 50.55a(a)(3)(i), the staff authorized the use of the proposed alternative for WBN Unit 2.

18 Control Room Design Review

18.1 General

As part of the U.S. Nuclear Regulatory Commission's (NRC's) task actions following the Three Mile Island Unit 2 (TMI-2) accident (Item I.D.1, NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," issued May 1980, and NUREG-0737, "Clarification of TMI Action Plan Requirements," issued November 1980) the NRC staff required all licensees and applicants for operating licenses to conduct a detailed control room design review (DCRDR) to identify and correct human engineering discrepancies (HEDs).

TVA performed a preliminary design review of the Watts Bar Nuclear Plant (WBN) control room and submitted its findings to the NRC staff in a report dated January 13, 1981 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML073521188). In Appendix D to the safety evaluation report (SER) for WBN (NUREG-0847, June 1982), the NRC staff reviewed the applicant's submittals, the control room review, and other clarifying information. The staff determined that the proposed corrective actions to the HEDs specified in Appendix D to the SER identified and reported by the applicant were acceptable. Although the staff review identified HEDs, the staff found that, overall, the control room was designed to promote effective and efficient operator actions. Subsequently, TVA identified the DCRDR as a special program in the TVA Corporate Nuclear Performance Plan, which was evaluated, in part, by the NRC staff as documented in NUREG-1232, Volume 4, "[SER] on the Watts Bar Nuclear Performance Plan," dated December 28, 1989. The NRC staff documented its further analysis in Supplemental Safety Evaluation Report (SSER) 6, April 1991. The staff conducted a series of evaluation activities leading up to the completion of SSER 6, including a pre-implementation audit at WBN from November 11-18, 1988, that was performed in accordance with NUREG-0700, "Guidelines for Control Room Design Reviews," dated September 1981. By letter dated April 28, 1989 (ADAMS Accession No. ML073541023), the NRC staff forwarded to TVA its safety evaluation of the DCRDR, in which the staff concluded that:

> the DCRDR activities for [WBN Units 1 and 2] will meet the requirements of NUREG-0737, Supplement 1, when TVA provides NRC with a supplemental DCRDR summary report which adequately addresses the concerns described in the following...
>
> 2.2 System Function and Task Analyses to Identify Control Room Operators Tasks and Information and Control Requirements During Emergency Operations
>
> 2.3 Comparison of Display and Control Requirements with a Control Room Inventory
>
> 2.5 Assessment of HEDs to Determine Which Are Significant and Should be Corrected
>
> 2.7 Verification that Selected Design Improvements Will Provide the Necessary Correction
>
> 2.8 Verification that Selected Design Improvements Will Not Introduce New HEDs.

On March 28, 1990 (ADAMS Accession No. ML073541180), TVA submitted a supplemental DCRDR summary report, which stated that the open items for the DCRDR HEDs were completed. The NRC staff conducted a DCRDR audit from August 21–23, 1990. By letter dated October 24, 1990 (ADAMS Accession No. ML073550207), TVA submitted a list and schedule for correcting all remaining Category 1 and 2 HEDs and reaffirmed its commitment to correct all Unit 1 HEDs before fuel load. The NRC staff concluded in SSER 6 that the DCRDR program implemented at WBN Unit 1 satisfies the DCRDR programmatic requirements of NUREG-0737, Supplement 1, dated January 1983. The staff also concluded that the proposed corrective actions to the HEDs identified and reported by TVA were acceptable.

By letters dated January 29, 2008, and September 26, 2008 (ADAMS Accession Nos. ML080320443 and ML082750019, respectively), TVA provided its regulatory framework for the completion of construction and licensing activities for WBN Unit 2 corrective action, special programs, and unresolved safety issues. TVA proposed to use the same approach to resolve the DCRDR special program issue for WBN Unit 2 as it used for WBN Unit 1. In its letter to TVA dated February 11, 2009 (ADAMS Accession No. ML090210107), the NRC staff stated that, "based on the TVA description and the staff's review documented in NUREG-1232, Volume 4 and the applicable supplements of NUREG-0847, there is reasonable assurance that, when implemented as described, this issue will be appropriately resolved for WBN Unit 2. Use of the Unit 1 approach is acceptable."

In SSER 21, dated February 2009, the NRC staff stated that it had "reviewed the information provided by TVA and concluded that, based on the TVA description and the staff's review (documented in NUREG-1232, Volume 4, and the applicable supplements of NUREG-0847), there is reasonable assurance that, when implemented as described, certain [special program] issues can be designated as acceptable for implementation at WBN Unit 2." In SSER 21, Section 1.13.2, the staff identified the DCRDR as a resolved special program issue. The NRC staff also reviewed WBN Unit 2 Final Safety Analysis Report Amendment 99, dated May 27, 2010 (ADAMS Accession No. ML101610290), and determined that there were no changes to the TVA DCRDR special program.

18.2 Conclusions

Since the NRC staff has approved the DCRDR special program approach for WBN Unit 1, and TVA proposed to use the same approach for WBN Unit 2, there is reasonable assurance that, when implemented as described by TVA, the DCRDR TMI task action (Item I.D.1 of NUREG-0660 and NUREG-0737) will be appropriately resolved for WBN Unit 2.

21 FINANCIAL QUALIFICATIONS

21.1 Tennessee Valley Authority Financial Qualifications for Watts Bar Nuclear Plant Unit 2

The regulation in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.33(f) requires that, except for an electric utility, each applicant for a license provide "information sufficient to demonstrate to the Commission the financial qualification of the applicant to carry out...the activities for which the permit or license is sought." In its final rule dated March 31, 1982 (47 FR 13750), the U.S. Nuclear Regulatory Commission (NRC) amended its regulations in 10 CFR Part 2, "Rules of Practice for Domestic Licensing Proceedings and Issuance of Orders," and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to eliminate entirely the requirements for financial qualification review and findings for electric utilities that are applying for construction permits or operating licenses for production or utilization facilities.

The regulations at 10 CFR 50.2 define an electric utility as

> any entity that generates or distributes electricity and which recovers the cost of this electricity, either directly or indirectly, through rates established by the entity itself or by a separate regulatory authority. Investor owned utilities, including generation or distribution subsidiaries, public utility districts, municipalities, rural electric cooperatives, and State and Federal agencies, including associations of the foregoing, are included within the meaning of "electric utility."

The Tennessee Valley Authority (TVA), a corporation owned by the U.S. government, provides electricity for 9 million people in parts of seven southeastern states. TVA is self-regulated, and the TVA Act of 1933, as amended, gives the TVA Board of Directors the sole responsibility for establishing the rates that TVA charges for electric power. These rates are not subject to review or approval by any State or Federal regulatory body. Based on the foregoing, the NRC staff concludes that TVA is an electric utility as defined by 10 CFR 50.2. Therefore, in accordance with NRC regulations, the NRC staff will not conduct a review of TVA's financial qualifications.

21.2 Foreign Ownership, Control, or Domination

In its application, TVA has provided the names and addresses of the directors and officers of TVA and stated that all are U.S. citizens. According to the applicant, TVA is not owned, controlled, or dominated by any alien, foreign corporation, or foreign government. The NRC staff has reviewed the information provided by TVA and finds it to be acceptable.

22 FINANCIAL PROTECTION AND INDEMNITY REQUIREMENTS

22.3 Operating Licenses

Before the issuance of an operating license under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," the Tennessee Valley Authority (TVA) is required to provide satisfactory documentation that it has obtained the financial protection required by 10 CFR 140.11(a)(4), and not less than the amount required by 10 CFR 50.54(w) with respect to insurance from private sources or an equivalent amount of protection covering the licensee's obligation. This is Open Item 25 (Appendix HH) until TVA provides the necessary documentation and the U.S. Nuclear Regulatory Commission staff has reviewed and approved it.

