NRC-009-R

Final Safety Evaluation Report for Combined Licenses for William States Lee III Nuclear Station Units 1 and 2

U. S. Nuclear Regulatory Commission
Office of New Reactors
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ABSTRACT

This final safety evaluation report[1] (FSER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's technical review of the combined license (COL) application submitted by the applicant for the William States Lee III Nuclear Station (WLS) Units 1 and 2. The applicant, Duke Energy Carolinas, LLC (DEC or the applicant), is a wholly owned subsidiary of Duke Energy Corporation.

By letter dated December 12, 2007, the applicant submitted its application to the NRC for COLs for two AP1000 advanced passive pressurized-water reactors pursuant to the requirements of Sections 103 and 185(b) of the Atomic Energy Act of 1954, as amended; Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, "Licenses, certifications and approvals for nuclear power plants," and the associated material licenses under 10 CFR Part 30, "Rules of general applicability to domestic licensing of byproduct material"; 10 CFR Part 40, "Domestic licensing of source material"; and 10 CFR Part 70, "Domestic licensing of special nuclear material." These reactors are identified as WLS Units 1 and 2, and would be located at a greenfield site in Cherokee County, South Carolina. The applicant submitted its final update to the COL application, Revision 11, on April 11, 2016.

The application incorporated by reference 10 CFR Part 52, Appendix D, "Design Certification Rule for the AP1000 Design," including the AP1000 Design Certification Document (DCD) Revision 19. The results of the NRC staff's evaluation of the AP1000 DCD are documented in NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," and its supplements.

This FSER presents the results of the staff's review of information submitted in conjunction with the COL application, except those matters resolved as part of the referenced design certification rule. Appendix A to this FSER identifies certain license conditions and inspections, tests, analyses and acceptance criteria (ITAAC) that the staff recommends the Commission impose, should COLs be issued to the applicant. In addition to the ITAAC in Appendix A, the ITAAC found in the AP1000 DCD Revision 19 Tier 1 material will also be incorporated into the COLs, should COLs be issued to the applicant.

[1] This FSER documents the NRC staff's position on all safety issues associated with the combined license application. The Advisory Committee on Reactor Safeguards (ACRS) independently reviewed those aspects of the application that concern safety, as well as the advanced safety evaluation report without open items (an earlier version of this document), and provided the results of its review to the Commission in reports dated December 14, 2015 and April 18, 2016. These reports are included as Appendix F to this FSER.

The staff's review[2] of the application, as documented in this FSER, supports the following conclusions with respect to the safety aspects of the COL application: 1) the applicable standards and requirements of the Atomic Energy Act and Commission regulations have been met; 2) required notifications to other agencies or bodies have been duly made; 3) there is reasonable assurance that the facility will be constructed and will operate in conformity with the license, the provisions of the Atomic Energy Act, and the Commission's regulations; 4) the applicant is technically and financially qualified to engage in the activities authorized; and 5) issuance of the license will not be inimical to the common defense and security or to the health and safety of the public.

[2] An environmental review was also performed of the COL application, and its evaluation and conclusions are documented in NUREG-2111, "Final Environmental Impact Statement for Combined Licenses for William States Lee III Nuclear Station Units 1 and 2," dated December 2013.

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The chapter and section layout of this FSER is consistent with the format of (1) NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)"; (2) Regulatory Guide (RG) 1.206, "Combined License Applications for Nuclear Power Plants"; and (3) the applicant's final safety analysis report (FSAR). Where applicable, references to other regulatory actions (e.g., design certifications) are included in the text of the safety evaluation report (SER).

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EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52 include requirements for licensing new nuclear power plants.[3] These regulations include the NRC's requirements for design certification and combined license (COL) applications. The COL process (10 CFR Part 52, Subpart C, "Combined Licenses") allows an applicant to seek authorization to construct and operate a new nuclear power plant.

This FSER describes the results of a review by the NRC staff of a COL application submitted for two new reactors to be located at the William States Lee III Nuclear Station (WLS) Units 1 and 2 site. The applicant is Duke Energy Carolinas, LLC (DEC). The staff's review was to determine the applicant's compliance with the requirements of Subpart C of 10 CFR Part 52, as well as the applicable requirements under 10 CFR Parts 30, 40, and 70 governing the possession and use of source, byproduct and special nuclear materials. This FSER identifies the staff's conclusions with respect to the COL safety review.

The NRC regulations in 10 CFR Part 51, "Environmental protection regulations for domestic licensing and related regulatory functions," also require an applicant to submit an environmental report. The NRC reviews the environmental report as part of the Agency's responsibilities under the National Environmental Policy Act of 1969, as amended. The NRC presents the results of that review in a final environmental impact statement (FEIS), which is a report separate from this FSER. The staff's FEIS, NUREG-2111, "Final Environmental Impact Statement for Combined Licenses (COLs) for William States Lee II Nuclear Station Units 1 and 2," was issued in December 2013, and can be accessed through the Agencywide Documents Access and Management System (ADAMS) at Accession Nos. ML13340A005, ML13340A006, and ML13340A007.[4]

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[3] Applicants may also choose to seek a construction permit (CP) and operating license in accordance with 10 CFR Part 50, "Domestic licensing of production and utilization facilities," instead of using the 10 CFR Part 52 process.

[4] The Agencywide Documents Access and Management System (ADAMS) is the NRC's information system that provides access to all image and text documents that the NRC has made public since November 1, 1999, as well as bibliographic records (some with abstracts and full text) that the NRC made public before November 1999. Documents available to the public may be accessed via the Internet at http://www.nrc.gov/reading-rm/adams.html#web-based-adams. Documents may also be viewed by visiting the NRC's Public Document Room at One White Flint North, 11555 Rockville Pike, Rockville, Maryland. Telephone assistance for using web-based ADAMS is available at (800) 397-4209 between 8:30 a.m. and 4:15 p.m., Eastern Time, Monday through Friday, except Federal holidays. The staff is also making this FSER available on the NRC's new reactor licensing public web site at http://www.nrc.gov/reactors/new-reactors/col/levy/documents.html.

By letter dated December 12, 2007, the applicant submitted its initial application to the NRC for COLs for two AP1000 advanced passive pressurized-water reactors (PWRs) (ADAMS Accession No. ML073510494) to be located at the WLS site. The application identified the two units as WLS Units 1 and 2. The WLS site is located in Cherokee County, South Carolina, approximately 35 miles southwest of Charlotte, North Carolina, approximately 25 miles northeast of Spartanburg, South Carolina, and approximately 7.5 miles southeast of Gaffney, South Carolina.

The application incorporated by reference 10 CFR Part 52, Appendix D, “Design Certification Rule for the AP1000 Design,” including the AP1000 Design Certification Document (DCD) Revision 19. The results of the NRC staff’s evaluation of the AP1000 DCD are documented in NUREG-1793, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design,” and its supplements. The applicant submitted its final update to the COL application, Revision 11, on April 11, 2016.

Appendix A to this FSER identifies certain license conditions, and inspections, tests, analyses and acceptance criteria (ITAAC) that the staff recommends the Commission impose, should COLs be issued to the applicant. In addition to the ITAAC in Appendix A, the ITAAC found in the AP1000 DCD Revision 19 Tier 1 material will also be incorporated into the COLs should COLs be issued to the applicant.

Inspections and audits conducted by the NRC have verified, where appropriate, the conclusions in this FSER. The inspections focused on selected information in the COL application and its references. The FSER identifies applicable inspection reports as reference documents.

The NRC’s Advisory Committee on Reactor Safeguards (ACRS) also reviewed the bases for the conclusions in this report. The ACRS independently reviewed those aspects of the application that concern safety, as well as the advanced safety evaluation report without open items (an earlier version of this document), and provided the results of its review to the Commission in a report dated December 14, 2015. Appendix F includes a copy of this report by the ACRS on the COL application, as required by 10 CFR 52.87, “Referral to the Advisory Committee on Reactor Safeguards (ACRS).”

ABBREVIATIONS

χ/Q	atmospheric dispersion
A2LA	American Association for Laboratory Accreditation
AB	annex building
ac	alternating current
ACI	American Concrete Institute
ACP	access control point
ACRS	Advisory Committee on Reactor Safeguards
ADAMS	Agencywide Documents Access and Management System
ADS	automatic depressurization system
AE	architect-engineer
AEA	Atomic Energy Act of 1954
AFFF	aqueous film forming foam
AFUDC	allowance for funds used during construction
AHPS	Advanced Hydrologic Prediction Service
ALARA	as low as is reasonable achievable
ALI	annual limit on intake
ALWR	advanced light-water reactor
AMP	amperes
ANI	American Nuclear Insurers
ANS	Alert and Notification Systems
ANS	American Nuclear Society
ANSI	American National Standards Institute
ANSS	Advanced National Seismic System
AOO	anticipated operational occurrence
AOV	air-operated valve
ASA	Applicable Safety Analyses
ASCE	American Society of Civil Engineers
ASE	advanced safety evaluation
ASHRAE	American Society of Heating, Refrigerating and Air-Conditioning Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATE	advisory to evacuate
ATWS	anticipated transients without scram
AWWA	American Water Works Association
B&PV	Boiler and Pressure Vessel (ASME BPV Code)
BDBE	beyond-design basis event
BE	best estimate

BL	Bulletin
BLN	Bellefonte Nuclear Station
BPV	Boiler & Pressure Vessel
BTP	Branch Technical Position
BWR	boiling-water reactor
C	Celsius
C&C	command & control
CAS	central alarm station
CAV	cumulative absolute velocity
CCS	component cooling water system
CDF	core damage frequency
CDI	conceptual design information
CDM	certified design material
CDRS	control rod drive system
CEM	Coastal Engineering Manual
CFBC	Cross Florida Barge Canal
CFD	computational fluid dynamics
cfm	cubic feet per minute
CFR	*Code of Federal Regulations*
cfs	cubic feet per second
cGy	centiGray
CLSM	controlled low strength material
cm	centimeters
CMT	core makeup tank
COL	combined license
CP	construction permit
CR	control room
CR3	Crystal River Unit 3
CRD	control rod drive
CRDM	control rod drive mechanism
CRDS	control rod drive system
CREC	Crystal River Energy Complex
CRNP	Crystal River Nuclear Plant
CRR	cyclic resistance ratio
CS	containment system
CS	core supports
CS	critical system
CSA	control support area
CSC	Coastal Services Center

CSDRS	certified seismic design response spectra
CTA	critical target area
CVCS	chemical and volume control system
CVS	chemical and volume control system
CWS	circulating water system
DAC	derived air concentration
DAS	Diverse Actuation System
DBA	design-basis accident
DBE	design-basis event
DBT	design-basis threat
dc	direct current
DC	design certification
DCA	design certification amendment
DCD	design control document
DCP	Design Change Package
DCRA	design-centered review approach
DE	deaggregation earthquakes
DEI	dose equivalent iodine
DEM	digital elevation model
DEM	Division of Emergency Management
DEP	Departure
DF	design factor
DG	diesel generator
DHBRC	Department of Health, Bureau of Radiation Control
DHEC	Department of Health and Environmental Control
DHS	Department of Homeland Security
DNBR	departure from nucleate boiling ratio
DOE	Department of Energy
DOT	Department of Transportation
D-RAP	Design Reliability Assurance Program
DTS	demineralized water treatment system
DVI	direct vessel injection
DWS	demineralized water system
EAB	exclusion area boundary
EAL	emergency action level
EAS	Emergency Alert System
EC	Emergency Coordinator
ECC-GC	extended continental crust Gulf Coast

ECCS	emergency core cooling system
ECL	effective concentration limit
ED	Emergency Director
EDMG	Extensive Damage Mitigation Guidelines
EIA	Energy Information Agency
EIS	Environmental Impact Statement
ENC	Emergency News Center
ENS	Emergency Notification System
EOC	emergency operation center
EOF	emergency operations facility
EOP	emergency operating procedure
EOP	emergency operating plan
EP	Emergency Plan
EP	emergency planning
EPA	Environmental Protection Agency
EPAct	Energy Policy Act of 2005
EPC	engineering, procurement, and construction
EPDM	ethylene propylene diene monomer
EPIP	emergency plan implementing procedure
EP-ITAAC	emergency planning-inspections, tests, analyses, and acceptance criteria
EPRI	Electric Power Research Institute
EPZ	emergency planning zone
EQ	environmental qualification
EQL	equivalent linear
EQMEL	Environmental Qualification Master Equipment List
ERDS	emergency response data system
ERF	emergency response facility
ERM	Eastern rift margin
ERO	emergency response officer
ERO	Emergency Response Organization
ERT	emergency response team
ESATCOM	Emergency Satellite Communications System
ESF	engineered safety feature
ESFAS	engineered safety features actuation system
ESP	Early Site Permit
ETE	evacuation time estimate
ETS	Emergency Telephone System
F	Fahrenheit
FAC	flow-accelerated corrosion

FBI	Federal Bureau of Investigation
FDLE	Department of Law Enforcement
FEIS	final environmental impact statement
FEM	Finite Element Model
FEMA	Federal Emergency Management Agency
FERC	Federal Energy Regulatory Commission
FFD	fitness for duty
FHA	Fuel Handling Accident
FIRS	foundation input response spectra
FIV	flow induced vibration
FMCRD	fine motion control rod drive
FMEA	failure mode and effects analysis
fps	feet per second
FPS	fire protection system
FPSC	Florida Public Service Commission
FR	*Federal Register*
FRCC	Florida Reliability Coordinating Council
FRS	floor response spectra
FS	factor of safety
FSAR	final safety analysis report
FSER	final safety evaluation report
ft	feet
GALL	Generic Aging Lessons Learned
GCSZ	Gulf Coastal Source Zones
GDC	General Design Criteria (Criterion)
GE	General Emergency
GG&S	Geotechnical, Geological, and Seismological
GL	Generic Letter
GMRS	ground motion response spectra
gpm	gallons per minute
GSI	Generic Safety Issue
GSI	geologic strength index
GSU	generator step-up
GTS	generic technical specification
GWMS	gaseous waste management system
HCM	Highway Capacity Manual
HCLPF	high confidence in low probability of failure
HEPA	high efficiency particulate air