25 QUALITY OF CONSTRUCTION, OPERATIONAL READINESS, AND QUALITY ASSURANCE EFFECTIVENESS

25.9 Program for Maintenance and Preservation of the Licensing Basis for Units 1 and 2

Background:

On July 25, 2007, the U.S. Nuclear Regulatory Commission (NRC) issued a Staff Requirements Memorandum (SRM) based on SECY 07-0096, "Possible Reactivation of Construction and Licensing Activities for the Watts Bar Nuclear Plant Unit 2" (ADAMS Accession No. ML072060688). The SRM directed, in part, the NRC staff to use the current licensing basis for WBN Unit 1 as the reference basis for the review and licensing of WBN Unit 2.

By letter dated August 3, 2007, the Tennessee Valley Authority (TVA) informed the NRC of its intent to complete construction and licensing of WBN Unit 2. In this letter, TVA described its intent to align the licensing and design bases for WBN Units 1 and 2 to the fullest practical extent. The NRC staff examined the information provided in TVA's August 3, 2007, TVA letter, along with previously completed NRC safety evaluations, in order to reconstitute the status of remaining licensing issues.

By letter dated May 8, 2008, the NRC staff described the methodology it was using to identify the overall review scope for the WBN Unit 2 operating license review, and for screening previously completed reviews to determine if they remained valid and relevant. The NRC staff's initial effort was to define the complete scope of topics that must be reviewed to complete the WBN Unit 2 licensing review. The Table of Contents of NUREG-0847 defines the majority of topics and provides the basic organizational structure used during this review effort. Topics previously identified and resolved by the TVA Nuclear Performance Plan (NPP; original version dated November 1, 1985) were addressed and documented in NUREG-1232, Volume 4, issued December 28, 1989. Additional topics, such as the Maintenance Rule, were identified by review of the Table of Contents of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants." In addition to remaining Unit 2 NPP items, issues identified in generic communications, action items from NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980, and other regulatory topics were included. In general, the staff divided its review into the following categories:

- Inventory and evaluation of previous review topics
- Effect of WBN design and licensing basis changes
- Generic issue resolution before plant operation
- Potential degradation of SSCs
- Applicability of new regulations

By letters dated January 29, March 13, and June 16, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML080320443, ML080770237, and ML081700293, respectively), TVA submitted its regulatory framework for the completion of construction and licensing activities for WBN Unit 2. The NRC staff reviewed TVA's submittals and performed its own independent assessment. The staff documented the detailed results of its overall assessment of the regulatory framework in a letter to TVA dated October 10, 2008 (ADAMS Accession No. ML082840361). The staff also summarized its assessment in Supplemental Safety Evaluation Report (SSER) 21, dated February 2009, Section 1.7, "Summary of Outstanding Issues," and identified the relevant document in which the issue was last addressed. The list of topics in SSER 21, Section 1.7, is maintained by the NRC staff in its entirety and updated as the staff completes its review of the topics and accepts their resolution. In its future submittals or amendments to the WBN final safety analysis report, TVA should address the topics identified as open.

The NRC staff recognized that there might be circumstances that could result in the need to reopen a safety evaluation report topic that it had previously considered resolved. Even if a topic was previously reviewed and resolved in NUREG-0847, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2," and its supplements, there might be circumstances that would lead to the revision or re-performance of a previous evaluation. In this regard, the NRC staff recognized that changes to the WBN Unit 1 design and licensing basis made after the issuance of the operating license, which will be implemented in Unit 2, would need to be assessed to determine any effect on previous safety evaluations prepared for WBN Unit 2. These changes include items approved by amendments under Title 10 of the *Code of Federal Regulations* (10 CFR) 50.90, "Application for Amendment of License, Construction Permit, or Early Site Permit," or changes implemented in accordance with 10 CFR 50.59, "Changes, Tests and Experiments." As a result, TVA must review any changes and plant modifications like these to determine whether or not any conclusions made in NUREG-0847 or its supplements are affected. Consistent with the Commission's staff requirements memorandum dated July 25, 2007 (ADAMS Accession No. ML072060688; SECY-07-0096, "Possible Reactivation of Construction and Licensing Activities for the Watts Bar Nuclear Plant Unit 2"), TVA's process must also assess the effect of any exemptions, reliefs, and other actions that were specifically granted for Unit 1 to determine if the same allowance is appropriate for Unit 2. Therefore, the NRC staff required TVA to develop and implement a licensing-basis preservation program to confirm the validity of previously completed NRC safety evaluations to the final WBN Unit 2 design, or to identify topics that require supplemental review.

Evaluation:

In its submittal dated December 9, 2008 (ADAMS Accession No. ML083460177), TVA provided its licensing-basis preservation program. Specifically, TVA submitted New Generation Development and Construction (NGDC) Procedure PP-20, "Unit 2 Licensing Basis Preservation," which described the methods or processes for determining potential changes that could affect the Unit 2 licensing basis during the ongoing design and construction activities.

From the time that the WBN Unit 1 operating license was granted, FSAR changes have been made in accordance with 10 CFR 50.71(e) to reflect modifications installed in the plant. TVA performed an initial review of those WBN Unit 1 FSAR changes to identify and categorize significant changes that will eventually become a part of the combined Unit 1 and Unit 2 FSAR at Unit 2 fuel load. In addition, TVA submitted the 1995 version of the FSAR as of Amendment 91 on July 2, 2008. This version was accurate when Unit 1 received its operating license in 1995 and will serve as the Unit 2 baseline document.

In general, TVA categorized its licensing basis preservation into the following categories:

- Effect of WBN Unit 1 design and licensing basis changes
- Unit 1 configurations to be applied to Unit 2
- Unit 1 specific exemptions, reliefs, and other actions
- Generic issue resolution before plant operation
- Potential degradation of SSCs

- Applicability of new regulations

The NRC staff reviewed TVA's submittal and found that it did not provide sufficient detail to conclude that the specified actions were sufficient to address the purpose of the program. By letter dated April 28, 2009 (ADAMS Accession No. ML090990533), the staff asked TVA to provide a more detailed summary of the process steps used to implement each of the screening questions, any corresponding evaluation criteria, and documentation of the results, as well as TVA's plans for assessing the performance of the overall process.

By letter dated September 8, 2009 (ADAMS Accession No. ML092570391), TVA provided additional information and details about the process steps in its Unit 2 licensing-basis preservation program. Enclosure 1 to TVA's letter provided excerpts from Bechtel Corporation (TVA's contractor) Procedure 25402-3DP-G04G-00081, Revision 002, "Engineering Document Construction Release" (EDCR), which addressed how each Unit 2 design change would be evaluated for licensing-basis impact. Specifically, in EDCR Attachment J, "Licensing Basis Preservation Review Form," TVA identified the reviews to be performed for each EDCR issue to ensure that the Unit 2 licensing basis is implemented in the same manner as the corresponding Unit 1 licensing basis to obtain the same conclusion, to the maximum extent practical, as identified in NGDC PP-20. Because some Unit 2 structures, systems, and components are shared with Unit 1, the Attachment J reviews also include the construction activities for multi-unit operation so as not to invalidate previous WBN Unit 1 licensing-basis conclusions in NGDC PP-20. TVA also stated that the process in Appendix J addresses the corrective action program plans, special programs, generic communications, exemptions, reliefs, and other actions identified in NGDC PP-20.

The NRC staff reviewed TVA's program to preserve the licensing basis for WBN Units 1 and 2 in accordance with SRM-SECY-07-0096 and using the assessment methodology documented in the staff's letter to TVA dated May 8, 2008. The staff concludes that TVA's program for maintenance and preservation of the licensing basis for WBN, if properly implemented, provides reasonable assurance that any effects on previously reviewed and resolved safety evaluation report topics will be evaluated for WBN Unit 2. TVA's implementation of NGDC PP-20 and EDCR Appendix J will be audited or inspected by the NRC. This is Open Item 12 (Appendix HH).

APPENDIX A

CHRONOLOGY OF RADIOLOGICAL REVIEW OF WATTS BAR NUCLEAR PLANT, UNIT 2, OPERATING LICENSE REVIEW

In Supplement No. 20 to this safety evaluation report (SER), the staff of the U.S. Nuclear Regulatory Commission (NRC) provided a summary of the more significant correspondence exchanged between the NRC and the TVA (TVA or the applicant) during the review of the operating license application for Watts Bar Nuclear (WBN) Plant, Units 1 and 2. The NRC has made these documents available through its Agencywide Documents Access and Management System (ADAMS) or the Public Document Room.

The following list of documents presents the correspondence that has occurred subsequent to the applicant's letter notifying the NRC of its decision to reactivate construction of WBN Unit 2, which had been in a deferred status under the Commission's Policy Statement on Deferred Plants. This appendix contains most of the documents that are referenced in this supplement. However, it is not a complete listing of all correspondence. The reader may obtain the complete list, along with these documents, through ADAMS.

<u>NRC Letters and Summaries</u>

October 22, 2007	Letter, J.E. Dyer to W.R. McCollum (TVA), regarding the NRC staff action related to resumption of plant construction and continuation of operating license application reviews. (ADAMS Accession No. ML072570319)
October 23, 2007	Letter, J. Williams to W. Campbell (TVA), requesting information the NRC staff will need for its own assessment of the status of the licensing review. (ADAMS Accession No. ML072840337)
May 8, 2008	Letter, L. Raghavan to A. Bhatnagar (TVA), providing the NRCs initial assessment of the remaining operating license review scope for WBN Unit 2. (ADAMS Accession No. ML080860026)
June 3, 2008	Letter, J. Williams to A. Bhatnagar (TVA), requesting supplemental information to support acceptance review for the Supplemental Environmental Statement. (ADAMS Accession No. ML081210270)
May 28, 2008	Letter, P. Milano to A. Bhatnagar (TVA), providing results of the staff's assessment of all generic communications items. (ADAMS Accession No. ML081490110)
June 20, 2008	Letter, J. Williams to A. Bhatnagar (TVA), forwarding environmental assessment and finding of no significant impact related to request for extension of latest construction completion date. (ADAMS Accession No. ML081500030)
July 7, 2008	Letter, J. Williams to A. Bhatnagar (TVA), forwarding Order extending construction completion date of construction permit. (ADAMS Accession No. ML081500115)
August 25, 2008	Letter, P. Milano to A. Bhatnagar (TVA), providing staff's assessment of status of TVA responses to the NRC generic communications issued before 1995. (ADAMS Accession No. ML082330421)
October 2, 2008	Letter, L. Raghavan to A. Bhatnagar (TVA), forwarding relief request authorizing use of ASME Code Case N-520. (ADAMS Accession No. ML082560373)

October 10, 2008	Letter, P. Milano to A. Bhatnagar (TVA), providing staff's assessment of status of regulatory framework for completion of construction and licensing activities. (ADAMS Accession No. ML082840361)
February 11, 2009	Letter, P. Milano to A. Bhatnagar (TVA), providing staff's assessment of regulatory framework for completion of corrective action and special programs. (ADAMS Accession No. ML090210107)
April 28, 2009	Letter, P. Milano to A. Bhatnagar (TVA), Request For Additional Information On Programs For Licensing Basis Preservation and Construction Refurbishment (TAC NO. MD6581). (ADAMS Accession No. ML090990533)
July 1, 2010	Letter, P. Milano to A. Bhatnagar -WATTS BAR NUCLEAR PLANT, UNIT 2 -REVISED NOTICE OF RECEIPT OF UPDATE ANTITRUST INFORMATION AND OPPORTUNITY FOR PUBLIC COMMENT (TAC NO. ME0853). (ADAMS Accession No. ML1018204780)
July 2, 2010	Letter, P. Milano to A. Bhatnager WATTS BAR NUCLEAR PLANT, UNIT 2 -PROGRAM FOR CONSTRUCTION REFURBISHMENT (TAC NO. ME1708). (ADAMS Accession No. ML101720050)
July 2, 2010	Letter, J. Wiebe to A. Bhatnagar WATTS BAR NUCLEAR PLANT, UNIT 2 -REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSEE'S FINAL SAFETY ANALYSIS REPORT AMENDMENT RELATED TO MECHANICAL AND CIVIL ENGINEERING SYSTEMS (TAC NO. ME2731). (ADAMS Accession No. ML101530474)
July 7, 2010	Letter, J. Wiebe to A. Bhatnagar WATTS BAR NUCLEAR PLANT, UNIT 2 -REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSEE'S FINAL SAFETY ANALYSIS REPORT AMENDMENT RELATED TO SPENT FUEL STORAGE AND HANDLING REVIEW (TAC NO. ME2734). (ADAMS Accession No. ML101620047)
July 8, 2010	Letter, J. Wiebe to A. Bhatnagar WATTS BAR NUCLEAR PLANT, UNIT 2 -REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSEE'S FINAL SAFETY ANALYSIS REPORT AMENDMENT RELATED TO QUALITY AND VENDOR BRANCH REVIEW (TAC NO. ME2731). (ADAMS Accession No. ML101600026)
July 12, 2010	Letter, J. Wiebe to A. Bhatnagar WATTS BAR NUCLEAR PLANT, UNIT 2 -REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSEE'S FINAL SAFETY ANALYSIS REPORT AMENDMENT RELATED TO ELECTRICAL ENGINEERING SYSTEMS (TAC NO. ME2731). (ADAMS Accession No. ML101530354)
July 12, 2010	Letter, J. Wiebe to A. Bhatnagar WATTS BAR NUCLEAR PLANT, UNIT 2 -REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSEE'S FINAL SAFETY ANALYSIS REPORT AMENDMENT RELATED TO ACCIDENT DOSE (TAC NO. ME3091). (ADAMS Accession No. ML101600278)
July 13, 2010	Letter, J. Poole to A. Bhatnagar WATTS BAR NUCLEAR PLANT, UNIT 2 -REQUEST FOR ADDITIONAL INFORMATION REGARDING ANTITRUST REVIEW (TAC NO. ME0853). (ADAMS Accession No. ML101890406)