HFE	human factors engineering
HP	health physics
HPN	Health Physics Network
HPS	Health Physics Society
hr	hour
HRA	human reliability analysis
HRHF	hard rock high frequency
HRTS	Hot Ringdown Telephone System
HSI	human-system interface
HV	high voltage
HVAC	heating, ventilation, and air conditioning
HX	heat exchanger
Hz	Hertz
HZP	Hot Zero Power
I&C	instrumentation and control
IBC	International Building Code
ICMO	interim compensatory order
IDLH	immediate danger to life and health
IDS	1E dc and uninterruptible power supply system
IEEE	Institute of Electrical and Electronic Engineers
IFR	Interim Findings Report
IGSCC	intergranular stress corrosion cracking
IHP	integrated head package
IIS	incore instrumentation system
ILAC	International Laboratory Accreditation Cooperation
in	inch
INPO	Institute of Nuclear Power Operations
IRWST	in-containment refueling water storage tank
ISA	independent safety assessment
ISC	International Seismological Centre
ISG	Interim Staff Guidance
ISI	inservice inspection
ISL	Information Systems Laboratory, Inc.
IST	inservice testing
ITAAC	inspections, tests, analyses, and acceptance criteria
ITP	Initial Test Program
JOG	Joint Owners Group
JTWG	Joint Test Working Group

kg/m^3	kilogram per cubic meter
kg/yr	kilograms per year
km	kilometers
kPa	kilopascal
kV	kilovolt
kWe	kilowatt electric
LAN	Local Area Network
lb/ft^2	pounds per square foot
LB	lower bound
LBB	leak-before-break
LCCWS	low capacity chilled water subsystem
LCD	Local Climatological Data
LCO	limiting condition for operation
LEFM	Leading Flow Edge Meter
LLB	Lower LB case
LLEA	local law enforcement agency
LLHS	light load handling system
LLNL	Lawrence Livermore National Laboratory
LLRW	low-level radioactive waste
LMA	left margin annotation
LNP	Levy Nuclear Plant
LOA	letter of agreement
LOAC	Loss of AC Power to Plant Auxiliaries
LOCA	loss-of-coolant accident
LOLA	loss of large area
LOOP	loss of offsite power
LPZ	low population zone
LRF	large release frequency
LSS	low strategic significance
LRA	locked rotor accident
LTOP	low-temperature overpressure protection
LWA	Limited Work Authorization
LWMS	liquid waste management system
LWR	light-water reactor
M	magnitude
m	meter
m/s	meters per second

m^3/s	cubic meters per second
Ma	million years ago
MAAP	Modular Accident Analysis Program
m_b	body-wave magnitude
Mbtu/hr	one million British thermal units/hour
MC&A	material control and accounting
MCL	Management Counterpart Link
MCR	main control room
MCRE	main control room envelope
M_d	duration magnitude
MEI	maximally exposed individual
MERL	Mobile Emergency Radiological Laboratory
MESE	Mesozoic and younger extended prior
mgd	million gallons per day
mGy	milliGray
mi	miles
MIDC-A	Midcontinent A
MIT	Massachusetts Institute of Technology
M_L	local magnitude
mld	million liters per day
MLU	Multi-Layer Unsteady
mm	millimeters
Mmax	maximum magnitude
MOA	Memorandum of Agreement
MOM	maximum envelope of water
MOU	Memorandum of Understanding
MOV	motor-operated valve
MOX	mixed-oxide
mph	miles per hour
MR	Maintenance Rule
MRA	Mutual Recognition Arrangement
mrad	millirad
mrem	millirem
MSD	Mitigative Strategies Description
MSLB	Main Steam Line Break
MSSS	main steam supply system
MST	Mitigative Strategies Table
mSv	milliSievert
MT	magnetic particle
MW	megawatts

MWe	megawatts electric
MWt	megawatts thermal
N	North
NCDC	National Climatic Data Center
NDQAM	Nuclear Development Quality Assurance Manual
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NGS	National Geodetic Survey
NI	nuclear island
NIRMA	Nuclear Information and Records Management Association
NIST	National Institute of Standards and Technology
NMFS	New Madrid Fault System
NNS	non-nuclear safety
NOAA	National Oceanic and Atmospheric Administration
NOUE	Notification of Unusual Event
NOV	Notice of Violation
NPSH	net positive suction head
NRC	U.S. Nuclear Regulatory Commission
NRCOC	NRC Headquarters Operations Center
NRF	National Response Framework
NRO	Office of New Reactors
NS	nonseismic
NSM	Nuclear Shift Manager
NSSS	nuclear steam system supplier
NSW	nonlinear shallow-water
NTTF	Near-Term Task Force
NUMARC	Nuclear Management and Resources Council
NVLAP	National Voluntary Laboratory Accreditation Program
NW	northwest
NWS	National Weather Service
OBE	operating basis earthquake
ODCM	Offsite Dose Calculation Manual
OE	operating experience
OER	operating experience review
OHLHS	overhead heavy load handling system
OM	Operation and Maintenance (ASME OM Code)
OPRAA	operational phase reliability assurance activity
ORE	occupational radiation exposure

ORO	Offsite-Response Organizations
OSC	Operational Support Center
OSHA	Occupational Safety and Health Administration
PA	protected area
PABS	private automatic branch system
PAM	Postaccident Monitoring
PAP	primary access point
PAR	protective action recommendation
PBSRS	performance based surface horizontal and vertical response spectra
PCCAWST	passive containment cooling ancillary water storage tank
PCCWST	passive containment cooling water storage tank
pcf	pounds per cubic foot
pcf	per cubic foot
PCP	Process Control Program
PCS	passive containment cooling system
PDP	procedure development program
PE	Polyethylene
PEC	Progress Energy Carolinas, Inc.
PEF	Progress Energy Florida
PF	performance goal
PGA	peak ground acceleration
PGM	Plant General Manager
PGP	procedures generation package
PID	Public Information Director
P&IDs	piping and instrumentation diagrams
PLT	point load test
PM	preventive maintenance
PMCL	Protective Measures Counterpart Link
PMP	probable maximum precipitation
PMS	protection and safety monitoring
PMSS	probable maximum storm surge
PMT	pressuremeter tests
PORV	power-operated relief valve
POV	power-operated valve
ppm	parts per million
PRA	probabilistic risk assessment
PRHR	passive residual heat removal
PRP	Peer Review Panel
psf	pounds per square foot

PSHA	probabilistic seismic hazard analysis
PSI	preservice inspection
psi	pounds per square inch
psig	pounds per square inch gauge
PS-ITAAC	physical security inspections, tests, analyses, and acceptance criteria
PSP	Physical Security Plan
P-T	pressure temperature
PT	liquid penetrant
PT&O	plant test and operations
PTAC	Plant Transmission Activities Coordinator
PTS	plant-specific technical specifications
Pu	per unit
PWR	pressurized-water reactor
PWS	potable water system
PWSCC	primary water stress corrosion cracking
PXS	passive core cooling system
QA	quality assurance
QDPS	Qualified Data Processing System
QAPD	Quality Assurance Program description
QAPD	Quality Assurance Program Document
QC	quality control
RAI	request for additional information
RAP	reliability assurance program
RAT	reserve auxiliary transformer
RB	radwaste building
RCA	radiation controlled area
RCCA	rod cluster control assembly
RCL	reactor coolant loop
RCOL	reference combined license
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
REA	Rod Ejection Accident
REAC/TS	Radiation Emergency Assistance Center/Training Site
rem	roentgen equivalent man
REMP	Radiological Emergency Management Plan
REP	radiological emergency preparedness
RG	regulatory guide

RH	relative humidity
RIS	Regulatory Issue Summary
RLME	repeated large magnitude earthquake
RMS	rock mass rating
RMS	radiation monitoring system
RMS	root-mean-square
RNS	residual heat removal system
RO	reactor operator
RPP	Radiation Protection Program
RPV	reactor pressure vessel
RSCL	Reactor Safety Counterpart Link
RSW	Remote Shutdown Workstation
RTDP	revised thermal design procedure
RT_{NDT}	nil-ductility reference transition temperature
RTNSS	regulatory treatment of nonsafety systems
RTP	rated thermal power
RT_{PTS}	pressurized thermal shock reference temperature
RV	reactor vessel
RVSP	reactor vessel surveillance capsule program
RWS	raw water system
RXS	reactor system
S&PC	steam and power conversion
SAE	Site Area Emergency
SAMSON	Solar and Meteorological Surface Observation Network
SAR	safety analysis report
SAT	systematic approach to training
SBO	station blackout
SC	steel concrete composite
SCBA	self-contained breathing apparatus
SCOR	soil column Outcrop response
SCOR	soil column outcropping response
SCE&G	South Carolina Electric and Gas Company
SCOL	subsequent combined license
SCOR	soil column outcrop response spectra
SCDOT	South Carolina Department of Transportation
SCP	Safeguards Contingency Plan
SCPSC	South Carolina Public Service Commission
SCR	stable continental region
SE	safety evaluation

SEC	Securities and Exchange Commission
SER	safety evaluation report
SFP	spent fuel pool
SFS	spent fuel pool cooling system
SG	steam generator
SGI	safeguards information
SGTR	steam generator tube rupture
SLOSH	Sea, Lake, and Overland Surge from Hurricanes
s/m	seconds per cubic meter
SNC	Southern Nuclear Operating Company
SNM	special nuclear material
SMA	seismic margins analysis
SNMPPP	Special Nuclear Material Physical Protection Program
SP	Setpoint Program
SPDS	safety parameter display system
SPT	standard penetration test
sq	square
sq mi	square mile
SR	surveillance requirement
SRM	Staff Requirements Memorandum
SRO	senior reactor operator
SRP	standard review plan
SSAR	Site Safety Analysis Report
SSC	seismic source characterization
SSCs	structures, systems, and components
SSE	safe shutdown earthquake
SSI	soil-structure interaction
SS-ITAAC	site-specific inspections, tests, analyses and acceptance criteria
SSHAC	Senior Seismic Hazard Analysis Committee
STA	Shift Technical Advisor
STD	Standard
STS	standard technical specification
SUNSI	Sensitive Unclassified Non-Safeguards Information
SUP	Supplement
Sv	Sievert
SWFWMD	Southwest Florida Water Management District
SWMS	solid waste management system
SWPT	State Warning Point-Tallahassee
SWS	service water system
SWFWMD	South West Florida Water Management District

T&QP	Training and Qualification Plan
TAC	total annual cost
TB	turbine building
TCP	traffic control point
TCS	turbine building closed cooling water system
TEDE	total effective dose equivalent
TG	turbine-generator
TGS	turbine generator system
TLD	thermoluminescent dosimeter
TMI	Three Mile Island
TR	technical report
TS	technical specification
TSC	Technical Support Center
TSCSR	Truncated Soil Column Surface Response
TSO	transmission system operator
TSP	trisodium phosphate
TSTF	Technical Specification Task Force Traveler
TSTF	Technical Specification Task Force
TVA	Tennessee Valley Authority
U	unconfined compressive strength
UAT	unit auxiliary transformer
UB	upper bound
UCS	unconfined compressive strength
UCSS	updated Charleston seismic source
UF_6)	uranium hexafluoride
UFM	ultrasonic flow meter
UFSAR	Updated Final Safety Analysis Report
UHF	ultra high frequency
UHRS	uniform hazard response spectra
UPS	uninterruptible power supply
USACE	United States Army Corps of Engineers
USE	upper shelf energy
USGCRP	United States Global Change Research Program
URD	Utility Requirements Document
USGS	United States Geological Survey
UT	ultrasonic
V&V	verification and validation

VAC	volts alternating current
VBS	nonradioactive ventilation system
VCSNS	V.C. Summer Nuclear Station
Vdc	volts direct current
VEGP	Vogtle Electric Generating Plant
VES	main control room emergency habitability system
VFS	containment air filtration system
V/H	vertical to horizontal
VHRA	very high radiation area
WAC	waste acceptance criteria
WCAP	Westinghouse Commercial Atomic Power
WEC	Westinghouse Electric Company
WSW	worst meteorological sector
WUS	Western United States
WWRB	waste water retention basin
WWS	waste water system
WWS	worst case
YFS	yard fire system

1 INTRODUCTION AND INTERFACES

This chapter of the final safety evaluation report (FSER) is organized as follows:

- Section 1.1 provides an overview of the entire combined license (COL) application
- Section 1.2 provides the regulatory basis for the COL licensing process
- Section 1.3 provides an overview of the COL application principal review matters and where the staff's review of the 10 parts of the COL application is documented
- Section 1.4 documents the staff's review of Chapter 1 of the final safety analysis report (FSAR)
- Section 1.5 documents regulatory findings that are in addition to those directly related to the staff's review of the FSAR

1.1 Summary of Application

In a December 12, 2007, letter as supplemented by several additional letters, Duke Energy Carolinas, LLC (DEC or the applicant), a wholly owned subsidiary of Duke Energy Corporation, submitted an application to the U.S. Nuclear Regulatory Commission (NRC or the Commission) for a combined license (COL) for two Westinghouse AP1000 advanced passive pressurized water reactors (PWRs) pursuant to the requirements of Sections 103 and 185(b) of the *Atomic Energy Act*, and Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, "Licenses, Certifications and Approvals for Nuclear Power Plants." These reactors will be identified as William States Lee III Nuclear Station (WLS), Units 1 and 2, and will be located in the eastern portion of Cherokee County in north central South Carolina, approximately 35 miles southwest of Charlotte, North Carolina; approximately 25 miles northeast of Spartanburg, South Carolina; and approximately 7.5 miles southeast of Gaffney, South Carolina. DEC will be the licensed owner and operator of WLS Units 1 and 2.

According to the COL application, Duke Energy Carolinas, LLC, is a limited liability company duly organized and existing under the laws of the State of North Carolina. It is engaged in the business of generating, transmitting, distributing and selling electric power and energy. It is a "public utility" under the laws of North Carolina and subject to the jurisdiction of the North Carolina Utilities Commission (NCUC) with respect to its operations in that State. The company also transacts business and is an "electrical utility" under the laws of the State of South Carolina; accordingly, its operations in that State are subject to the jurisdiction of the Public Service Commission of South Carolina (PSCSC). DEC owns and operates regulated electrical facilities, including seven (7) nuclear units licensed by the NRC, as well as electrical distribution and transmission facilities.

The COL application incorporates the Design Control Document (DCD) for a simplified passive advanced light water reactor plant provided by Westinghouse Electric Corporation, the entity originally sponsoring and obtaining the AP1000 design certification documented in 10 CFR Part 52, Appendix D, Design Certification Rule for the AP1000 Design.

In addition, as discussed in Section 1.5.5 of this report, the applicant submitted a request for the associated material licenses under 10 CFR Part 30, “Rules of General Applicability to Domestic Licensing of Byproduct Material”; 10 CFR Part 40, “Domestic Licensing of Source Material”; and 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material.”

Unless otherwise noted, this FSER (also referred to as the safety evaluation report (SER) or advanced safety evaluation (ASE) in later sections of this document) is based on Revision 11 of the WLS COL application, which was submitted in April 11, 2016.

As indicated in the applicant’s April 11, 2016 submission, the applicant incorporated by reference 10 CFR Part 52, Appendix D, “Design Certification Rule for the AP1000 Design,” and the Westinghouse Electric Corporation’s (Westinghouse’s) application for amendment to portions of the Design Control Document (DCD) Revision 19.

The AP1000 nuclear reactor design is a PWR with a power rating of 3400 megawatts thermal (MWt) and an electrical output of at least 1000 megawatts electric (MWe). The AP1000 design uses safety systems that rely on passive means, such as gravity, natural circulation, condensation and evaporation, and stored energy, for accident prevention and mitigation.

In developing the FSER for WLS Units 1 and 2, the staff reviewed the AP1000 DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to a particular review topic.

The WLS COL application is organized as follows:

- **Part 1 General and Administrative Information**

Part 1 provides an introduction to the application and includes certain corporate information regarding DEC pursuant to 10 CFR 50.33(a) – (d).

- **Part 2 Final Safety Analysis Report**

Part 2 includes information pursuant to the requirements of 10 CFR 52.79, “Contents of applications; technical information in final safety analysis report” and, in general, adheres to the content and format guidance provided in Regulatory Guide (RG) 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition).”

- **Part 3 Environmental Report**

Part 3 includes environmental information pursuant to the requirements of 10 CFR 52.80, “Contents of applications; additional technical information” and 10 CFR 51.50(c).

- **Part 4 Technical Specifications**

Part 4 addresses how the AP1000 Generic Technical Specifications (GTS) and Bases are incorporated by reference into the WLS Plant-Specific Technical Specifications (PTS) and Bases. Specifically, Section A addresses completion of bracketed information. Section B provides a complete copy of the WLS PTS and Bases.

- **Part 5 Emergency Plan**

Part 5 includes the WLS COL Emergency Plan, supporting information (e.g., evacuation time estimates (ETEs)), and applicable offsite State and local emergency plans.