July 15, 2010	Letter, J. Wiebe to A. Bhatnagar WATTS BAR NUCLEAR PLANT, UNIT 2 -REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSEE'S FINAL SAFETY ANALYSIS REPORT AMENDMENT RELATED TO CHAPTER 8, "ELECTRIC POWER" (TAC NO. ME2731). (ADAMS Accession No. ML101800156)
July 29, 2010	Letter, J. Wiebe to A. Bhatnagar WATTS BAR NUCLEAR PLANT, UNIT 2 -REQUEST FOR ADDITIONAL INFORMATION REGARDING PRESSURE LOCKING AND THERMAL BINDING OF SAFETY-RELATED POWER-OPERATED GATE VALVES (TAC NO. MD6717). (ADAMS Accession No. ML101960502)
July 29, 2010	Letter, J. Lamb to A. Bhatnagar WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2 -REQUEST FOR WITHHOLDING INFORMATION FROM PUBLIC DISCLOSURE REGARDING WATTS BAR NUCLEAR PLANT CYBER SECURITY PLAN AND CYBER SECURITY PLAN IMPLEMENTATION SCHEDULE (TAC NOS. ME2677 AND ME2679). (ADAMS Accession No. ML102090019)
July 30, 2010	Letter, J. Poole to A. Bhatnagar WATTS BAR NUCLEAR PLANT, UNIT 2 – REQUEST FOR ADDITIONAL INFORMATION REGARDING FINAL SAFETY ANALYSIS REPORT AMENDMENT RELATED TO CONDENSATE AND FEEDWATER (TAC NO. ME3091). (ADAMS Accession No. ML101940474)
July 30, 2010	Letter, J. Poole to A. Bhatnagar WATTS BAR NUCLEAR PLANT, UNIT 2 – REQUEST FOR ADDITIONAL INFORMATION REGARDING FINAL SAFETY ANALYSIS REPORT AMENDMENT RELATED TO MECHANICAL ENGINEERING SYSTEMS (TAC NO. ME3091). (ADAMS Accession No. ML101950616)
July 30, 2010	Letter, J. Poole to A. Bhatnagar WATTS BAR NUCLEAR PLANT, UNIT 2 -REQUEST FOR ADDITIONAL INFORMATION REGARDING FINAL SAFETY ANALYSIS REPORT AMENDMENT RELATED SECTION 3.11 (TAC NO. ME2731). (ADAMS Accession No. ML102020281)
August 5, 2010	Letter, R. Haag to A. Bhatnagar WATTS BAR NUCLEAR PLANT UNIT 2 CONSTRUCTION - NRC INTEGRATED INSPECTION REPORT 05000391/2010603 AND NOTICE OF VIOLATION. (ADAMS Accession No. ML102170465)
August 9, 2010	Letter, J. Poole to A. Bhatnagar WATTS BAR NUCLEAR PLANT, UNIT 2 – REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSEE'S FINAL SAFETY ANALYSIS REPORT AMENDMENT RELATED TO ELECTRICAL ENGINEERING SYSTEMS (TAC NO. ME2731). (ADAMS Accession No. ML101870682
August 12, 2010	Letter, S. Campbell to A. Bhatnagar WATTS BAR NUCLEAR PLANT, UNIT 2 – SAFETY EVALUATION REGARDING GENERIC LETTER 1995-07, "PRESSURE LOCKING AND THERMAL BINDING OF SAFETY-RELATED POWER-OPERATED GATE VALVES" (TAC NO. MD6717). (ADAMS Accession No. ML101190443)
August 13, 2010	Letter, R. Haag to A. Bhatnagar WATTS BAR NUCLEAR PLANT UNIT 2 CONSTRUCTION - NRC INSPECTION REPORT 05000391/2010606. (ADAMS Accession No. ML102280522)

August 19, 2010	Letter, J. Poole to A. Bhatnagar WATTS BAR NUCLEAR PLANT, UNIT 2 – REQUEST FOR ADDITIONAL INFORMATION REGARDING FINAL SAFETY ANALYSIS REPORT AMENDMENT RELATED TO SECTION 11 (TAC NO. ME3945). (ADAMS Accession No. ML102240513)
August 25, 2010	Letter, S. Bailey to A. Bhatnagar WATTS BAR NUCLEAR PLANT, UNIT 2 – REQUEST FOR ADDITIONAL INFORMATION REGARDING FINAL SAFETY ANALYSIS REPORT RELATED TO INSTRUMENTATION AND CONTROLS (TAC NO. ME2371). (ADAMS Accession No. ML102360705)
August 27, 2010	Letter, J. Wiebe to A. Bhatnagar WATTS BAR NUCLEAR PLANT, UNIT 2 -REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSEE'S FINAL SAFETY ANALYSIS REPORT AMENDMENT RELATED TO CHAPTERS 11, "RADIOACTIVE WASTE MANAGEMENT" AND 12, "RADIATION PROTECTION" (TAC NO. ME2731). (ADAMS Accession No. ML102360201)
September 7, 2010	Letter, J. Poole to A. Bhatnagar WATTS BAR NUCLEAR PLANT, UNIT 2 – REQUEST FOR ADDITIONAL INFORMATION REGARDING SECTION 3.5 OF THE PRESERVICE INSPECTION PROGRAM PLAN (TAC NO. ME4074). (ADAMS Accession No. ML102420625)
September 7, 2010	Letter, J. Poole to A. Bhatnagar WATTS BAR NUCLEAR PLANT, UNIT 2 – REQUEST FOR ADDITIONAL INFORMATION FROM THE VESSELS INTERNALS INTEGRITY BRANCH REGARDING PRESERVICE INSPECTION PROGRAM PLAN (TAC NO. ME3113). (ADAMS Accession No. ML102430175)
September 8, 2010	Letter, J. Wiebe to A. Bhatnagar WATTS BAR NUCLEAR PLANT, UNIT 2 – REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSEE'S FINAL SAFETY ANALYSIS REPORT AMENDMENT RELATED TO CHAPTER 14, "INITIAL TEST PROGRAM" (TAC NO. ME2731). (ADAMS Accession No. ML102380431)
September 16, 2010	Letter, P. Milano to A. Bhatnagar WATTS BAR NUCLEAR PLANT, UNIT 2 – REQUEST FOR ADDITIONAL INFORMATION REGARDING FINAL SAFETY ANALYSIS REPORT CHAPTERS 4, 5, 6 and 9 (TAC NO. ME4074). (ADAMS Accession No. ML102530464)
September 17, 2010	Letter, J. Poole to A. Bhatnagar WATTS BAR NUCLEAR PLANT, UNIT 2 – REQUEST FOR ADDITIONAL INFORMATION REGARDING FINAL SAFETY ANALYSIS REPORT AMENDMENT RELATED TO SECTION 9.2 (TAC NO. ME4074). (ADAMS Accession No. Ml102510313)
September 20, 2010	Letter, P. Milano to A. Bhatnagar WATTS BAR NUCLEAR PLANT, UNIT 2 – REQUEST FOR ADDITIONAL INFORMATION REGARDING FINAL SAFETY ANALYSIS REPORT RELATED TO SECTION 15 (TAC NO. ME4074). (ADAMS Accession No. ML102590244)
September 28, 2010	Letter, J. Poole to A. Bhatnagar WATTS BAR NUCLEAR PLANT, UNIT 2 – REQUEST FOR ADDITIONAL INFORMATION FROM THE COMPONENT PERFORMANCE AND TESTING BRANCH REGARDING PRESERVICE INSPECTION PROGRAM PLAN (TAC NO. ME3113). (ADAMS Accession No. ML102630535)

September 28, 2010	Letter, J. Poole to A. Bhatnagar WATTS BAR NUCLEAR PLANT, UNIT 2 – REQUEST FOR ADDITIONAL INFORMATION REGARDING FINAL SAFETY ANALYSIS REPORT AMENDMENT RELATED TO SECTIONS 3.2 AND 5.2 (TAC NOS. ME2731 AND ME3091). (ADAMS Accession No. ML102630598)
September 28, 2010	Letter, J. Poole to A. Bhatnagar WATTS BAR NUCLEAR PLANT, UNIT 2 – REQUEST FOR ADDITIONAL INFORMATION REGARDING FINAL SAFETY ANALYSIS REPORT AMENDMENT RELATED TO SECTION 7.3 (TAC NO. ME2731). (ADAMS Accession No. ML102640324)
September 29, 2010	Letter, S. Campbell to A. Bhatnagar WATTS BAR NUCLEAR PLANT, UNIT 2 -REQUEST FOR RELIEF REGARDING ALTERNATIVE RULES FOR RENEWAL OF ACTIVE OR EXPIRED N-TYPE CERTIFICATES (TAC NO. MD8314). (ADAMS Accession No. ML102700527)
October 4, 2010	Letter, J. Poole to A. Bhatnagar WATTS BAR NUCLEAR PLANT, UNIT 2 -REQUEST FOR ADDITIONAL INFORMATION REGARDING FINAL SAFETY ANALYSIS REPORT RELATED TO SECTION 15 (TAC NO. ME4074). (ADAMS Accession No. ML102700437)
October 20, 2010	Letter, J. Poole to A. Bhatnagar WATTS BAR NUCLEAR PLANT, UNIT 2 -REQUEST FOR ADDITIONAL INFORMATION REGARDING FINAL SAFETY ANALYSIS REPORT AMENDMENT RELATED TO SECTION 9.2.1 (TAC NO. ME4074). (ADAMS Accession No. ML102880442)
November 12, 2010	Letter, J. Wiebe to A. Bhatnagar WATTS BAR NUCLEAR PLANT, UNIT 2 – REQUEST FOR ADDITIONAL INFORMATION REGARDING INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS DESIGN REPORT (TAC NO. ME3334). (ADAMS Accession No. ML102990030)

TVA Letters

August 3, 2007	Letter, W.R. McCollum, Jr., informing NRC staff of TVA's intention to reactivate and complete construction activities at WBN Unit 2. (ADAMS Accession No. ML072190047)
September 7, 2007	Letter, M. Bajestani to NRC, providing response to certain NRC bulletins and generic letters for Watts Bar Unit 2. (ADAMS Accession No. ML072570676)
September 7, 2007	Letter, M. Bajestani to NRC, providing a summary of generic communications with supplemental information on references to closure documentation where applicable. (ADAMS Accession No. ML072570678)
October 11, 2007	Letter, M. Bajestani to NRC, informing the NRC that TVA does not see the need for additional exemptions or relief, other than those discussed in the attachment to this letter and those associated with preservice and inservice inspection programs. (ADAMS Accession No. ML072910331)
December 5, 2007	Letter, M. Bajestani to NRC, notifying NRC that TVA will use the Westinghouse Eagle-21 process protection system on WBN Unit 2. (ADAMS Accession No. ML073440022)

January 29, 2008	Letter, M. Bajestani to NRC, describing the regulatory framework for the completion of construction and licensing activities for WBN Unit 2 as committed to in TVA's August 3, 2007, letter. (ADAMS Accession No. ML080320443)
February 1, 2008	Letter, M. Bajestani to NRC, requesting authorization to use ASME Code Case N-520-2 for expired N-type Certificates. (ADAMS Accession No. ML080370185)
February 15, 2008	Letter, M. Bajestani to NRC, providing Final Supplemental Environmental Impact Statement for completion and operation of WBN Unit 2, as committed to in TVA's January 29, 2008, letter. (ADAMS Accession No. ML080510469)
March 6, 2008	Letter, M. Bajestani to NRC, requests further extension of the construction permit to March 31, 2013. (ADAMS Accession No. ML080710489)
March 13, 2008	Letter, M. Bajestani to NRC, supplementing the January 29, 2008, letter on the regulatory framework tables for the SER and its supplements to allow ease of review and provide additional definitions for section status. (ADAMS Accession No. ML080770237)
March 20, 2008	Letter, M. Bajestani to NRC, providing a comprehensive list of generic communications for WBN Unit 2, reviewed and categorized by status. (ADAMS Accession No. ML080850253)
May 8, 2008	Letter, M. Bajestani to NRC, providing additional information and supersedes in its entirety TVA's March 6, 2008, request to extend the expiration date for the WBN Unit 2 Construction Permit. (ADAMS Accession No. ML081340309)
May 29, 2008	Letter, M. Bajestani to NRC, providing proposed program methods to resolve those sub-issues of the cable issues corrective action program that are different from those used for WBN Unit 1. (ADAMS Accession No ML081560183)
June 16, 2008	Letter, M. Bajestani to NRC, submitting additional information on regulatory framework for completion of construction and licensing activities. (ADAMS Accession No. ML081700293)
September 26, 2008	Letter, M. Bajestani to NRC, submitting regulatory framework for corrective action programs and special programs. (ADAMS Accession No. ML082750019)
October 30, 2008	Letter, M. Bajestani to NRC, submitting preservice inspection program plan. (ADAMS Accession No. ML083090046)
December 9, 2008	Letter, M. Bajestani to NRC, submits licensing basis preservation and construction refurbishment program plan. (ADAMS Accession No. ML083460177)
January 14, 2009	Letter, M. Bajestani to NRC, providing additional information regarding the cable issues corrective action program. (ADAMS Accession No. ML090210473)
March 4, 2009	Letter, M. Bajestani to NRC, OL application update. (ADAMS Accession No. ML090700378)
April 6, 2009	Letter, M. Bajestani to NRC, providing additional information regarding the cable issues corrective action program. (ADAMS Accession No. ML091120183)

May 18, 2009	Letter, M. Bajestani to NRC, providing additional information regarding the cable issues corrective action program. (ADAMS Accession No. ML091410284)
April 30, 2009	Letter, M. Bajestani to NRC, Watts Bar Nuclear Plant (WBN) - Unit 2 - Final Safety Analysis Report (FSAR), Amendment 93. (ADAMS Accession No. ML091400067)
May 18, 2009	Letter, M. Bajestani to NRC Watts Bar Nuclear Plant (WBN) - Unit 2 - Additional Information Regarding WBN Unit 2 Corrective Action Programs (TAC No. MD9182). (ADAMS Accession No. ML091410284)
August 27, 2009	Letter, Edwin E. Freeman to NRC Watts Bar Nuclear Plant (WBN) - Unit 2 - Final Safety Analysis Report (FSAR), Amendment 94. (ADAMS Accession No. ML092460757)
September 8, 2009	Letter, M. Bajestani to NRC, providing additional information regarding licensing basis preservation and construction refurbishment program. (ADAMS Accession No. ML092570391)
January 15, 2010	Letter, R. M. Krich (TVA) to NRC, updates TVA's Nuclear Quality Assurance Program, TVA-NQA-PLN89-A, Revision 22), with no reduction in commitment. (ADAMS Accession No. ML1002109720)
June 30, 2010	Safety evaluation regarding generic letter and bulletins related to reactor pressure vessel head inspections (TAC Nos. MD6709, MD6710, and MD6711). (ADAMS Accession No. ML100539515)
July 2, 2010	Letter, Masoud Bajestani to NRC Watts Bar Nuclear Plant (WBN) Unit 2 - Submittal of Additional Information Requested During May 12, 2010, Request for Additional Information (RAI) Clarification Teleconference Regarding Environmental Review (TAC No. MD8203). (ADAMS Accession No. ML101930470)
July 12, 2010	Letter, Masoud Bajestani to NRC Watts Bar Nuclear Plant (WBN) Unit 2 - Submittal of Preoperational Test Instructions. (ADAMS Accession No. ML102030131)
July 14, 2010	Letter, Masoud Bajestani to NRC WATTS BAR NUCLEAR PLANT (WBN) UNIT 2 – ADDITIONAL INFORMATION REGARDING FINAL SAFETY ANALYSIS REPORT (FSAR),CHAPTER 7, "INSTRUMENTATION AND CONTROLS" REVIEW - REQUESTED COMMON Q PROPRIETARY DOCUMENTS. (ADAMS Accession No. ML102030547)
July 16, 2010	Letter, Masoud Bajestani to NRC WATTS BAR NUCLEAR PLANT (WBN) UNIT 2 – REQUEST FOR ADDITIONAL INFORMATION REGARDING FIRE PROTECTION PROGRAM (TAC NO. ME0853). (ADAMS Accession No. ML101970311)
July 23, 2010	Letter, M. Bajestani to NRC WATTS BAR NUCLEAR PLANT (WBN) UNIT 2 - REQUEST FOR ADDITIONAL INFORMATION REGARDING SEVERE ACCIDENT MANAGEMENT ALTERNATIVES (TAC NO. MD8203). (ADAMS Accession No. ML102100588)
July 29, 2010	Letter, M. Bajestani to NRC Watts Bar Nuclear Plant (WBN) Unit 2 - Response to Request for Additional Information Regarding Antitrust Review (TAC No. ME0853). (ADAMS Accession No. ML102160085)