- **Part 6 [Not Used - reserved for Limited Work Authorization/site redress information]**

- **Part 7 Departures Report**

Part 7 includes information regarding “departures” and “exemptions.” “Departures” refers to departures from the AP1000 DCD, Revision 19, incorporated by reference into the COL application. For each departure, Part 7 of the COL application identifies the portions of the AP1000 DCD and FSAR affected and includes a description, a justification, an evaluation against criteria in 10 CFR 52.63(b), and a concluding statement about whether the departure requires NRC approval pursuant to 10 CFR Part 52, Appendix D, Section VIII.B.5.

“Exemptions” refers to requests for exemptions from NRC regulations. For each exemption request, Part 7 identifies the regulation and specific wording from which an exemption is being requested and provides a discussion supporting the request.

- **Part 8 Security Plan**

Part 8 addresses the WLS Safeguards/Security Plan, which consists of the Physical Security Plan, the Training and Qualification Plan, the Safeguards Contingency Plan, and the Special Nuclear Material (SNM) Physical Protection Program Description. These security plans are submitted to the NRC as a separate licensing document in order to fulfill the requirements of 10 CFR 52.79(a)(35) and 10 CFR 52.79(a)(36). The Safeguards/Security Plan is categorized as Security Safeguards Information and is withheld from public disclosure pursuant to 10 CFR 73.21, “Protection of safeguards information: performance requirements.” The staff’s evaluation of the Safeguards and Security Plans is documented separately from this SER and is withheld from the public in accordance with 10 CFR 73.21. A non-sensitive summary of the staff’s evaluation of those plans is provided in Section 13.6 of this SER.

- **Part 9 Withheld Information**

Part 9 identifies sensitive information that is withheld from public disclosure under 10 CFR 2.390, “Public inspections, exemptions, requests for withholding.” The information in this part includes sensitive unclassified non-safeguards information (SUNSI), proprietary financial information, and figures from Part 2 of the application that meet the SUNSI guidance for withholding from the public. In addition, this part of the application includes the following information:

- The withheld portion of the Mitigative Strategies Description and Plans for loss of large areas of the plant due to explosions or fire, as required by 10 CFR 52.80(d)
- WLS Units 1 and 2 Cyber Security Plan, as required by 10 CFR 73.54, “Protection of Digital Computer and Communication Systems and Networks”
- The withheld portions of the COL application Part 2 – FSAR
- The withheld portions of the COL application Part 5 – Emergency Plan

- Administrative and Financial Information

- The staff's evaluation of the Safeguards and Security Plans is documented separately from this SER and is withheld from the public in accordance with 10 CFR 73.21. A non-sensitive summary of the staff's evaluation of those plans is provided in Section 13.6 of this SER.

- **Part 10 Proposed Combined License Conditions (Including ITAAC)**

Part 10 includes WLS proposed license conditions including inspections, tests, analyses, and acceptance criteria (ITAAC) information in accordance with 10 CFR 52.80. A table of the proposed license conditions is provided in Appendix A of this report.

The contents of the environmental protection plan (and associated license conditions) are not evaluated in this report. Part 10 of the application incorporates by reference the AP1000 DCD Tier 1 information including ITAAC. In addition, the application includes site-specific ITAAC (e.g., emergency planning, physical security, electrical, and piping).

- **Part 11 Enclosures**

Part 11 provides information submitted by the applicant in support of the WLS Units 1 and 2 application. Specifically, these sections include:

- Part 11A describes the DEC new nuclear deployment Quality Assurance Program Description (QAPD). The QAPD is the top-level policy document that establishes the quality assurance (QA) policy and assigns major functional responsibilities for COL/construction/preoperation and operation activities conducted by or for DEC.

- Part 11B includes mitigative strategies description and plans for loss of large areas of the plant due to explosions or fire, as required by 10 CFR 52.80(d).

- Part 11C of the application includes the cyber security plan. The SUNSI version of the cyber security plan is provided in Part 9 of the application.

- Part 11D of the application includes WLS Special Nuclear Material Control and Accounting Program Description.

- Part 11E of the application includes the new fuel shipping plan.

- Part 11F of the application contains supplemental information in support of the 10 CFR Part 70, "Domestic licensing of special nuclear material," special nuclear material license application.

1.2 Regulatory Basis

1.2.1 Applicable Regulations

10 CFR Part 52, Subpart C, "Combined Licenses," sets out the requirements and procedures applicable to Commission issuance of a COL for nuclear power facilities. The following are of particular significance:

- 10 CFR 52.79, “Contents of applications; technical information in final safety analysis report,” identifies the technical information for the FSAR
- 10 CFR 52.79(d) provides additional requirements for a COL referencing a standard certified design
- 10 CFR 52.80, “Contents of applications; additional technical information,” provides additional technical information outside of the FSAR (ITAAC and the environmental report)
- 10 CFR 52.81, “Standards for review of applications,” provides standards for reviewing the application
- 10 CFR 52.83, “Finality of referenced NRC approvals; partial initial decision on site suitability,” provides for the finality of referenced NRC approvals (i.e., standard design certification)
- 10 CFR 52.85, “Administrative review of applications; hearings,” provides requirements for administrative reviews and hearing
- 10 CFR 52.87, “Referral to the Advisory Committee on Reactor Safeguards (ACRS),” provides for referral to the ACRS

The staff reviewed this application according to the standards set out in:

- 10 CFR Part 20, “Standards for Protection Against Radiation”
- 10 CFR Part 30, “Rules of general applicability to domestic licensing of byproduct material”
- 10 CFR Part 40, “Domestic licensing of source material”
- 10 CFR Part 50, “Domestic licensing of production and utilization facilities”
- 10 CFR Part 51, “Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions”
- 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants”
- 10 CFR Part 55, “Operators’ Licenses”
- 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material”
- 10 CFR Part 73, “Physical Protection of Plants and Materials”
- 10 CFR Part 74, “Material Control and Accounting of Special Nuclear Material”
- 10 CFR Part 100, “Reactor Site Criteria”
- 10 CFR Part 140, “Financial Protection Requirements and Indemnity Agreements”

The staff evaluated the application against the acceptance criteria provided in the following:

- NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition”
- NUREG-1555: “Standard Review Plans for Environmental Reviews for Nuclear Power Plants”
- NUREG-1577, “Standard Review Plan on Power Reactor Licensee Financial Qualifications and Decommissioning Funding Assurance”
- NUREG-0800, “Standard Review Plan on Foreign Ownership, Control, or Domination”

In addition, the staff considered the format and content guidance in RG 1.206[1] for the COL application.

1.2.2 Finality of Referenced NRC Approvals

In accordance with 10 CFR 52.83, “Finality of referenced NRC approvals; partial initial decision on site suitability,” if the application for a COL references a design certification rule (DCR), the scope and nature of matters resolved in the design certification (DC) for the application and any COL issued are governed by the relevant provisions. For the AP1000 DCR, finality is based on 10 CFR 52.63, “Finality of standard design certifications.”

Based on the finality afforded to referenced certified designs, the scope of this COL application review, as it relates to the referenced certified design, is limited to items that fall outside the scope of the certified design (e.g., COL information items, design information replacing conceptual design information (CDI), and programmatic elements that are the responsibility of the COL, and departures from the certified design).

The contents of the application are specified in 10 CFR 52.79(a), which requires the information submitted in the FSAR to describe the facility; identify the design bases and the limits on its operation; and present a safety analysis of the structures, systems, and components (SSCs) of the facility as a whole. For a COL application that references a DC, Section 10 CFR 52.79(d) requires the AP1000 DCD to be included in the FSAR or incorporated by reference into the FSAR. Addition, a COL application the references a design certification (DC) must also contain the information and analysis required to be submitted within the scope of the COL application but is outside the scope of the AP1000 DCD. This combined information addresses plant- and site-specific information and includes all COL action or information items; design information the replaces CDI; and programmatic information that was not review and approved in connection with the DC rulemaking.

During its evaluation of the COL application, the staff confirmed that the complete set of information required to be addressed in the COL application was addressed in the DC, the DC

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[1] 10 CFR Part 52, Appendix D, Section IV.A.2.a requires the COL application to include a plant-specific DCD that describes the same type of information and uses the same organization and numbering as the generic DCD. The generic DCD used RG 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition),” Revision 3 as a guide for the format and content. RG 1.206 was issued after the initial certification of the AP1000; thus, there are anticipated differences between the WLS Units 1 and 2 COL application and the guidance of RG 1.206.

as supplemented by the COL application or completely included in the COL application. Following this confirmation, the staff's review of the COL application is limited to the COL review items.

1.2.3 Overview of the Design-Centered Review Approach

The design-centered review approach (DCRA) is described in Regulatory Issue Summary (RIS) 2006-06, "New Reactor Standardization Needed to Support the Design-Centered Licensing Review Approach." The DCRA is endorsed by the Commission's Staff Requirements Memorandum (SRM) SECY-06-0187, "Semiannual Update of the Status of New Reactor Licensing Activities and Future Planning for New Reactors," November 16, 2006. The DCRA, which is the Commission's policy intended to promote standardization of COL applications, is beyond the scope of information included in the DC. This policy directs the staff to perform one technical review for each standard issue outside the scope of the DC, and use this decision to support decisions on multiple COL applications. In this context, "standard" refers to essentially identical information. In some cases the staff has expanded the use of this standard approach to other areas with essentially identical information for regulatory purposes. For example, the QA plan for the AP1000 COL applicants is essentially identical with the exception of title names being different. Other areas where this approach was used include cyber security, technical specifications, and loss of large area fire reviews and may include information provided by the applicant(s) to resolve plant-specific issues.

The first COL application submitted for staff review is designated in a design center as the reference COL (RCOL) application, and the subsequent applications in the design center are designated as subsequent COL (SCOL) applications. The WLS Units 1 and 2 COL application has been designated as an SCOL application in the AP1000 design center[2].

DEC, as an SCOL applicant in the AP1000 design center, organized and annotated its FSAR, Part 2 of the COL application, to clearly identify: (a) sections that incorporate by reference the AP1000 DCD; (b) sections that are standard for COL applicants in the AP1000 design center; and (c) sections that are site-specific and thus only apply to WLS Units 1 and 2. The following notations have been used by the applicant for the departures from and/or supplements to the referenced DCD included in this COL application:

- STD – standard (STD) information that is identical in each COL referencing the AP1000
- WLS – plant-specific information that is specific to this application
- DEP – represents a departure (DEP) from the AP1000 DCD
- COL – represents a COL information item identified in the AP1000 DCD

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[2] In an April 28, 2009, letter, the NuStart Energy Development, LLC, consortium informed the NRC that it had changed the RCOL designation for the AP1000 design center from Bellefonte Nuclear Plant (BLN) Units 3 and 4 to the Vogtle Electric Generating Plant (VEGP) Units 3 and 4. The transition of the RCOL from BLN Units 3 and 4 to VEGP Units 3 and 4 occurred after the issuance of the BLN Units 3 and 4 SER with open items. As part of the transition, the staff concluded that the BLN evaluation material identified as Standard (STD COL, STD SUP, STD DEP and Interfaces for Standard Design) in the BLN SER was directly applicable to the VEGP review. As a result, standard content material from the SER for the RCOL (VEGP) application and referenced in the WLS SER includes evaluation material from the SER for the BLN COL application.

- SUP – represents information that supplements (SUP) information in the AP1000 DCD

- CDI – represents design information replacing conceptual design information (CDI) included in the AP1000 DCD but not addressed within the scope of the AP1000 DCD review

The following text is added to the Technical Evaluation sections in this report whenever the staff uses standard content evaluation material to resolve departures and/or supplements to the referenced DCD:

> Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (Vogtle Electric Generating Plant [VEGP] Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:
>
> - The staff compared the VEGP COL FSAR, Revision 5 to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from requests for additional information (RAIs).
>
> - The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.
>
> - The staff verified that the site-specific differences were not relevant.
>
> The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the BLN Units 3 and 4 COL application.

To support the text added to the Technical Evaluation sections as described above, the staff evaluated any differences between the information provided by the WLS applicant and that provided by the VEGP applicant, regarding details in the application for the standard content material, to determine whether the standard content material of the VEGP SER is still applicable to the WLS application. These evaluations are in the SER sections that reference the standard content.

The staff compared the VEGP COL FSAR Revision 2 to the WLS COL FSAR at the time of the development of the ASE. The ASE included confirmatory items. Subsequent to the issuance of the ASE, DEC updated the standard portions of its application to be consistent with the VEGP COL application to close the standard content confirmatory items. A complete comparison between the subsequent VEGP COL FSAR revisions to the WLS COL FSAR revisions was not performed. However, the staff confirmed that responses to standard content confirmatory items were endorsed by DEC and that the changes discussed in the standard confirmatory items were made in the WLS COL FSAR.

The staff applied the design-centered review approach described above in Chapter 21 “Design Changes Proposed In Accordance With ISG-11” of this SER in conducting its evaluation of the five requests by the applicant to depart from the AP1000 certified design. These five departure requests were identical to departure requests in the Levy Nuclear Plant (LNP) COL review. Therefore, consistent with the NRC’s DCRA, , the staff referenced evaluations that were completed for the first time in the LNP review. The referenced evaluations in Chapter 21 of this SER are captured by use of italicized, double-indented formatting.

1.3 Principal Review Matters

The staff’s evaluations related to the COL application review are addressed as follows:

- **Part 1 General and Administrative Information**

The staff’s evaluation of the corporate information regarding DEC pursuant to 10 CFR 50.33 is provided in Section 1.5.1 of this report.

- **Part 2 Final Safety Analysis Report**

The staff’s evaluation of information in the WLS COL FSAR is provided in the corresponding sections of this report.

There are two SER chapters that have been issued that do not have a corresponding chapter in the FSAR.

Chapter 20 describes the staff’s evaluations and conclusions relating to the Fukushima Near-Term Task Force (NTTF) recommendations that are applicable to the WLS Units 1 and 2 COL application. The applicable recommendations address four topics: a reevaluation of the seismic hazard (related to Recommendation 2.1), mitigation strategies for beyond-design-basis external events (related to Recommendation 4.2), spent fuel pool instrumentation (related to Recommendation 7.1), and emergency preparedness staffing and communications (related to Recommendation 9.3).

Chapter 21 describes the staff’s evaluations and conclusions for departures from the certified design identified by the applicant in accordance with Interim Staff Guidance DC/COL ISG-011, "Finalizing Licensing-Basis Information."

- **Part 3 Environmental Report**

The applicant submitted an Environmental Report pursuant to the requirements of 10 CFR 51.50(c). The staff’s evaluation of this information is provided in the Environmental Impact Statement.

- **Part 4 Technical Specifications**

Chapter 16 of this report includes the staff’s evaluation of the WLS Units 1 and 2 PTS and Bases (specifically completion of bracketed text).

- **Part 5 Emergency Plan**

Chapter 13 of this report includes the staff's evaluation of the WLS Emergency Plan, including related ITAAC, supporting information such as ETEs, and the applicable offsite State and local emergency plans.

- **Part 6 Limited Work Authorization**

Part 6 of the application is not used and, therefore, has no corresponding staff review.

- **Part 7 Departures Report**

The staff's evaluation of the departures and exemptions in Part 7 is provided in the applicable chapters of this SER. The Table 1-1, below, lists the departures identified in the application and identifies where the evaluation appears in this SER. Several of the departures, as marked, correspond to exemptions requested by the applicant.

Table 1-1. Departures Identified in Part 7 of the COL Application

Description of Departure	Location of Evaluation in this Report
STD DEP 1.1-1. Departure for organization and numbering for the FSAR sections	1.5.4
WLS DEP 1.8-1. Departure correcting an inconsistency in regulatory citation in an interface description	1.4.4
WLS DEP 2.0-1 Lee Site Foundation Response Spectra	3.7
WLS DEP 3.2-1. Departure adding downspouts and downspout screens to the condensate return portion of the Passive Core Cooling System	21.1
WLS DEP 3.8-1. Lee Passive Earth Pressures	3.8
WLS DEP 3.11-1. Departure revising the "Envir. Zone" numbers for Spent Fuel Pool Level instruments	3.11
WLS DEP 6.2-1. Departure revising the ITAAC Acceptance Criteria for the in-containment PXS compartment vents to reflect the current plant configuration.[3]	21.4
WLS DEP 6.3-1. Departure to quantify the term "indefinitely" as used in the AP1000 DCD for maintenance of safe shutdown conditions using the PRHR HX during non-LOCA accidents.	21.1

Description of Departure	Location of Evaluation in this Report
WLS DEP 6.4-1. Departure revising estimated maximum doses to control room operators to meet 10 CFR Part 50, Appendix A, General Design Criterion 19, "Control Room".[3]	21.2
WLS DEP 6.4-2. Departure revising the heat generated in the control room during accident conditions and the conditions for actuating the normal ventilation system supplemental filtration and the emergency ventilation system.[3]	21.3
WLS DEP 7.3-1. Departure modifying the engineered safety features to provide an operating bypass for the boron dilution block to meet the requirements of IEEE 603-1991 in accordance with 10 CFR 50.55a(h), "Protection and safety systems."[3]	21.5
WLS DEP 8.3-1. Departure for Class 1E voltage regulating transformer current limiting features	8.3.2
WLS DEP 18.8-1. Emergency Response Facility Locations	13.3

Part 7 of the COL application, Part B, requests seven exemptions, as listed in Table 1-2.

Table 1-2. Exemption Requests Identified in Part 7 of the COL Application

Description of Exemption	Location of Evaluation in this Report
Exemption from 10 CFR Part 52, Appendix D, Section IV.A.2.a related to COL application organization and numbering	1.5.4
Exemption from the requirements of 10 CFR 70.22(b), 10 CFR 70.32(c), 10 CFR 74.31, 10 CFR 74.41 and 10 CFR 74.51, for SNM Material Control and Accounting Program Description	1.5.4
Exemption from AP1000 DCD Tier 1 Tables 2.2.3-1 and 2.2.3-2 and TS Surveillance Requirement (SR) 3.5.4.7 related to Containment Cooling Changes in regard to Passive Core Cooling System Condensate Return	21.1

[3] These departures include revisions to either AP1000 Tier 1 information or generic Technical Specification (TS) and correspond to exemptions requested by the applicant.

Description of Exemption	Location of Evaluation in this Report
Exemption from AP1000 DCD Tier 1 Subsection 2.7.1 and Tables 2.2.5-1 and 2.2.5-5 and TS Limiting Condition for Operation 3.7.4 and TS SR 3.7.4.1 related to Main Control Room Dose	21.2
Exemption from AP1000 DCD Tier 1 Tables 2.2.5-1, 2.2.5-4, 2.5.2-3 and 2.5.2-4, and TS 3.3.2 and 3.7.6 related to Main Control Room Heatup	21.3
Exemption from AP1000 Tier 1 Table 2.3.9-3 related to Combustible Gas Control in Containment	21.4
Exemption from AP1000 TS Table 3.3.2-1 related to Source Range Neutron Flux Doubling Block Permissive	21.5
Exemption from 10 CFR 52.93(a)(1)[4]	1.5.4

- **Part 8 Security Plan**

The staff's evaluation of the Physical Security Plan, the Training and Qualification Plan, and the Safeguards Contingency Plan is documented separately from this report and is withheld from the public in accordance with 10 CFR 73.21. A non-sensitive summary of the staff's evaluation of those plans is provided in Section 13.6 of this SER.

- **Part 9 Withheld Information**

The staff's evaluation of the withheld information occurs in the context of the specific subject being reviewed and is documented accordingly. A summary of the staff's evaluation of the Mitigative Strategies Description and Plans for loss of large areas of the plant due to explosions is provided in Appendix 19A of this report. The staff's complete evaluation is documented separately from this SER and is withheld from the public in accordance with 10 CFR 2.390.

The staff's evaluation of the WLS Units 1 and 2 Cyber Security Plan is provided in Section 13.8 of this SER.

- **Part 10 Proposed Combined License Conditions (Including ITAAC)**

The staff's evaluation of the proposed COL conditions and ITAAC is provided in the applicable SER chapters. Appendix A identifies the proposed license conditions and ITAAC and the location of the evaluations. Each license condition is sequentially numbered in individual

[4] Part 7 of the WLS COL application does not include an exemption request related to the requirements found in 10 CFR 52.93(a)(1). As discussed in Section 1.5.4 of this report, the staff determined that an exemption from this regulation is necessary.

chapters of this SER. The license conditions and ITAAC are based on the provisions of 10 CFR 52.97, "Issuance of combined license."

- **Part 11 Enclosures**

Part 11 includes enclosures submitted by the applicant in support of the WLS Unis 1 and 2 COL application. Specifically, these enclosures include:

- Part 11A of the WLS COL application regarding the QAPD is documented in Chapter 17 of this SER.
- Part 11B of the WLS COL application regarding mitigative strategies description and plans for loss of large areas of the plant due to explosions is provided in Appendix 19A of the SER. The staff's complete evaluation is documented separately from this SER and is withheld as non-public in accordance with 10 CFR 2.390.
- Part 11C of the application regarding the cyber security plan is in Section 13.8 of this SER.
- Part 11D of the application regarding the SNM, material control and accounting (MC&A) program description is in Section 1.5.5 of this SER.
- Part 11E of the application regarding the new fuel shipping plan is in Section 1.5.5 of this SER.
- Part 11F of the application regarding supplemental information in support of the 10 CFR Part 70 license is in Section 1.5.5 of this SER.

<u>Organization of this SER</u>

The staff's SER is structured as follows:

- The SER adheres to the "finality" afforded to COL applications that incorporate by reference a standard certified design. As such, this report does not repeat any technical evaluation of material incorporated by reference; rather, it points to the corresponding review findings of NUREG-1793 and its supplements. However, the referenced DCD and the WLS COL FSAR are considered in the staff's SER to the extent necessary to ensure that the expected scope of information to be included in a COL application is addressed adequately in either the AP1000 DCD or COL FSAR or in both.
- For sections that were completely incorporated by reference without any supplements or departures, the SER simply points to the AP1000 DCD and related NUREG-1793 and its supplements and confirms that all the relevant review items were addressed in the AP1000 DCD and the staff's evaluation was documented in NUREG-1793 and its supplements.
- For subject matter within the scope of the COL application that supplements or departs from the AP1000 DCD, this SER generally follows a six-section organization as follows:
    - "Introduction" section provides a brief overview of the specific subject matter.
    - "Summary of Application" section identifies whether portions of the review have

received finality and clearly identifies the scope of review for the COL.

- "Regulatory Basis" section identifies the regulatory criteria for the information addressed by the COL application.
- "Technical Evaluation" section focuses on the information addressed by the COL application.
- "Post Combined License Activities" section identifies the proposed license conditions, ITAAC or FSAR information commitments that are post-COL activities.
- "Conclusion" section summarizes how the technical evaluation resulted in a reasonable assurance determination by the staff that the relevant acceptance criteria have been met.

1.4 Staff Review of WLS COL FSAR Chapter 1

1.4.1 Introduction

There are two types of information provided in WLS COL FSAR Chapter 1:

1. General information that enables the reviewer or reader to obtain a basic understanding of the overall facility without having to refer to the subsequent chapters. A review of the remainder of the application can then be completed with a better perspective and recognition of the relative safety significance of each individual item in the overall plant description.
2. Specific information relating to qualifications of the applicant, construction impacts and regulatory considerations that applies throughout the balance of the application (e.g., conformance to the acceptance criteria in NUREG-0800).

This section of the report will identify the information incorporated by reference, summarize all of the new information provided, and document the staff's evaluation of the sections addressing regulatory considerations.

1.4.2 Summary of Application

The information related to COL/SUP items included in WLS COL FSAR, Revision 11, Chapter 1 encompasses the statements of fact or information recommended by RG 1.206. No staff technical evaluation was necessary where the statements were strictly background information. However, where technical evaluation of these COL/SUPs was necessary, the evaluation is not in this section of the report, but in subsequent sections as referenced below.

WLS COL FSAR Section 1.1 Introduction

WLS COL FSAR Section 1.1, incorporates by reference AP1000 DCD Section 1.1, "Introduction," with the following supplements.

- STD SUP 1.1-1

The applicant specified the incorporation of Revision 19 of the Westinghouse AP1000 DCD in all sections of the WLS COL FSAR. Additionally, the applicant incorporated by reference Nuclear Energy Institute (NEI) technical reports as identified in WLS COL FSAR Table 1.6-201.

- WLS SUP 1.1-2

The applicant clarified that the WLS COL FSAR was submitted to NRC by DEC under Section 103 of the *Atomic Energy Act* to construct and operate two nuclear power plants under the provisions of 10 CFR Part 52, Subpart C, "Combined Licenses."

- WLS COL 2.1-1

The applicant provided additional information in WLS COL FSAR Section 2.1-1 to address COL Information Item 2.1-1 (COL Action Item 2.1.1-1). Specifically, WLS Units 1 and 2 are to be located in the eastern portion of Cherokee County in north central South Carolina (SC); approximately 35 miles southwest of Charlotte, North Carolina (NC); approximately 25 miles northeast of Spartanburg, SC; and approximately 7.5 miles southeast of Gaffney, SC. This is a brief introductory summary of the plant location. An expanded discussion of WLS COL 2.1-1 is included in WLS COL FSAR Section 2.1.

- WLS COL 1.1-1

The applicant provided the anticipated schedule for site preparation and construction of two AP1000 reactors at WLS Units 1 and 2 in WLS COL FSAR Table 1.1-203. The applicant committed to provide a site-specific construction plan and startup schedule after issuance of the COL and after a positive decision had been made to construct the plant.

- STD SUP 1.1-6

The applicant identified that, while the WLS COL FSAR generally follows the AP1000 DCD organization and numbering, there were some organization and numbering differences that were adopted, where necessary, to include additional material, such as additional content identified in RG 1.206.

Related to this is STD DEP 1.1-1, "Administrative departure for organization and numbering of the FSAR sections," in WLS COL FSAR Section 1.8 and Part 7 of the WLS COL application. The staff's evaluation of this departure is included in Section 1.5.4 of this SER.

- STD SUP 1.1-3

The applicant provided additional information to describe annotations used in the left hand column of the WLS COL FSAR to identify departures, supplementary information, COL items, and CDI.

- STD SUP 1.1-4

The applicant provided additional information to indicate how proprietary, personal or sensitive information withheld from public disclosure pursuant to 10 CFR 2.390 and RIS 2005-026, "Control of Sensitive Unclassified Nonsafeguards Information Related to Nuclear Power Reactors," is identified in the WLS COL FSAR. Proprietary and sensitive material was provided in Part 9 of the COL application.

- WLS SUP 1.1-5

The applicant provided additional information to identify acronyms and system designations used in the WLS COL FSAR that are in addition to those identified in the AP1000 DCD.

WLS COL FSAR Section 1.2 General Plant Description

WLS COL FSAR Section 1.2, incorporates by reference Section 1.2, "General Plant Description," of the AP1000 DCD, Revision 19 with the following departures and supplements:

- WLS DEP 18.8-1

The applicant provided WLS COL FSAR Figure 1.2-201 to replace AP1000 DCD Figure 1.2-18 to reflect the proposed relocation of the Technical Support Center (TSC) and the Operations Support Center (OSC). The staff's evaluation of the locations of the TSC and OSC is discussed in Section 13.3 of this SER.

- WLS COL 2.1-1; WLS COL 3.3-1; and WLS COL 3.5-1

The applicant provided additional information on the site plan for WLS Units 1 and 2 summarizing the principal structures and facilities, parking areas, roads, and transmission lines. The location and orientation of the power block complex are also described. These COL information items are expanded in other sections of the WLS COL FSAR.[5]

WLS COL FSAR Section 1.3 Comparisons with Similar Facility Designs

WLS COL FSAR Section 1.3 incorporates by reference AP1000 DCD Section 1.3, "Comparisons with Similar Facility Designs" with no departures or supplements.

Section 1.4 Identification of Agents and Contractors

WLS COL FSAR Section 1.4 incorporates by reference AP1000 DCD, Revision 19, Section 1.4, "Identification of Agents and Contractors" with the following supplements:

- WLS SUP 1.4-1

The applicant provided additional information to identify Duke Energy Carolinas LLC (a subsidiary of Duke Energy Corporation) as the agent acting on behalf of itself for WLS Units 1 and 2. Additionally, the applicant identified Duke Energy Carolinas, LLC will own and operate WLS Units 1 and 2.

[5] WLS COL FSAR Table 1.8-202 provides a COL information item index of occurrences in the WLS COL FSAR.

Duke Energy Carolinas LLC is the principal subsidiary of Duke Energy Corporation. Duke Energy Corporation, one of the largest electric power companies in the United States, supplies and delivers energy to approximately 7.2 million U.S. customers. The company has nearly 50,000 megawatts of electric generating capacity in the Midwest, Florida, and the Carolinas.

- WLS SUP 1.4-2

The applicant addressed the contractors participating in the preparation of the COL application in WLS COL FSAR Section 1.4.2.8.

Further, the applicant addressed the specialized consulting firms that will assist with the design, construction, and operation of WLS Units 1 and 2 in proposed license condition 7, included in WLS COL application Part 10.

- WLS SUP 1.4-3

The applicant provided additional information related to specialized consulting firms that assisted in preparing the COL application for WLS.

WLS COL FSAR Section 1.5 Requirements for Further Technical Information

WLS COL FSAR Section 1.5, incorporates by reference AP1000 DCD, Revision 19, Section 1.5, “Requirements for Further Technical Information,” with no departures or supplements. This section of the AP1000 DCD provides information related to testing conducted during the AP600 conceptual design program to provide input into the plant design and to demonstrate the feasibility of unique design features. The AP1000 DCD also describes the analyses performed to show that the AP600 and AP1000 exhibit a similar range of conditions such that the AP600 tests are sufficient to support the AP1000 safety analysis.

WLS COL FSAR Section 1.6 Material Referenced

WLS COL FSAR Section 1.6, incorporates by reference AP1000 DCD, Revision 19, Section 1.6, “Material Referenced,” with the following supplements:

- STD SUP 1.6-1

The applicant provided additional information to identify the technical documents incorporated by reference in the WLS COL FSAR in addition to those technical documents incorporated by reference in the AP1000 DCD.

Section 1.7 Drawings and Other Detailed Information

WLS COL FSAR Section 1.7 incorporates by reference AP1000 DCD, Revision 19, Section 1.7, “Drawings and Other Detailed Information,” with the following supplements:

- WLS SUP 1.7-1

The applicant identified the site-specific system drawings. These are the circulating water system, raw water system, and transmission switchyard and offsite power system diagram.

Section 1.8 Interfaces for Standard Design

WLS COL FSAR Section 1.8, incorporates by reference AP1000 DCD, Revision 19, Section 1.8, “Interfaces for Standard Design,” with the following departures and supplements:

- WLS SUP 1.8-1

The applicant identified departures in WLS COL FSAR Table 1.8-201, “Summary of FSAR Departures from the AP1000 DCD.” The departures are:

- STD DEP 1.1-1 related to numbering and organization of the WLS COL FSAR sections to be consistent with RG 1.206 and NUREG-0800

- WLS DEP 1.8-1 related to the correcting the regulatory citation error in AP1000 DCD.