July 30, 2010	Letter, M. Bajestani to NRC Response to Request for Additional Information Regarding the Safety Evaluation Report for 10 CFR 70 License Application for Watts Bar Nuclear Plant, Unit 2 (TAC No. L32918). (ADAMS Accession No. ML102160348)
July 30, 2010	Letter, M. Bajestani to NRC Watts Bar Nuclear Plant (WBN) Unit 2 - Status of Regulatory Framework for the Completion of Construction and Licensing for Unit 2 - Revision 3 (TAC No. MD6311), and Status of Generic Communications for Unit 2 - Revision 3 (TAC No. MD8314). (ADAMS Accession No. ML102170077)
July 30, 2010	Letter, M Bajestani to NRC Watts Bar Nuclear Plant (WBN) Unit 2 - Response to Request for Additional Information Regarding Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves (TAC No. MD6717). (ADAMS Accession No. ML102140191)
August 5, 2010	Letter, R. M. Krich to NRC Response to NRC Regulatory Issue Summary 2010-08,Preparation and Scheduling of Operator Licensing Examinations. (ADAMS Accession No. ML102210179)
August 5, 2010	Letter, M. Bajestani to NRC Watts Bar Nuclear Plant (WBN) Unit 2 - Response to Request For Additional Information (RAI) Regarding Licensee's Final Safety Analysis Report Amendment Related To Chapter 8, "Electric Power" (TAC No. ME2731). (ADAMS Accession No.ML102290350)
August 6, 2010	Letter, M. Bajestani to NRC Watts Bar Nuclear Plant (WBN) Unit 2 - Final Safety Analysis Report (FSAR) - Response to Requests for Additional Information. (ADAMS Accession No. ML102210440)
August 11, 2010	Letter, M. Bajestani to NRC WATTS BAR NUCLEAR PLANT (WBN) UNIT 2 - NRC AUGUST 3-4, 2010, INSTRUMENTATION AND CONTROLS MEETING - REQUEST FOR ADDITIONAL INFORMATION. (ADAMS Accession No. ML102240382)
August 12, 2010	Letter, M. Bajestani to NRC WATTS BAR NUCLEAR PLANT (WBN) UNIT 2 - REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING INDIVIDUAL PLANT EXAMINATION (TAC NO. ME3334). (ADAMS Accession No. ML102280585)
August 16, 2010	Letter, M. Bajestani to NRC Watts Bar Nuclear Plant (WBN) - Unit 2 - Change to Developmental TS Section 4.2.2, "Control Rod Assemblies." (ADAMS Accession No. ML102290075)
August 16, 2010	Letter, D. Curran to NRC Seventh Update to Disclosures Pursuant to 10 CFR § 2.336; *Tennessee Valley Authority* (Watts Bar Nuclear Plant, Unit 2), Docket No. 50-391-OL. (ADAMS Accession No. ML102280257)
August 30, 2010	Letter, M. Bajestani to NRC Watts Bar Nuclear Plant (WBN) Unit 2 - Submittal of Pre-op Test Instruction. (ADAMS Accession No. ML102420679)
August 30, 2010	Letter, M. Bajestani to NRC Watts Bar Nuclear Plant (WBN) Unit 2 - Final Safety Analysis Report (FSAR) - Response to Requests for Additional. Information. (ADAMS Accession No. ML102510580)
September 1, 2010	Letter, M. Bajestani to NRC Watts Bar Nuclear Plant, Unit 2 • Request for Additional Information (RAI) Regarding Final Safety Analysis Report Amendment Related Section 3.11 (TAC No. ME2731). (ADAMS Accession No. ML10244030)

September 1, 2010	Letter, M. Bajestani to NRC WATTS BAR NUCLEAR PLANT (WBN) - UNIT 2 - FINAL SAFETY ANALYSIS REPORT (FSAR), AMENDMENT 100. (ADAMS Accession No. ML102530216)
September 2, 2010	Letter, E. Freeman to NRC WATTS BAR NUCLEAR PLANT (WBN) UNIT 2 - INSTRUMENTATION AND CONTROLS STAFF INFORMATION REQUESTS. (ADAMS Accession No. ML102660074)
September 10, 2010	Letter, M. Bajestani to NRC Watts Bar Nuclear Plant (WBN) Unit 2 - Response to Generic Letter 2004-02, Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors. (ADAMS Accession No. ML102580175)
September 17, 2010	Letter. M. Bajestani to NRC Watts Bar Nuclear Plant (WBN) Unit 2 - Response to NRC Question Regarding WBN Unit 2 Emergency Plan (TAC No. ME0853). (ADAMS Accession No. ML102600474)
September 17, 2010	Letter, M. Bajestani to NRC WATTS BAR NUCLEAR PLANT (WBN) UNIT 2 - Response to NRC REQUEST FOR ADDITIONAL INFORMATION REGARDING SEVERE ACCIDENT MANAGEMENT ALTERNATIVES SECPOP2000 ERRORS (TAC NO. MD8203). (ADAMS Accession No. ML102600582)
September 29, 2010	Letter, M. Bajestani to NRC Watts Bar Nuclear Plant (WBN) Unit 2 • Response to Request for Additional Information (RAI) Regarding the Preservice Inspection Program Plan (TAC Nos. ME4074 and ME3113). (ADAMS Accession No. ML103730184)
October 13, 2010	Letter, M. Bajestani to NRC Watts Bar Nuclear Plant (WBN) Unit 2 - Response to Request for Additional Information Regarding the Preservice Inspection Program Plan (TAC Nos. ME4074 and ME3113). (ADAMS Accession No. ML102861846)
October 29, 2010	Letter, M. Bajestani to NRC WATTS BAR NUCLEAR PLANT (WBN) - UNIT 2 - FINAL SAFETY ANALYSIS REPORT (FSAR), AMENDMENT 101. (ADAMS Accession No. ML103160410)
November 9, 2010	Letter, M. Bajestani to NRC Watts Bar Nuclear Plant (WBN) Unit 2 - Final Safety Analysis Report (FSAR) - Response to Requests for Additional Information. (ADAMS Accession No. ML103200146)
November 18, 2010	Letter, R.M. Krich to NRC Response to Request for Additional Information Regarding Target Set Development. (ADAMS Accession No. ML103280018)
December 6, 2010	Letter, M. Bajestani to NRC Watts Bar Nuclear Plant (WBN) Unit 2 - Safety Evaluation Report Supplement 22 (SSER22) - Response to Requests for Additional Information. (ADAMS Accession No. ML103420569)
December 10, 2010	Letter, M. Bajestani to NRC Watts Bar Nuclear Plant (WBN) Unit 2 - Final Safety Analysis Report (FSAR) - Response to Requests for Additional Information. (ADAMS Accession No. ML103480708)
December 14, 2010	Letter, M. Bajestani to NRC WATTS BAR NUCLEAR PLANT (WBN) UNIT 2 – REFURBISHMENT PROGRAM AND LICENSING BASIS PRESERVATION SELF ASSESSMENTS. (ADAMS Accession No. ML103510694)

December 17, 2010	Letter, M. Bajestani to NRC Watts Bar Nuclear Plant (WBN) Unit 2 - Safety Evaluation Report Supplement 22 (SSER22) - Response to Requests for Additional Information. (ADAMS Accession No. ML103540560)
December 21, 2010	Letter, M. Bajestani to NRC Watts Bar Nuclear Plant (WBN) Unit 2 - Submittal of Pre-op Test Instruction. (ADAMS Accession No. ML103610114)

APPENDIX HH

WATTS BAR UNIT 2 ACTION ITEMS TABLE

This table provides a status of required action items associated of all open items, confirmatory issues, and proposed license conditions that the staff has identified.