- WLS DEP 1.8-1 – This departure addresses an error in DCD Table 1.8-1, Item 13.1, that incorrectly references Appendix O of 10 CFR Part 50. This departure is evaluated in Section 1.4.4 of this document.

- WLS DEP 3.2-1 – The condensate return portion of the Passive Core Cooling System has been upgraded to add downspouts and plug fabrication holes in the Polar Crane Girder in order to maximize the return of condensate to the In-Containment Refueling Water Storage Tank and ensure long-term operation of the Passive Residual Heat Removal Heat Exchanger to meet design requirements. This departure is evaluated in Section 21.1 of this document.

- WLS DEP 3.11-1 – DCD Table 3.11-1 “Envir. Zone” numbers for Spent Fuel Pool Level Instruments SFS-JE-LT019A, SFS-JE-LT019B, and SFS-JE-LT019C are revised to be consistent with the location of the instruments. This departure is evaluated in Section 3.11 of this document.

- WLS SUP 1.8-2

The applicant provided a list of the COL information items in the AP1000 DCD. In WLS COL FSAR Table 1.8-202, DEC provides the sections of the application addressing these issues. The table further identifies the AP1000 COL items as an “applicant” item, a “holder” item or both. An applicant item is completely addressed in the application. DEC’s definition of a COL holder item is an item that cannot be resolved prior to issuance of the COL. These items are regulatory commitments of the COL holder and will be completed as specified in the appropriate section of the referenced DCD and their completion is the subject of a COL license condition presented in Part 10 of this COL application.

- WLS SUP 1.8-3

The applicant provided in WLS COL FSAR Table 1.8-203 a list of interface items from the AP1000 DCD and the corresponding WLS COL FSAR section(s) that address those interface items.

- WLS DEP 1.8-1

The applicant provided a departure to address an error in AP1000 DCD Table 1.8-1 listing of plant interfaces where Item 13.1 incorrectly references 10 CFR Part 50, Appendix O. This departure is evaluated in Section 1.4.4 of this report,

Section 1.9 Compliance With Regulatory Criteria

WLS COL FSAR Section 1.9, incorporates by reference AP1000 DCD, Revision 19, Section 1.9, "Compliance with Regulatory Criteria," with the following supplements:

- STD COL 1.9-1 and WLS COL 1.9-1

The applicant provided additional information in STD COL 1.9-1 (corresponding to COL Information Item 1.9-1) and WLS COL 1.9-1 related to regulatory guides cited in the WLS COL FSAR. WLS COL FSAR Table 1.9-201 identifies the regulatory guide revision and provides the WLS COL FSAR cross-references. In addition, WLS COL FSAR Appendix 1AA, "Conformance with Regulatory Guides," was developed by the applicant to supplement the detailed discussion presented in the referenced AP1000 DCD Appendix 1A, "Conformance with Regulatory Guides." Specifically, WLS COL FSAR Appendix 1AA delineates conformance to design aspects as stated in the AP1000 DCD and conformance to programmatic and/or operational issues as presented in the WLS COL FSAR. In certain regulatory guides, design aspects were beyond the scope of the AP1000 DCD and are also presented in the WLS COL FSAR.

- STD COL 1.9-2 and WLS COL 1.9-2

The applicant provided additional information in STD COL 1.9-2 and WLS COL 1.9-2 (corresponding to the first un-numbered COL information item identified at the end of AP1000 DCD Table 1.8-2) related to operational experience. WLS COL FSAR Table 1.9-204 provides a list of Bulletins and Generic Letters (GLs), the appropriate WLS COL FSAR cross-references and whether the subject matter was addressed in the AP1000 DCD.

- STD COL 1.9-3

The applicant provided additional information in STD COL 1.9-3 (related to the second un-numbered COL information item identified at the end of AP1000 DCD Table 1.8-2) related to review of unresolved safety issues and generic safety issues (GSIs). Specifically, WLS COL FSAR Table 1.9-203 lists Three Mile Island (TMI) Action Plan items, Task Action Plan items, New Generic Issues, Human Factors issues, and Chernobyl Issues and states how they were considered in the AP1000 DCD and COL application. In addition, the applicant provided discussion on four new generic issues: Issue 186 related to heavy load drops; Issue 189 related to susceptibility of certain containments to early failure from hydrogen combustion; Issue 191 related to PWR sump performance; and Issue 196 related to the use of Boral in long-term dry storage casks for spent reactor fuel.

- STD SUP 1.9-1

The applicant provided additional information related to conformance with NUREG-0800. Specifically WLS COL FSAR Table 1.9-202 delineates conformance with NUREG-0800 for design aspects as stated in the AP1000 DCD and conformance for subjects beyond the scope of the AP1000 DCD as presented in the WLS COL FSAR.

- STD SUP 1.9-2

The applicant clarified that the severe accident mitigation design alternatives evaluation for the AP1000 DCD Appendix 1B is not incorporated into the WLS COL FSAR; but is addressed in the COL application Environmental Report.

- STD SUP 1.9-3 and WLS SUP 1.9-4

The applicant provided information related to station blackout (SBO) procedures and training for operators to include actions necessary to restore offsite power after 72 hours by addressing alternating current (ac) power restoration and severe weather guidance in accordance with NUMARC-87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors."

WLS COL FSAR Section 1.10 Nuclear Power Plants to Be Operated On Multi-Unit Sites

The applicant provided an assessment of the potential impacts of construction of one unit on SSCs important to safety for an operating unit, in accordance with 10 CFR 52.79(a)(31). This section of the WLS COL FSAR provides an assessment of potential construction activity hazards, SSCs important to safety for the operating unit and related limiting conditions for operation (LCOs) for the operating unit, potentially impacted SSCs and LCOs and applicable managerial and administrative controls to be used to provide assurance that the LCOs for operating units are not exceeded as a result of construction activities at the multi-unit sites.

- STD SUP 1.10-1

The applicant identified this as a new section in the WLS COL application that was not part of the referenced AP1000 DCD.

- WLS SUP 1.10-1

The applicant identified that the power blocks for WLS Units 1 and 2 have a minimum separation of at least 800 feet between plant centerlines. In the standard portion of the application there is a discussion that the primary consideration in setting this separation distance is the space needed to support plant construction via the use of a heavy-lift crane.

License Conditions

- The applicant proposed that the ITAAC identified in the tables in WLS COL application, Part 10, Appendix B be incorporated into the COL.

1.4.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of NRC regulations for the introductory information in WLS COL FSAR Chapter 1 are given in NUREG-0800, Section 1.0.

The applicable regulatory requirements for the introductory information are as follows:

- 10 CFR 50.43(e), as it relates to requirements for approval of applications for a DC, COL, manufacturing license, or operating license that propose nuclear reactor designs that differ significantly from light-water reactor (LWR) designs that were licensed before 1997, or use simplified, inherent, passive, or other innovative means to accomplish their safety functions.

- 10 CFR 52.77, “Contents of applications; general information,” and 10 CFR 52.79, as they relate to general introductory matters.

- 10 CFR 52.79(a)(17), as it relates to compliance with technically relevant positions of the TMI requirements.

- 10 CFR 52.79(a)(20), as it relates to proposed technical resolutions of those unresolved safety issues and medium- and high priority GSIs that are identified in the version of NUREG-0933, “Resolution of Generic Safety Issues (Formerly entitled ‘A Prioritization of Generic Safety Issues’),” current on the date up to 6 months before the docket date of the application and that are technically relevant to the design.

- 10 CFR 52.79(a)(31) regarding nuclear power plants to be operated on multi-unit sites, as it relates to an evaluation of the potential hazards to the SSCs important to safety of operating units resulting from construction activities, as well as a description of the managerial and administrative controls to be used to provide assurance that the LCOs are not exceeded as a result of construction activities at the multi-unit sites.

- 10 CFR 52.79(a)(37), as it relates to the information necessary to demonstrate how operating experience insights have been incorporated into the plant design.

- 10 CFR 52.79(a)(41), as it relates to an evaluation of the application against the applicable NRC review guidance in effect 6 months before the docket date of the application.

- 10 CFR 52.79(d)(2) requires that for a COL referencing a standard DC, the FSAR demonstrate that the interface requirements established for the design under 10 CFR 52.47, “Contents of applications; technical information,” have been met.

- 10 CFR 52.97(a)(1)(iv) regarding technical and financial qualifications.

The related acceptance criteria from NUREG-0800, Chapter 1 are as follows:

- For regulatory considerations, acceptance is based on addressing the regulatory requirements as discussed in FSAR Chapter 1 or in the referenced FSAR section. The NUREG-0800 acceptance criteria associated with the referenced section will be reviewed in the context of that review.

- For performance of new safety features, the information is sufficient to provide reasonable assurance that: (1) these new safety features will perform as predicted in WLS COL FSAR; (2) the effects of system interactions are acceptable; and (3) the applicant provides sufficient data to validate analytical codes. The design qualification testing requirements may be met with either separate effects or integral system tests; prototype tests; or a combination of tests, analyses, and operating experience.

For conformance to regulatory criteria, RG 1.206 states an applicant should perform a similar evaluation for conformance with RGs that were in effect six months prior to the submittal of the COL application.

1.4.4 Technical Evaluation

The staff reviewed WLS COL Section 1 and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[6] The staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to this introduction. The results of the staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff completed its review and concluded that the evaluation performed for the standard content directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this report provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the Bellefonte Nuclear Station (BLN) Units 3 and 4 COL application. Any confirmatory items in the standard content material retain the numbers assigned in the VEGP SER. Confirmatory items that are first identified in this report section have a WLS designation (e.g., WLS Confirmatory Item 1.4-1).

The staff reviewed the information in the WLS COL FSAR:

<u>WLS COL FSAR Sections 1.1, 1.2, 1.3, 1.6, and 1.7</u>

There are no specific NUREG-0800 acceptance criteria related to the general information presented in Sections 1.1, 1.2, 1.3, 1.6, and 1.7, and no specific regulatory findings. The information provides the reader with a basic overview of the nuclear power plant and the construct of the WLS COL FSAR, itself.

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[6] See Section 1.2.2, "Finality of Referenced NRC Approvals" of this report for a discussion of the staff's review related to verification of the scope of information to be included within a COL application that references a DC.

In WLS COL FSAR Section 1.1, WLS COL 1.1-1 the applicant provided an overall anticipated schedule for site preparation and construction of two AP1000 reactors at the Lee Nuclear Site and is shown in WLS COL FSAR Table 1.1-203 and is based on various considerations.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 1.4.4:

> *In a letter dated November 11, 2010, the applicant added a discussion of incorporation of the proprietary information and safeguards information referenced in the AP1000 DCD. This information is included to meet the requirements of 10 CFR Part 52, Appendix D, Section IV.A.3, which indicates the applicant must "include, in the plant specific DCD, the proprietary information and safeguards information referenced in the AP1000 DCD" and, therefore, is acceptable. The incorporation of the above information into a future revision of the VEGP COL FSAR is* ***Confirmatory Item 1.4-1.***
>
> *Resolution of Standard Content Confirmatory Item 1.4-1*
>
> *Confirmatory Item 1.4-1 is an applicant commitment to revise FSAR Section 1.1 to include a discussion of incorporation of the proprietary information and safeguards information referenced in the AP1000 DCD. The staff verified that the VEGP COL FSAR was appropriately revised. As a result, Confirmatory Item 1.4-1 is now closed.*

WLS COL FSAR Section 1.4

- WLS SUP 1.4-1 and WLS SUP 1.4-3

This evaluation is limited to DEC's technical qualification to hold a 10 CFR Part 52 license in accordance with 10 CFR 52.97(a)(1)(iv). The financial qualifications that are also a requirement of 10 CFR 52.97(a)(1)(iv) are evaluated in Section 1.5.1 of this report.

In WLS COL FSAR Section 1.4, DEC provided justification for why it believes it is qualified to hold a 10 CFR Part 52 license. The WLS COL FSAR states that it has over 45 years of experience in the design, construction and operation of nuclear power stations, and currently has seven nuclear operating units that generates over 7000 megawatts of electricity. DEC operates Catawba Units 1 and 2, McGuire Units 1 and 2, and Oconee Units 1, 2, and 3. Since DEC holds 10 CFR Part 50 licenses for nuclear power plants and has demonstrated its ability to build and operate these plants, the staff finds that DEC is qualified to hold a 10 CFR Part 52 license. The staff notes that WLS COL FSAR Section 17.5, discusses the QA program to be implemented at the receipt of the WLS COL. The staff's evaluation of WLS COL FSAR Section 17.5 is discussed in Section 17.5 of this report. Based on DEC's experience with building and operating nuclear power plants and the staff's evaluation of DEC's QA program, the staff finds that DEC is technically qualified to hold a 10 CFR Part 52 license in accordance with 10 CFR 52.97(a)(1)(iv).

- WLS SUP 1.4-2

In WLS SUP 1.4-2 the applicant provided the names of contractors and description of the specialized services provided in the preparation of the COL application.

DEC received support from the following contractors in preparing the COL:

- AMEC Environment & Infrastructure (MACTEC Engineering and Consulting, Inc.)
- Furgo Consultants Inc. (formerly William Lettis & Associates, Inc.)
- Enercon Services, Inc.
- Burns & Roe Enterprises, Inc.
- Chicago Bridge and Iron (Stone & Webster)
- Atkins
- HDR/DTA

The staff finds this acceptable because the applicant identified contractors beyond those identified in the DCD and provided a description of the specialized consulting services rendered in preparation of the COL application.

WLS COL FSAR Section 1.5

10 CFR 50.43(e) requires additional testing or analysis for applications for a DC or COL that propose nuclear reactor designs that differ significantly from LWR designs that were licensed before 1997, or use simplified, inherent, passive, or other innovative means to accomplish their safety functions. This requirement was addressed in the AP1000 DCD and evaluated by the staff in NUREG-1793, Chapter 21, “Testing and Computer Code Evaluation.” The COL application does not include any additional design features that require additional testing.

WLS COL FSAR Section 1.6

There are no specific NUREG-0800 acceptance criteria related to the information presented in Section 1.6 and no specific regulatory findings.

WLS COL FSAR Section 1.7

There are no specific NUREG-0800 acceptance criteria related to the information presented in Section 1.7 and no specific regulatory findings.

WLS COL FSAR Section 1.8

- WLS SUP 1.8-1

As discussed in WLS COL FSAR Section 1.4.2, the applicant identified two departures in WLS COL FSAR Table 1.8-201 from the referenced AP1000 DCD (STD DEP 1.1-1 and WLS DEP 1.8-1). Section 1.3 of this report provides a cross-reference to where these departures are discussed in this report.