Item	Type	Action Required	Lead	Status
(1)	CI	Review evaluations and corrective actions associated with a power assisted cable pull. (NRC safety evaluation dated August 31, 2009, ADAMS Accession No. ML092151155)	NRR	Open
(2)	CI	Conduct appropriate inspection activities to verify cable lengths used in calculations and analysis match as-installed configuration. (NRC safety evaluation dated August 31, 2009, ADAMS Accession No. ML092151155)	RII	Open
(3)	CI	Confirm TVA submitted update to FSAR section 8.3.1.4.1. (NRC safety evaluation dated August 31, 2009, ADAMS Accession No. ML092151155)	NRR	Open
(4)	CI	Conduct appropriate inspection activities to verify that TVA's maximum SWBP criteria for signal level and coaxial cables do not exceed the cable manufacturers maximum SWBP criteria. (NRC safety evaluation dated August 31, 2009, ADAMS Accession No. ML092151155)	RII	Open
(5)	CI	Verify timely submittal of pre-startup core map and perform technical review. (TVA letter dated September 7, 2007, ADAMS Accession No. ML072570676)	NRR	Open
(6)	CI	Verify implementation of TSTF-449. (TVA letter dated September 7, 2007, ADAMS Accession No. ML072570676)	NRR	Open
(7)	CI	Verify commitment completion and review electrical design calculations. (TVA letter dated October 9, 1990, ADAMS Accession No. ML073551056)	NRR	Open
(8)	CI	Verify rod control system operability during power ascension. TVA should provide a pre-startup map to the NRC staff indicating the rodded fuel assemblies and a projected end of cycle burnup of each rodded assembly for the initial fuel cycle 6-months prior to fuel load. (NRC safety evaluation dated May 3, 2010, ADAMS Accession No. ML101200035)	NRR	Open
(9)	CI	Confirm that education and experience of management and principal supervisory positions down through the shift supervisory level conform to Regulatory Guide 1.8. (Section 13.1.3)	NRR	Open

(10)	CI	Confirm that TVA has an adequate number of licensed and non-licensed operators in the training pipeline to support the preoperational test program, fuel loading, and dual unit operation. (Section 13.1.3)	NRR	Open
(11)	CI	The plant administrative procedures should clearly state that, when the Assistant Shift Engineer assumes his duties as Fire Brigade Leader, his control room duties are temporarily assumed by the Shift Supervisor (Shift Engineer), or by another SRO, if one is available. The plant administrative procedures should clearly describe this transfer of control room duties. (Section 13.1.3)	NRR	Open
(12)		TVA's implementation of NGDC PP-20 and EDCR Appendix J is subject to future NRC audit and inspection. (Section 25.9)	NRR / Region	Open
(13)		TVA is expected to submit an IST program and specific relief requests for WBN Unit 2 nine months before the projected date of OL issuance. (Section 3.9.6)	NRR	Open
(14)		TVA stated that the Unit 2 PTLR is included in the Unit 2 System Description for the Reactor Coolant System (WBN2-68-4001), which will be revised to reflect required revisions to the PTLR by September 17, 2010. (Section 5.3.1)	NRR	Open
(15)		TVA should confirm to the NRC staff the completion of Primary Stress Corrosion Cracking (PWSCC) mitigation activities on the Alloy 600 dissimilar metal butt welds (DMBWs) in the primary loop piping. (Section 3.6.3)	NRR	Open
(16)		Based on the uniqueness of EQ, the NRC staff must perform a detailed inspection and evaluation prior to fuel load to determine how the WBN Unit 2 EQ program complies with the requirements of 10 CFR 50.49. (Section 3.11.2)	NRR	Open
(17)		The NRC staff should verify the accuracy of the WBN Unit 2 EQ list prior to fuel load. (Section 3.11.2.1)	NRR	Open
(18)		Based on the extensive layup period of equipment within WBN Unit 2, the NRC staff must review, prior to fuel load, the assumptions used by TVA to re-establish a baseline for the qualified life of equipment. The purpose of the staff's review is to ensure that TVA has addressed the effects of environmental conditions on equipment during the layup period. (Section 3.11.2.2)	NRR	Open
(19)		The NRC staff should complete its review of TVA's EQ Program procedures for WBN Unit 2 prior to fuel load. (Section 3.11.2.2.1)	NRR	Open
(20)		Resolve whether or not routine maintenance activities should result in increasing the EQ of the 6.9 kV motors to Category I status in accordance with 10 CFR 50.49. (Section 3.11.2.2.1).	NRR	Open

(21)		The NRC staff should confirm that the Electrical Penetration Assemblies (EPAs) are installed in the tested configuration, and that the feedthrough module is manufactured by the same company and is consistent with the EQ test report for the EPA. (Section 3.11.2.2.1)	NRR	Open
(22)		TVA must clarify its use of the term "equivalent" (e.g., identical, similar) regarding the replacement terminal blocks to the NRC staff. If the blocks are similar, then a similarity analysis should be completed and presented to the NRC for review. (Section 3.11.2.2.1)	NRR	Open
(23)		Resolve whether or not TVA's reasoning for not upgrading the MSIV solenoid valves to Category I is a sound reason to the contrary, as specified in 10 CFR 50.49(l). (Section 3.11.2.2.1)	NRR	Open
(24)		The NRC staff requires supporting documentation from TVA to justify its establishment of a mild environment threshold for total integrated dose of less than $1x10^3$ rads for electronic components such as semiconductors or electronic components containing organic material. (Section 3.11.2.2.1)	NRR	Open
(25)		Prior to the issuance of an operating license, TVA is required to provide satisfactory documentation that it has obtained the maximum secondary liability insurance coverage pursuant to 10 CFR 140.11(a)(4), and not less than the amount required by 10 CFR 50.54(w) with respect to property insurance, and the NRC staff has reviewed and approved the documentation. (Section 22.3)	NRR	Open
(26)		For the scenario with an accident in one unit and concurrent shutdown of the second unit without offsite power, TVA stated that Unit 2 pre-operational testing will validate the diesel response to sequencing of loads on the Unit 2 emergency diesel generators (EDGs). The NRC staff will evaluate the status of this issue and will update the status of the EDG load response in a future SSER. (Section 8.1)	NRR	Open
(27)		TVA should provide a summary of margin studies based on scenarios described in Section 8.1 for CSSTs A, B, C, and D. (Section 8.2.2)	NRR	Open
(28)		TVA should provide to the NRC staff a detailed discussion showing that the load tap changer is able to maintain the 6.9 kV bus voltage control band given the normal and post-contingency transmission operating voltage band, bounding voltage drop on the grid, and plant conditions. (Section 8.2.2)	NRR	Open
(29)		TVA should provide the transmission system specifics (grid stability analyses) to the NRC staff. In order to verify compliance with GDC 17, the results of the grid stability analyses must indicate that loss of the largest electric supply to the grid, loss of the largest load from	NRR	Open