- WLS SUP 1.8-2

WLS SUP 1.8-2 includes the same type of information as VEGP SUP 1.8-2. Therefore, the following portion of this technical evaluation section is reproduced from VEGP SER Section 1.4.4:

In Sections 1.3 and 1.4.4 of the BLN SER, the staff identified a standard content ***Open Item 1-2*** *related to the decision regarding which of the BLN COL FSAR commitments, if any, should become a license condition. On January 21, 2010, the NRC issued ISG-15, "Final Interim Staff Guidance on the Post-Combined License Commitments," ESP/DC/COL-ISG-15. This guidance discusses options regarding completion of COL items that cannot be completed until after issuance of the COL. The VEGP applicant identified that certain COL information items cannot be resolved prior to the issuance of a COL. The applicant has identified proposed License Condition 2 in Part 10 of the COL application to ensure these COL items will be completed by the identified implementation milestones through completion of the action identified. The determination that these COL information items cannot be resolved prior to issuance of a COL is discussed in the relevant SER section related to the topic. In addition, using the guidance of ISG-15, the staff has identified certain FSAR commitments in individual sections of this SER and these FSAR commitments are listed in Appendix A.3 of this SER. The staff considers* ***Open Item 1-2*** *is resolved.*

- WLS SUP 1.8-3

AP1000 DCD Table 1.8-1 presents interface items for the AP1000. This section of the AP1000 DCD identifies certain interfaces with the standard design that have to be addressed in accordance with 10 CFR 52.47(a)(1)(vii).[7] As required by 10 CFR 52.79(d)(2), the COL application must demonstrate how these interface items have been met. In the WLS COL FSAR, the applicant provided WLS COL FSAR Table 1.8-203, which explicitly identifies the FSAR location of information addressing the interface items identified in AP1000 DCD Section 1.8. The staff's review of the identified FSAR locations confirmed that interface items are adequately addressed in the WLS COL FSAR. The technical discussions related to specific interface requirements are addressed in related sections of this report (e.g., Sections 8.2.2 and 11.3).

- WLS DEP 1.8-1

This Tier 2 departure, appearing in the WLS COL FSAR Table 1.8-203 listing of AP1000 plant interfaces, corrects an error in AP1000 DCD Table 1.8-1, Item 13.1. This interface addresses the design features that affect plans for coping with emergencies in the operation of the reactor facility or a major portion thereof. The departure changes the incorrect regulatory reference from 10 CFR Part 50, Appendix O, to 10 CFR 52.137(a)(11). In issuing the final rule for 10 CFR Part 52 in the *Federal Register* (FR) (see 72 FR 49352), the requirement relating to providing this interface information was moved from f 10 CFR Part 50, Appendix O to a new location in 10 CFR 52.137 (see 72 FR 49391). Therefore, the staff finds it reasonable that this departure does not require prior NRC approval because it made a technical correction only and did not make a substantive change to the interface item.

[7] Following the update to 10 CFR Part 52 (72 FR 49517), this provision has changed to 10 CFR 52.47(a)(25).

WLS COL FSAR Section 1.9

In this section of the application, the applicant demonstrates conformance to regulatory guides and NUREG-0800 and addresses unresolved safety issues, GSIs, TMI action items, and operating experience.

STD COL 1.9-1 and WLS COL 1.9-1

In comparing VEGP COL FSAR Table 1.9-201 and Appendix 1AA to the respective tables in the WLS COL FSAR, the staff notes that there are several differences. These differences are associated with site-specific information and are reflected in the WLS COL FSAR by a "WLS COL 1.9-1" designation. The staff reviewed the site-specific differences in the respective tables and appendices and determined that the WLS COL 1.9-1 information in these tables was updated consistent with the update provided for the standard information; therefore, the staff considers the standard content open item as it relates to issues associated with the site-specific information resolved.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 1.4.4[8]:

AP1000 COL Information Item

- *STD COL 1.9-1*

Regarding RGs, the applicant provides in BLN COL FSAR Table 1.9-201 a cross-reference between the RG and where it is discussed in the application, and Appendix 1AA, "Conformance with Regulatory Guides," to supplement the detailed discussion presented in Appendix 1A, "Conformance with Regulatory Guides," of the referenced DCD. The technical discussions related to this appendix are addressed in the related technical sections of the BLN COL FSAR. In addition, BLN COL FSAR Table 1.9-201 provides a listing of all RGs, the specific revision, and provides BLN COL FSAR and DCD cross-references.

The staff issued three RAIs associated with how the RG information in Table 1.9-201 and Appendix 1AA of the BLN COL FSAR is presented. In addition, there were two specific RAIs associated with how an individual RG is discussed in Table 1.9-201 and Appendix 1AA. A description of the RAIs and their responses follows.

RAI 1-5

In RAI 1-5, the staff noted that BLN COL FSAR Appendix 1AA lists the later version of the RG when compared with DCD Table 1.9-1 but in some cases does not discuss compliance with the later version. In other cases, exceptions to the RG were identified but not justified.

[8] The text reproduced from VEGP SER Section 1.4.4 is unaltered, but is presented in sequential order of the COL and SUP items.

<u>RAI 1-7</u>

In RAI 1-7, the staff noted that not all RGs listed in Appendix 1AA provided a cross-reference to where they were discussed in accordance with the guidance in Section 1 of NUREG-0800.

<u>RAI 1-11</u>

In RAI 1-11, the staff noted that the information that TVA provided in response to RAIs 1-5 and 1-7 conflicted with information that TVA provided in response to another RAI. TVA was requested to reconcile these differences.

<u>RAIs 1-1 and 1-10</u>

These RAIs are associated with specific RGs and RAI 1-1 and RAI 1-10 are evaluated in Chapters 13 and 12, of this SER, respectively.

In TVA's response to RAIs 1-5 and 1-7, TVA committed to make changes to BLN COL FSAR Table 1.9-201 and Appendix 1AA to:

- *Add an additional statement to Appendix 1AA that specifically addresses the later version of the RG.*
- *Revise BLN COL FSAR Sections 1.9.1.1, 1.9.1.2, 1.9.1.3, and 1.9.1.4, to reflect that one method of identifying and justifying an alternative to an RG is the use of previous revisions of the RG for design aspects as stated in the DCD in order to preserve the finality of the certified design.*
- *Revise BLN COL FSAR Table 1.9-201 to address the RG listed in Appendix 1AA, thereby providing a more complete cross reference of where each RG is discussed in the COL application.*

In response to RAI 1-11, TVA committed to revising BLN COL FSAR Table 1.9-201 and Appendix 1AA to ensure that they are consistent with commitments made in other RAI responses.

The staff's evaluation of the RGs is addressed in Chapters 2 through 19 of this SER as needed. At a minimum the NRC staff's FSER sections will discuss any RG that involves an exception.

The staff finds TVA's responses to RAIs 1-5 and 1-7 acceptable. However, the staff notes that BLN COL FSAR Table 1.9-201 and Appendix 1AA will most likely need additional changes based on the staff's evaluation of the RGs in this SER and TVA's response to RAI 1-11. The NRC staff is still evaluating TVA's response to RAI 1-11 and has not yet made a determination of whether the response is acceptable. This is Open Item 1.4-2. The updating of BLN COL FSAR Table 1.9-201 to reflect changes committed to by TVA in response to RAI 1-11 and the updating of this information to reflect TVA's commitments in other RAI responses is Confirmatory Item 1.4-2.

Resolution of Standard Content Confirmatory Item 1.4-2

The NRC staff verified that VEGP COL FSAR Table 1.9-201 was updated to provide an acceptable cross reference of where each RG is discussed in the COL application. As a result, Confirmatory Item 1.4-2 is resolved for VEGP.

Resolution of Standard Content Open Item 1.4-2

In a letter dated September 21, 2009, the VEGP applicant provided clarification to a previously submitted response dated January 27, 2009 from the BLN applicant. Specifically, the applicant proposed to revise the discussion in the "General comment" portion related to preserving the finality of the certified design in VEGP COL FSAR Sections 1.9.1.1, 1.9.1.2, 1.9.1.3, 1.9.1.4 and Appendix 1AA Note (b); to clarify in VEGP COL FSAR Section 17.5 the "DCD scope" and the "remaining scope" discussion for QA-related RGs (including RG 1.28; RG 1.30, "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment (Safety Guide 30)"; RG 1.33, "Quality Assurance Program Requirements (Operation)," Revision 2; RG 1.38, "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants," Revision 2; RG 1.39, "Housekeeping Requirements for Water-Cooled Nuclear Power Plants," Revision 2; RG 1.94, "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants," Revision 1; and RG 1.116, "Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems"). In addition, the applicant proposed to revise the VEGP COL FSAR, Appendix 1AA Note (c) to clarify the purpose of a "General" entry under the column labeled "Section Criteria" discussion. It is stated that a "Criteria Section" entry of "General" indicates a scope for the conformance statement of "all regulatory guide positions related to programmatic and/or operational aspects." Thus an associated conformance statement of "Conforms" indicates that the applicant "complies with all regulatory guide positions related to programmatic and or/or operational aspects." The proposed clarifications clearly provide the scope of conformance to the RGs and, therefore, they are acceptable. The staff verified that the VEGP COL FSAR was updated to reflect the above. The staff considers Open Item 1.4-2 resolved for VEGP.

<u>STD COL 1.9-2, WLS COL 1.9-2, STD SUP 1.9-3, and STD SUP 1.9-1</u>

The following portion of this technical evaluation section is reproduced from VEGP SER Section 1.4.4:

- *STD COL 1.9-2 (related to the first un-numbered COL information item identified at the end of DCD Table 1.8-2)*

Regarding demonstration of operating experience from Bulletins and GLs, as required by 10 CFR 52.79(a)(37), BLN COL FSAR Table 1.9-204 provides a list of Bulletins and GLs, the appropriate BLN COL FSAR cross-references, and whether the subject matter was addressed in the DCD. The technical discussions related to the specific safety issues are addressed in the related *sections of the BLN COL FSAR and are addressed in Chapters 2 through 19 of this SER as needed.*

The evaluation of GSI 163, "Multiple Steam Generator Tube Leakage," is described below because otherwise its evaluation would be spread across several SER chapters.

GSI 163 identified a safety concern associated with the potential multiple steam generator (SG) tube leaks triggered by a main steam line break outside containment that cannot be isolated. The issue was evaluated as part of the AP1000 DCD review and was resolved for the AP1000 design. The evaluation was documented in NUREG-1793, Chapter 20. The evaluation states in part the following:

> *The staff agrees that the issue should be closed for the AP1000 design. Issue 163 concerns the possibility that a multiple steam generator tube rupture (SGTR), resulting from a main steam line break and degraded SG tubes, could result in core damage due to depletion of the reactor coolant and safety injection fluid in the refueling water storage tank. For the AP1000 design, an SGTR is mitigated using the passive core cooling system, initially through the passive residual heat removal heat exchanger, and the core makeup tanks (CMTs). After the CMTs drain to the low level to actuate the automatic depressurization system, the reactor coolant depressurization would result in gravity injection from the in containment refueling water storage tank (IRWST), and eventually from the containment recirculation. The scenario that the safety injection from the refueling water storage tank, which is outside the containment in the existing plants, will be depleted to result in core damage is not likely for the AP1000 design because the IRWST and containment recirculation will continue to provide core cooling.*

Since the resolution of Issue 163 is an ongoing NRC effort, any future requirements for the resolution of this issue will be required of the COL applicant, if applicable to the AP1000 design.

Subsequent to the original issuance of NUREG-1793, GSI 163 was closed via a July 16, 2009, memorandum. In the safety evaluation accompanying the closure of the issue, the following is stated.

> *the staff concludes that the technical specification requirements relating to SG tube integrity provide reasonable assurance that all tubes will exhibit acceptable structural margins against burst or rupture during normal operation and DBAs (including MSLB [main steam line break]), and that leakage from one or multiple tubes under DBAs will be limited to very small amounts, consistent with the applicable regulations for offsite and control room dose.*

Therefore, in addition to the unique design features of the AP1000 cited in NUREG-1793 and its supplements as a basis for closure of the issue, the staff *notes that for PWR designs in general the issue is resolved based on the technical specification requirements. The staff discusses these technical specification requirements in Section 5.4, "Component and Subsystem Design," of this SER. Based on the evaluation in NUREG-1793 and its supplements, and based on the staff's evaluation of the SG tube surveillance program in Section 5.4 of this SER, the staff considers GSI 163 resolved for VEGP*.

- *STD COL 1.9-3*

Regarding consideration of new and generic safety issues as required by 10 CFR 52.79(a)(17) and 10 CFR 52.79(a)(20), BLN COL FSAR Table 1.9-203, provides a listing of the TMI Action Plan items, Task Action Plan items, New Generic Issues, Human Factors issues, and Chernobyl Issues and states how they were considered in the DCD and COL application. The technical discussions related to the specific safety issues are addressed in the related sections of the BLN COL FSAR.

In addition, the applicant provided discussion of four new generic issues: Issue 186 related to heavy load drops; Issue 189 related to susceptibility of certain containments to early failure from hydrogen combustion; Issue 191 related to PWR sump performance; and Issue 196 related to the use of Boral in long-term dry storage casks for spent reactor fuel.

The applicant identified that neither Issue 189 nor Issue 196 is applicable to the design or application and that therefore neither is addressed in the BLN COL FSAR. Issue 186 states that there are not any planned heavy load lifts outside those described in the DCD; nonetheless, special procedures to address heavy loads are discussed in Subsection 9.1.5.3. Related to Issue 191, the applicant provided a reference to the protective coatings program and containment cleanliness program in Subsections 6.1.2.1.6 and 6.3.8.1 of the BLN COL FSAR, respectively.

Issue 186 and Issue 196 are evaluated in Chapter 9 of this SER. Issues 189 and 191 are evaluated in Chapter 6 of this SER.

- *STD SUP 1.9-1*

Regarding conformance with regulatory review criteria as required by 10 CFR 52.79(a)(41), BLN COL FSAR Table 1.9-202 provides the applicant's review of conformance with the acceptance criteria of NUREG-0800. The technical discussions related to the specific acceptance criteria of NUREG-0800 are addressed in the related sections of the BLN COL FSAR and addressed in Chapters 2 through 19 of this SER as needed.

<u>STD SUP 1.9-2</u>

The applicant clarified that the severe accident mitigation design alternatives evaluation for the AP1000 in Appendix 1B to the AP1000 DCD is not incorporated into the WLS COL FSAR; but is addressed in the WLS COL Environmental Report. The staff reviewed this information as part of its development of the Final Environmental Impact Statement. Therefore, no further evaluation is needed for STD SUP 1.9-2.

STD SUP 1.9-3 and WLS SUP 1.9-4

The following portion of this technical evaluation section is reproduced from of VEGP SER Section 1.4.4:

- *STD SUP 1.9-3*

This COL supplemental item is addressed as VEGP SUP 8.1-2 [WLS SUP 8.1-2] *in SER Section 8.1*.

WLS COL FSAR Section 1.10

In this section of the application, the applicant provided an assessment of the potential hazards due to construction of one unit on SSCs important to safety for an operating unit, in accordance with 10 CFR 52.79(a)(31).

STD SUP 1.10-1

The following portion of this technical evaluation section is reproduced from VEGP SER Section 1.4.4:

- *STD SUP 1.10-1*

The NRC staff reviewed the information in BLN COL FSAR Table 1.10-201, identifying the potential hazards from construction activities, BLN COL FSAR Table 1.10-202 that cross-references the construction hazard with the impacted SSCs, and BLN COL FSAR Table 1.10-203, identifying the specific managerial and administrative controls to preclude or mitigate the construction hazard. There is the potential that review of other areas of the application could impact the hazards and management programs identified in the Bellefonte application. For example, site runoff from construction of Unit 4, if not properly controlled, could impact the operation of Unit 3. Site runoff is evaluated in Section 2.4 of this report. The staff has not yet completed its review of this application against the requirements of 10 CFR 52.79(a)(31). This is part of Open Item 1.4-3.

In the application, TVA stated that controls within Section 1.10 of the FSAR are not required unless there is an operating unit on the site. To clarify this FSAR commitment, the staff requests TVA to revise the application to positively state these programs will be in place when there is an operating unit on the site. This is Open Item 1.4-4.

Resolution of Standard Content Open Item 1.4-4

In a letter dated July 29, 2009, the applicant proposed to revise VEGP COL FSAR Section 1.10.3 to positively state that these programs will be in place when there is an operating unit on the site. The staff verified that the VEGP COL FSAR was appropriately updated to include the above. As a result, Open Item 1.4-4 is resolved.

WLS SUP 1.10-1

The supplemental information states that the power blocks for WLS Units 1 and 2 have a minimum separation of at least 800 feet between plant centerlines and notes that new units SSCs important to safety are described in WLS COL FSAR Chapter 3 and the LCOs for WLS Units 1 and 2 are identified in Part 4 of the WLS COL application. In the standard portion of WLS COL FSAR Section 1.10, there is a discussion that the primary consideration in setting the 800-ft separation distance is the space needed to support plant construction via the use of a heavy-lift crane.

The site-specific supplemental information is provided to supplement the standard information above and provides with specificity the location of the SSCs and LCOs required by 10 CFR 52.79(a)(31). The staff's review of this SUP item is included in the resolution of Open Item 1.4-3.

STD SUP 1.10-1

The following portion of this technical evaluation section is reproduced from of VEGP SER Section 1.4.4:

> *Resolution of Standard Content Open Item 1.4-3*
>
> *A new draft ISG-22 has been issued to assist the staff with the evaluation of COL applicants' compliance with the requirements of 10 CFR 52.79(a)(31). The above draft ISG document was made available to the public including the The regulation at 10 CFR 52.79(a)(31) requires, in part, that applicants for a COL intending to construct and operate new nuclear power plants on multi-unit sites provide an evaluation of the potential hazards to the SSCs important to safety for operating units resulting from construction activities on the new units. The requirement in 10 CFR 52.79(a)(31) can be viewed as having two subparts: applicant and was discussed at a public meeting on August 26, 2010.*
>
> 1. *The COL applicant must evaluate the potential hazards from constructing new plants on SSCs important to safety for existing operating plants that are located at the site.*
>
> 2. *The COL applicant must evaluate the potential hazards from constructing new plants on SSCs important to safety for newly constructed plants that begin operation at the site*
>
> *The interim guidance recommends that the applicant provide a construction impact evaluation plan that includes:*
>
> - *A discussion of the construction activity identification process and the impact evaluation criteria used to identify and evaluate the construction activities that may pose potential hazards to the SSCs important to safety for operating unit(s).*
>
> - *A table of those construction activities and the potential hazards that are identified using that construction impact evaluation plan, the SSCs important to safety for the operating unit potentially impacted by the construction activity, and expected mitigation method.*

- *Identification of the managerial and administrative controls, such as proposed license conditions that may involve construction schedule constraints or other restrictions on construction activities, that are credited to preclude and/or mitigate the impacts of potential construction hazards to the SSCs important to safety for the operating unit(s).*

- *A discussion of the process for communications and interactions planned and credited between the construction organization and the operations organization to ensure appropriate coordination and authorization of construction activities and implementation of the prevention or mitigation activities as necessary.*

- *A memorandum of understanding or agreement (MOU or MOA) between the COL applicant and the operating unit(s) licensee as a mechanism for communications, interactions, and coordination to manage the impact of the construction activities.*

- *An implementation schedule corresponding to construction tasks or milestones to ensure the plan is reviewed on a recurring basis and maintained current as construction progresses.*

The staff reviewed the VEGP COL FSAR Section 1.10, which provides information to address compliance with 10 CFR 52.79(a)(31). In order to complete the staff's review, in RAI 1.5-2, the staff requested that the applicant provide a construction impact evaluation plan that includes:

- *A discussion of the process for communications and interactions planned and credited between the construction organization and the operations organization to ensure appropriate coordination and authorization of construction activities and implementation of the prevention or mitigation activities as necessary.*

- *A memorandum of understanding or agreement (MOU or MOA) between the COL applicant and the operating unit(s) licensee as a mechanism for communications, interactions, and coordination to manage the impact of the construction activities.*

- *An implementation schedule corresponding to construction tasks or milestones to ensure the plan is reviewed on a recurring basis and maintained current as construction progresses.*

In addition, the applicant was requested to identify the managerial and administrative controls (VEGP COL FSAR Table 1.10-203) that are credited to preclude and/or mitigate the impacts of potential construction hazards to the SSCs important to safety for the operating units (VEGP Units 1 and 2).

In a letter dated November 2, 2010, the applicant stated:

- *VEGP COL FSAR Sections 1.10.2 and 13AA will be revised to include the discussion of the process for communications and interactions planned* *and credited between the construction organization and the operations organization.*

- *The COL applicant and the operating unit(s) licensee are the same entity, thus, no MOU or MOA is considered necessary.*

- *VEGP COL FSAR Sections 1.10.3 and 13AA will be revised to include the discussion of the implementation schedule corresponding to construction tasks or milestones.*

- *VEGP COL FSAR will be revised to indicate that managerial and administrative controls are developed and implemented as work progresses on site. These controls are intended to preclude and/or mitigate the impacts of potential construction hazards to the SSCs important to safety for the operating units.*

The proposed changes to the VEGP COL FSAR meet the draft guidance of ISG-22 and, therefore, meet the requirements of 10 CFR 50.79(a)(31). The incorporation of the above proposed changes into a future revision of the VEGP COL FSAR is ***Confirmatory Item 1.4-2****.*

Resolution of Standard Content Confirmatory Item 1.4-2

Confirmatory Item 1.4-2 is an applicant commitment to revise FSAR Sections 1.10.2 and 1.10.3 and Appendix 13A to address guidance included in ISG-22. The staff verified that the VEGP COL FSAR was appropriately revised. As a result, Confirmatory Item 1.4-2 is now closed.

License Conditions

- *Part 10, License Condition 1, ITAAC*

The applicant proposed that the ITAAC identified in the tables in Appendix B of Part 10 of the VEGP COL application be incorporated into the COL. The proposed license condition also states that after the Commission has made the finding required by 10 CFR 52.103(g), "Operation under a combined license," the ITAAC do not constitute regulatory requirements; except for specific ITAAC, which are subject to a hearing under 10 CFR 52.103(a), their expiration will occur upon final Commission action in such proceeding.

The ITAAC identified in tables in Appendix B of Part 10 of the VEGP COL application are evaluated throughout this SER. The remaining text of the proposed license condition is already covered by regulatory requirements of 10 CFR 52.103(h). Therefore, there is no need for a license condition.

1.4.5 Post Combined License Activities

There are no post-COL activities related to this section.

1.4.6 Conclusion

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to principal review matters, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

1.5 Additional Regulatory Considerations

1.5.1 10 CFR 52.97(a)(1)(iv) Applicant Financial Qualifications and Evaluation of Financial Qualification in accordance with 10 CFR 50.33

BACKGROUND:

Duke Energy Carolinas, LLC

According to the COL application, Duke Energy Corporation, the holding company of Duke Energy Carolinas LLC (Duke), is one of the largest electric holding companies in the United States. Through its regulated electric and gas utility operating companies, Duke Energy Corporation operates more than 36,000 MW of electric generation; over 75 percent of which is subject to cost of service ratemaking.

Duke is a wholly owned subsidiary of Duke Energy Corporation and is a limited liability company duly organized and existing under the laws of the State of North Carolina. Duke is engaged in the business of generating, transmitting, distributing and selling electric power and energy.

Duke owns and operates regulated electrical facilities, including seven (7) nuclear units licensed by the NRC, as well as electrical distribution and transmission facilities. Lee Units 1 and 2 (Lee 1 and 2) will be used to produce electricity for sale.

REGULATORY EVALUATION:

The applicant's request for the NRC to issue two combined licenses under Section 103 of the Atomic Energy Act of 1954, as amended, for construction and operation is subject to, among other things, the requirements of the Atomic Energy Act of 1954, as amended; Subpart C to 10 CFR Part 52, 10 CFR Part 50 and 10 CFR Part 140.

In its review, the NRC staff used guidance in NUREG-1577, "Standard Review Plan on Power Reactor Licensee Financial Qualifications and Decommissioning Funding Assurance," Revision 1, issued February 1999, to evaluate the financial qualifications of the applicant to construct, operate, and decommission the proposed facility.

In addressing foreign ownership, control, or domination (FOCD), the NRC staff used guidance in the Standard Review Plan (SRP), "Foreign Ownership, Control, and Domination of applicants for Reactor Licenses, "dated June 1999 (SRP on FOCD) to determine whether the applicant is owned, controlled or dominated by an alien, a foreign corporation, or a foreign government. The

NRC published the SRP on FOCD in the *Federal Register* on September 28, 1999 (64 FR 52357-52359).

The staff also used guidance in NUREG-1307, Revision 15, "Report on Waste Burial Charges: Changes in Decommissioning Waste Disposal Costs at Low-Level Waste Burial Facilities," to independently validate the licensee's calculation of the minimum funding needed for decommissioning.

The safety evaluation documents the staff's review and analysis of financial qualifications, decommissioning funding assurance, FOCD, and nuclear insurance and indemnity. In addition, this safety evaluation contains proprietary information that is withheld from public disclosure per 10 CFR 2.390 as commercially sensitive.

FINANCIAL QUALIFICATIONS:

Pursuant to 10 CFR 52.77, the application must include all of the information required by 10 CFR 50.33.

Construction:

Pursuant to 10 CFR 50.33(f)(1):

> [T]he applicant[s] shall submit information that demonstrates that the applicant[s] possess or [have] reasonable assurance of obtaining the funds necessary to cover estimated construction costs and related fuel cycle costs. The applicant[s] shall submit estimates of the total construction costs of the facility and related fuel cycle costs, and shall indicate the source(s) of funds to cover these costs.

Under 10 CFR Part 50, Appendix C, "A guide for the Financial Data and Related Information Required To Establish Financial Qualifications for Construction Permits and Combined Licenses," Section I.A.1:

> [E]ach applicant's estimate of the total cost of the proposed facility should be broken down as follows and be accompanied by a statement describing the bases from which the estimate is derived:
>
> (a) Total nuclear production plant costs; [and]
>
> (b) Transmission, distribution, and general plant costs; [and]
>
> (c) Nuclear fuel inventory cost for first core.
>
> If the fuel is to be acquired by lease or other arrangement than purchase, the application should so state. The items to be included in these categories should be the same as those defined in the applicable electric plant and nuclear fuel inventory accounts prescribed by the Federal Energy Regulatory Commission [FERC] or an explanation given as to any departure from therefrom.

In accordance with 10 CFR 50.33(f) and 10 CFR Part 50, Appendix C, the projected overnight costs for the construction of two AP1000 nuclear units at the Lee site are outlined below.

PROJECTED PROJECT COST
WILLIAM STATES LEE III NUCLEAR STATION, UNITS 1 & 2
(Combined, in millions 2015 $)[9]

	Unit 1 and Unit 2	Common	Total
Total Nuclear Production Plant Costs......	[[		]]
Transmission, Distribution...... & General Plant Costs	[[		]]
Nuclear Fuel Inventory & Cost for First Core.........	[[		]]
TOTAL (OVERNIGHT COST)	[[		]]

The construction cost estimate is expressed in terms of "overnight cost," which is a term commonly used in describing the cost of large capital projects[10]. The applicant calculated combined Unit 1 and Unit 2 cost estimates for plant construction. According to the COL application, the assumed construction period for Unit 1 is from 2019 to 2024 and for Unit 2, 2020 to 2025. Lee Units 1 and 2 is expected to operate at an estimated combined gross electrical power output of approximately 2234 MWe. Therefore, the total overnight cost, including fuel costs as described above, is [[ ]] million. This is approximately [[ ]]/kWe installed. As stated in the application, in part, estimated plant costs are informed by project pricing from the Westinghouse Electric Corporation and Chicago Bridge and Iron (WEC/CB&I) consortium (escalated to 2015 dollars); evaluation of owner's costs including costs for transmission system ties and upgrades; and contingency costs for construction. In consideration of the information provided in the application and as summarized above, the NRC staff finds the applicant's AP1000 overnight construction cost estimate to be a reasonable projection based on a number of studies[11] that have been conducted by governmental agencies, universities and other entities, and is consistent with the publicly available cost estimates of other U.S. AP1000 projects. In particular, the U.S. Energy Information Administration's (EIA) June 2012 report, "Annual Energy Outlook 2012 with Projections to 2035," (DOE/EIA-0383(2012)), states that "…the overnight capital costs associated with building a nuclear power plant planned in 2012 are assumed to be $5,335 per kilowatt of capacity…" The staff applied a conservative annual adjustment factor ranging from 3% to 10% to the EIA overnight capital cost estimate to account for inflation beyond 2012, and determined that the EIA projected 2015 overnight cost would range from $5,830 to $7,101/kWe installed. The construction cost estimate is expressed in terms of "overnight cost," which is a term commonly used in describing

[9] Commercially sensitive data. The data in brackets cannot be released to the public.

[10] Overnight cost is the cost of a construction project if no interest was incurred during construction, as if the project was completed "overnight." An alternate definition is: the present value cost that would have to be paid as a lump sum up front to completely pay for a construction project. The overnight cost is frequently used when describing power plants.

[11] See, e.g., the 2003 the Massachusetts Institute of Technology (MIT) interdisciplinary study entitled The Future of Nuclear Power; the U.S. Department of Energy's Energy Information Agency (EIA) 2012 Annual Energy Outlook (AEO); the Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development 2005 update on Projected Costs of Generating Electricity; and the Keystone Center 2007 report entitled Nuclear Power Joint Fact-Finding.

the cost of large capital projects[12]. The applicant's overnight cost estimate of [[ ]]/kWe installed is slightly less than, but in line with, the most recent EIA 2012 range of overnight costs as adjusted for inflation. Accordingly, the NRC staff finds Duke's overnight cost estimate to be reasonable as presented in its COL application.

Sources of Construction Funds:

Pursuant to 10 CFR, Part 50, Appendix C, I.A.2:

> [t]he application should include a brief statement of the applicant's general financial plan for financing the cost of the facility, identifying the source or sources upon which the applicant relies for the necessary construction funds, e.g., internal sources such as undistributed earnings and depreciation accruals, or external sources such as borrowings.

Duke's Source of Construction Funds

According to the COL application, Duke intends to construct Lee as a regulated asset eligible for cost recovery under North Carolina Statute, G.S. 62-110.1, 62-110.7, 62-133(b) and South Carolina Title 58, Chapter 33, Sections 58-33-220, 58-33-225, 58-33-270, 58-33-280 outlining the recovery for reasonable and prudently incurred costs for a nuclear generation construction project

Duke expects to finance this project through a mixture of internally generated cash and external funding. The three primary sources are cash from operations, debt issued by Duke Energy Carolinas, LLC, and retained earnings and equity infused by its parent, Duke Energy Corporation, as needed, to balance the utility's regulated capital structure to a targeted level. Further, the applicant stated that it may borrow from Duke Energy Corporation to fund a portion of its capital requirements until such time as it is opportune to issue long-term debt securities (bonds or debentures). The staff concludes that Duke Energy Carolinas, LLC and Duke Energy Corporation have sufficient financing capacity to fund this project from the following sources: internally generated operating cash flows, commercial paper and bank facilities, and long-term and equity capital markets.

<u>Financial Statements</u>

Pursuant to 10 CFR, Part 50, Appendix C, I.A.3:

> [t]he application should also include the applicant's latest published annual financial report, together with any current interim financial statements that are pertinent. If an annual financial report is not published, the balance sheet and operating statement covering the latest complete accounting year together with all pertinent notes thereto and certification by a public accountant should be furnished.