		the grid, loss of the most critical transmission line, or loss of both units themselves, will not cause grid instability. (Section 8.2.2)		
(30)		TVA should confirm that all other safety-related equipment (in addition to the Class 1E motors) will have adequate starting and running voltage at the most limiting safety related components (such as motor operated valves, contactors, solenoid valves or relays) at the degraded voltage relay setpoint dropout setting. TVA should also confirm that the final Technical Specifications are properly derived from these analytical values for the degraded voltage settings. (Section 8.3.1.2)	NRR	Open
(31)		TVA should evaluate the re-sequencing of loads, with time delays involved, in the scenario of a LOCA followed by a delayed LOOP, and ensure that all loads will be sequenced within the time assumed in the accident analysis. (Section 8.3.1.11)	NRR	Open
(32)		TVA should provide to the NRC staff the details of the administrative limits of EDG voltage and speed range, and the basis for its conclusion that the impact is negligible, and describe how it accounts for the administrative limits in the Technical Specification surveillance requirements for EDG voltage and frequency. (Section 8.3.1.14)	NRR	Open
(33)	CI	TVA stated in Attachment 9 of its letter dated July 31, 2010, that certain design change notices (DCNs) are required or anticipated for completion of WBN Unit 2, and that these DCNs were unverified assumptions used in its analysis of the 125 V dc vital battery system. Verification of completion of these DCNs to the NRC staff is necessary prior to issuance of the operating license. (Section 8.3.2.3)	NRR	Open
(34)	CI	TVA stated that the method of compliance with Phase I guidelines would be substantially similar to the current Unit 1 program and that a new Section 3.12 will be added to the Unit 2 FSAR that will be materially equivalent to Section 3.12 of the current Unit 1 FSAR. (Section 9.1.4)	NRR	Open
(35)		TVA should provide information to the NRC staff that the CCS will produce feedwater purity in accordance with BTP MTEB 5-3 or, alternatively, provide justification for producing feedwater purity to another acceptable standard. (Section 10.4.6)	NRR	Open
(36)		TVA should provide information to the NRC staff to enable verification that the SGBS meets the requirements and guidance specified in the SER or provide justification that the SGBS meets other standards that demonstrate conformance to GDC 1 and GDC 14. (Section 10.4.8)	NRR	Open

(37)	CI	The NRC staff will review the combined WBN Unit 1 and 2 Appendix C prior to issuance of the Unit 2 OL to confirm (1) that the proposed Unit 2 changes were incorporated into Appendix C, and (2) that changes made to Appendix C for Unit 1 since Revision 92 and the changes made to the NP-REP since Revision 92 do not affect the bases of the staff's findings in this SER supplement. (Section 13.3.2)	NSIR	Open
(38)	CI	The NRC staff will confirm the availability and operability of the ERDS for Unit 2 prior to issuance of the Unit 2 OL. (Section 13.3.2.6)	NSIR	Open
(39)	CI	The NRC staff will confirm the adequacy of the communications capability to support dual unit operations prior to issuance of the Unit 2 OL. (Section 13.3.2.6)	NSIR	Open
(40)	CI	The NRC staff will confirm the adequacy of the emergency facilities and equipment to support dual unit operations prior to issuance of the Unit 2 OL. (Section 13.3.2.8)	NSIR	Open
(41)	CI	TVA committed to (1) update plant data displays as necessary to include Unit 2, and (2) to update dose assessment models to provide capabilities for assessing releases from both WBN units. The NRC staff will confirm the adequacy of these items prior to issuance of the Unit 2 OL. (Section 13.3.2.9)	NSIR	Open
(42)	CI	The NRC staff will confirm the adequacy of the accident assessment capabilities to support dual unit operations prior to issuance of the Unit 2 OL. (Section 13.3.2.9)	NSIR	Open
(43)	CI	Section V of Appendix E to 10 CFR Part 50 requires TVA to submit its detailed implementing procedures for its emergency plan no less than 180 days before the scheduled issuance of an operating license. Completion of this requirement will be confirmed by the NRC staff prior to the issuance of an operating license. (Section 13.3.2.18)	NSIR	Open
(44)		TVA should provide additional information to clarify how the initial and irradiated RT_{NDT} was determined. (Section 5.3.1)	NRR	Open
(45)	CI	TVA stated in its response to RAI 5.3.2-2, dated July 31, 2010, that the PTLR would be revised to incorporate the COMS arming temperature. (Section 5.3.2)	NRR	Open
(46)	CI	The LTOP lift settings were not included in the PTLR, but were provided in TVA's response to RAI 5.3.2-2 in its letter dated July 31, 2010. TVA stated in its RAI response that the PTLR would be revised to incorporate the LTOP lift settings into the PTLR. (Section 5.3.2)	NRR	Open
(47)		The NRC staff noted that TVA's changes to Section 6.2.6 in FSAR Amendment 97, regarding the implementation of Option B of Appendix J, were incomplete, because several statements remained	NRR	Open

		regarding performing water-sealed valve leakage tests "as specified in 10 CFR [Part] 50, Appendix J." With the adoption of Option B, the specified testing requirements are no longer applicable; Option A to Appendix J retains these requirements. The NRC discussed this discrepancy with TVA in a telephone conference on September 28, 2010. TVA stated that it would remove the inaccurate reference to Appendix J for specific water testing requirements in a future FSAR amendment. (Section 6.2.6)		
(48)		The NRC staff should verify that its conclusions in the review of FSAR Section 15.4.1 do not affect the conclusions of the staff regarding the acceptability of Section 6.5.3. (Section 6.5.3)	NRR	Open
(49)		The NRC staff was unable to determine how TVA linked the training qualification requirements of ANSI N45.2-1971 to TVA Procedure TI-119. Therefore, the implementation of training and qualification for inspectors will be the subject of future NRC staff inspections. (NRC letter dated July 2, 2010, ADAMS Accession No. ML101720050)	NRR	Open
(50)		TVA stated that about 5 percent of the anchor bolts for safety-related pipe supports do not have quality control documentation, because the pull tests have not yet been performed. Since the documentation is still under development, the NRC staff will conduct inspections to follow-up on the adequate implementation of this construction refurbishment program requirement. (NRC letter dated July 2, 2010, ADAMS Accession No. ML101720050)	NRR	Open
(51)		The implementation of TVA Procedure TI-119 will be the subject of NRC follow-up inspection to determine if the construction refurbishment program requirements are being adequately implemented. (NRC letter dated July 2, 2010, ADAMS Accession No. ML101720050)	NRR	Open

CI – Confirmatory Issue

NRC FORM 335 (12-2010) NRCMD 3.7 U.S. NUCLEAR REGULATORY COMMISSION **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. 22
2. TITLE AND SUBTITLE Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant Unit 2 Docket No. 50-391	3. DATE REPORT PUBLISHED MONTH: February YEAR: 2011 4. FIN OR GRANT NUMBER
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Same as above.

10. SUPPLEMENTARY NOTES

Docket No. 50-391

11. ABSTRACT *(200 words or less)*

This report supplements the safety evaluation report (SER), NUREG-0847 (June 1982), Supplement No. 21 (February 2009; Agencywide Documents Access and Management System (ADAMS) Accession No. ML090570741), with respect to the application filed by the Tennessee Valley Authority (TVA), as applicant and owner, for a license to operate Watts Bar Nuclear Plant (WBN) Unit 2 (Docket No. 50-391).

In its SER and supplemental SER (SSER) Nos. 1 through 20 issued by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission (NRC or the staff), the NRC staff documented its safety evaluation and determination that WBN Unit 1 met all applicable regulations. Based on satisfactory findings from all applicable inspections, on February 7, 1996, the NRC issued a full-power operating license for WBN Unit 1.

In SSER 21, the staff addressed TVA's application for a license to operate WBN Unit 2, and provided information regarding the status of items remaining to be resolved, which were open at the time that TVA deferred construction of WBN Unit 2, and which were not evaluated and resolved as part of the licensing of WBN Unit 1. In this and future SSERs, the staff will document its evaluation and closure of open items in its review of TVA's application for an operating license for WBN Unit 2.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	13. AVAILABILITY STATEMENT unlimited 14. SECURITY CLASSIFICATION *(This Page)* unclassified *(This Report)* unclassified 15. NUMBER OF PAGES 16. PRICE
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