<u>Duke Energy Corporation Financial Statements</u>

Duke Energy Corporation files financial statements with the U.S. Securities and Exchange Commission (SEC) at the time the COL application was submitted. Duke submitted, pursuant to

Appendix C.I.A 3, to 10 CFR Part 50, annual financial statements. The NRC staff did not identify anything in Duke's financial statements that warranted further inquiry.

Operating License

Pursuant to 10 CFR 50.33(f)(3),

> If the application is for a combined license under Subpart C of 10 CFR Part 52 of this chapter, the applicant shall submit the information described in paragraphs (f)(1) and (f)(2) of this section.

10 CFR 50.33(f) provides that each application shall state:

> [e]xcept for an electric utility applicant for a license to operate a utilization facility of the type described in 10 CFR 50.21(b) or 50.22, information sufficient to demonstrate to the Commission the financial qualification[s] of the applicant to carry out, in accordance with the regulations in this chapter, the activities for which the permit or license is sought.

10 CFR 50.2, "Definitions" states, in part, that an electric utility is:

> [a]ny 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.

The applicant is an electric utility as defined in 10 CFR 50.2. The applicant generates and distributes electricity and recovers the cost of this electricity through cost-of-service based rates established by the North Carolina Public Utility Commission, South Carolina Public Service Commission, and FERC.

Based on the foregoing, the staff finds that Duke is an electric utility and not subject to a financial qualifications review pursuant to 10 CFR 50.33(f)(2).

DECOMMISSIONING FUNDING ASSURANCE:

Regulatory Requirements:

Pursuant to 10 CFR 50.33(k)(1):

> [A]n application for [a …] combined license for a production or utilization facility, information in the form of a report, as described in 10 CFR 50.75, indicating how reasonable assurance will be available to decommission the facility.

Under 10 CFR 50.75, "Reporting and recordkeeping for decommissioning planning," the report must include a certification that the applicant will provide financial assurance for decommissioning using one or more of the methods allowed under the regulation at 10 CFR 50.75(e) no later than 30 days after the Commission publishes notice in the *Federal Register* under 10 CFR 52.103(a). In addition, the amount of the financial assurance may be more, but not less, than the amount stated in the table in 10 CFR 50.75(c)(1), as adjusted under 10 CFR 50.75(c)(2). Under 10 CFR 50.75(b)(4), a COL applicant need not obtain a financial instrument appropriate to the method to be used or submit a copy of the instrument to the

Commission. (Once the COL is granted, the holder of a COL must submit an instrument as provided in 10 CFR 50.75(e)(3)).

Additionally, the staff used the guidance in NUREG-1577, Rev. 1, in its review of the Lee 1 and 2 COL application.

Decommissioning Funding Estimate

The proposed plant is a simplified passive advanced light water reactor plant that is being licensed in accordance with the Westinghouse AP1000 certified design, as documented in the referenced DCD and its supplements**.** This design has a per unit thermal power rating of 3400 MWt. In its April 11, 2016 submittal, the applicant stated that it will provide decommissioning funding assurance in an amount of $517 million (2015 dollars) per unit. This value was derived using the methodology delineated in 10 CFR 50.75(c), and guidance in NUREG-1307, Revision 15. The NRC staff independently calculated the minimum funding needed for Lee Units 1 and 2 using the regulations and guidance described above, and obtained results similar to Duke's. Accordingly, the staff finds that the amount provided by Duke is acceptable.

Decommissioning Funding Mechanism

Pursuant to 10 CFR 50.75(b), a reactor licensee is required to provide decommissioning funding assurance by one or more of the methods described in 10 CFR 50.75(e), as determined to be acceptable to the NRC. According to the COL application, Duke has chosen to provide decommissioning funding assurance for Lee 1 and 2 using an external sinking fund. Duke's external sinking fund will be in the form of a trust; will be established in writing and maintained at all times in the United States with an entity that is an appropriate State or Federal government agency, or an entity whose operations are regulated and examined by a State or Federal agency; and will include the provisions required by 10 CFR 50.75(h)(2). The staff finds that Duke's use of an external sinking fund is acceptable since it will recover, either directly or indirectly, the estimated total cost of decommissioning through rates established by "cost of service" or similar ratemaking regulation. Therefore, the staff finds this method to be acceptable since it meets the requirements in 10 CFR 50.75(e)(1)(ii).

Certification Updates, Financial Instruments, and Annual Adjustment

According to the application, two years and one year before the scheduled date for initial loading of fuel, Duke will submit a report updating this certification in accordance with 10 CFR 50.75(e)(3) and providing copies of the financial instruments to be used. In addition, no later than 30 days after the NRC publishes the notice in the Federal Register under 10 CFR 52.103(a), Duke will submit a report containing a certification that the financial assurance for decommissioning is being provided in an amount specified in the most recent updated certification and will include a copy of the executed financial agreements obtained to satisfy the requirements of 10 CFR 50.75(e). Thereafter, the decommissioning funding amount will be adjusted annually using a rate at least equal to that stated in 10 CFR 50.75(c)(2). The staff finds Duke's proposed plan as described above and in the application to be reasonable.

ANTITRUST REVIEW:

The Energy Policy Act of 2005 (EPAct) removed the antitrust review authority contained in section 105.c of the Atomic Energy Act of 1954, as amended (AEA), regarding license applications for production or utilization facilities submitted under sections 103 or 104b of the AEA after the date of enactment of the EPAct. Accordingly, the NRC is not authorized to conduct an antitrust review in connection with this combined license application.

FOREIGN OWNERSHIP, CONTROL, or DOMINATION:

Section 103 of the AEA, in relevant part, prohibits the Commission from issuing a license for a nuclear power plant to:

> an alien or any corporation or other entity if the Commission knows or has reason to believe it is owned, controlled, or dominated by an alien, a foreign corporation or a foreign government.

10 CFR Part 50.38 "Ineligibility of certain applicants," is the regulatory provision that implements this statutory prohibition.

The NRC staff reviewed the application pursuant the guidance provided in the SRP on FOCD to determine whether the applicant is owned controlled, or dominated by an alien, a foreign corporation, or a foreign government.

Duke Foreign Ownership, Control, or Domination

According to the application, Duke is not owned, controlled or dominated by any alien, foreign corporation or foreign government. Duke is a limited liability company organized under the laws of the State of North Carolina and whose principal place of business is Charlotte, North Carolina. Duke is wholly owned by Duke Energy Corporation, a Delaware corporation whose principal place of business is Charlotte, North Carolina. The shares of common stock of Duke Energy Corporation are publicly traded and widely held. The application also contains the names and addresses of the directors and officers of Duke Energy Corporation and Duke and indicates that all are United States citizens.

The NRC Staff performed an independent analysis, including open-source research and verification of the information provided in the application related to the ownership of Duke, and found no evidence FOCD.

Based on this review, the staff does not know or have reason to believe that Duke is owned, controlled, or dominated by a foreign interest. Therefore, Duke conforms to the guidance provided in the SRP for FOCD and meets the requirements of 10 CFR 50.38.

NUCLEAR INSURANCE and INDEMNITY:

This section of the SER addresses the applicant's offsite and onsite insurance requirements found in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 140, "Financial protection requirements and indemnity agreements," and 10 CFR 50.54(w), respectively.

The provisions of the Price-Anderson Act (Section 170 of the Atomic Energy Act of 1954, as amended) and the Commission's regulations in 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," require, in part, that each holder of a license issued pursuant to 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," have and maintain financial protection. Further, 10 CFR 50.54(w) establishes requirements for each power reactor licensee to obtain insurance or provide an equivalent amount of protection for the onsite costs of an accident. Under these regulations, Duke is required to provide satisfactory documentation that it has obtained the amount of financial protection required by (1) 10 CFR 140.13, “Amount of financial protection required of certain holders of construction permits and combined licenses under 10 CFR part 52,” (2) 10 CFR 140.11(a)(4), and (3) 10 CFR 50.54(w). In addition, each licensee required to have and maintain financial protection under 10 CFR 140.11(a)(4) shall provide evidence that it maintains a guarantee of payment of deferred premiums pursuant to 10 CFR 140.21, “Licensee guarantees of payment of deferred premiums.” Finally, as required by 10 CFR 140.20, “Indemnity agreements and liens,” the NRC staff will amend Duke’s current indemnity agreement to include the addition of Lee Units 1 and 2.

The regulation in 10 CFR 140.13 provides the amount of financial protection required by a Part 52 license holder, who also holds a license under 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material,” during the period before the Commission makes the finding under 10 CFR 52.103(g) (i.e., a finding that the acceptance criteria in the license are met, which allows the licensee to initially load fuel and operate). Because the 10 CFR Part 70 license will be issued with the COL, Duke must have and maintain $1,000,000 in financial protection from issuance of the COL until the 10 CFR 52.103(g) finding is made. In addition, as required by 10 CFR 140.11(a)(4), after the 10 CFR 52.103(g) finding is made, each licensee must have and maintain financial protection in an amount equal to the sum of primary financial protection ($375,000,000) and the amount available as secondary financial protection.

By letter dated February 19, 2016, (ADAMS Accession No. ML16056A014), Duke’s insurance broker, Marsh USA, Inc., provided proof of insurance coverage from American Nuclear Insurers in the amount of $1,000,000. Duke’s $1,000,000 insurance policy will remain in effect until the 10 CFR 52.103(g) finding. Therefore, the staff concludes that the proof of financial protection provided by DEC will satisfy the requirements in 10 CFR 140.13.

The staff notes that although licensees of large operating reactors under 10 CFR Parts 50 and 10 CFR Part 52 must have and maintain financial protection under 10 CFR 140.11 (a) (4) upon NRC action authorizing operation, the timing provisions for reporting under 10 CFR Part 140.21 do not explicitly address the 10 CFR Part 52 process. Under the requirements in 10 CFR 140.11(a)(4), 10 CFR 140.21, the coverage for secondary financial protection and the guarantee of payment of deferred premiums, are only required for reactors authorized to load fuel and operate. Under 10 CFR Part 52 COL process, the license authorizes operation only upon a Commission finding pursuant to 10 CFR 52.103(g). Therefore, these requirements apply as of the date the Commission makes such a finding. While 10 CFR 50.54(w) by its terms applies upon a Commission finding under 10 CFR 52.103(g), Duke also included a reporting requirement for 10 CFR 50.54(w) in its proposed condition.

Duke proposed the following license condition to address the reporting of 10 CFR Section 140.11(a)(4) requirements for secondary financial protection, and the reporting of 50.54(w) requirements for onsite financial protection. The staff agreed with the proposed license condition but made some modifications. The staff’s recommended license condition is stated below:

- License Condition (1-1) – Before the scheduled date for initial fuel load, and within ninety (90) days after the NRC publishes the notice of intended operation in the Federal Register, Duke shall provide satisfactory documentary evidence to the Director of the Office of Nuclear Reactor Regulation, or designee, that it has obtained the appropriate amount of primary and secondary financial protection required of licensees pursuant to 10 CFR 140.11(a)(4) and the appropriate amount of financial protection per 10 CFR 50.54(w).

With the license condition as described above, the staff concludes that Duke will satisfy the requirements of 10 CFR 140.11(a)(4) with respect to obtaining an appropriate amount of secondary financial protection and 10 CFR 50.54(w) with respect to obtaining the appropriate amount of financial protection. The staff notes that it will conform any license condition to the correct format if the Commission determines to issue the license. For example, the staff may change “the Director of the Office of Nuclear Reactor Regulation” to “the Director of the Office of New Reactors” and the like.

Duke also proposed the following license condition to address the reporting of 10 CFR 140.21 for guarantee of payment of deferred premiums. The staff agreed with the proposed license condition but made some modifications. The staff’s recommended license condition is stated below:

- License Condition (1-2) – Before the scheduled date of initial fuel load, and within ninety (90) days after the NRC publishes the notice of intended operation in the *Federal Register,* Duke shall provide evidence to the NRC that it would have the ability to pay into the nuclear industry retrospective rating plan in the event of a nuclear incident and in the amount specified in 10 CFR Part 140.11(a)(4) [sic] for one calendar year using one of the following methods:

  (a) Surety bond,

  (b) Letter of credit,

  (c) Revolving credit/term loan arrangement,

  (d) Maintenance of escrow deposits of government securities, or

  (e) Annual certified financial statement showing either that a cash flow (i.e., cash available to a company after all operating expenses, taxes, interest charges, and dividends have been paid) can be generated and would be available for payment of retrospective premiums within three (3) months after submission of the statement, or a cash reserve or a combination of cash flow and cash reserve.

  Thereafter, Duke shall provide evidence of the guarantees of payment of deferred premiums in accordance with the provisions specified in 10 CFR 140.21.

With the license condition as described above, the staff concludes that Duke will satisfy the requirement in 10 CFR 140.21.

In consideration of the staff’s evaluation and license conditions as described above, the staff concludes that DEC will satisfy the provisions of the Price-Anderson Act (Section 170 of the Atomic Energy Act of 1954, as amended) and the Commission’s applicable regulations in 10 CFR Part 140, 10 CFR Part 52, and 10 CFR Part 50 for insurance and indemnity.

CONCLUSION:

Based on the foregoing evaluation above, in consideration of the proposed license conditions, the NRC staff finds reasonable assurance that Duke is financially qualified to engage in the proposed activities regarding William States Lee III, Units 1 and 2 and that Duke satisfies the NRC requirements relating to financial qualification, decommissioning funding assurance, FOCD, and nuclear insurance and indemnity. The staff finds this acceptable since it conforms to the guidance in NUREG-1577, the SRP on FOCD, NUREG-1307, and meets the applicable regulations in 10 CFR Part 52, 10 CFR Part 50, and 10 CFR Part 140 as described above.

1.5.2 Nuclear Waste Policy Act

Nuclear Waste Policy Act of 1982, as amended, Section 302(b), “The Commission, as it deems necessary or appropriate, may require as a precondition to the issuance or renewal of a license under Section 103 or 104 of the Atomic Energy Act of 1954 [42 U.S.C. 2133, 2134] that the applicant for such license shall have entered into an agreement with the Secretary for the disposal of high-level radioactive waste and spent nuclear fuel that may result from the use of such license.”

On November 4, 2008, DEC entered into a contract with the United States Department of Energy (DOE) establishing the terms and conditions associated with the DOE’s responsibility for disposal of spent nuclear fuel and high-level radioactive waste generated at the proposed WLS Units 1 and 2 (ADAMS Accession No. ML083510882). The COE contract numbers applicable to WLS Units 1 and 2 are DE-CR01-09RW09003 and DE-CR01-09RW09004 respectively.

Since DEC has entered into contracts with the DOE for the disposal of high-level radioactive waste and spent nuclear fuel for WLS Units 1 and 2, the staff considers the applicable requirements of Nuclear Waste Policy Act of 1982, Section 302(b) to be met.

1.5.3 Consultation with Department of Homeland Security and Notifications

1.5.3.1 *Consultation with Department of Homeland Security*

In accordance with *Energy Policy Act of 2005,* Section 657, the staff consulted with the Department of Homeland Security (DHS). By letter, dated July 16, 2008, DHS provided their New Reactor Consultation Report on the William S. Lee Nuclear Station proposed to be built in Gaffney, South Carolina. The New Reactor Consultation Report satisfies the requirements of Section 657 of the Energy Policy Act of 2005.

1.5.3.2 *Notifications*

As required by Section 182c of the Atomic Energy Policy Act of 1954, as amended, and 10CFR 50.43(a), on December 15, 2011, the NRC notified the Public Service Commission of South Carolina and the North Carolina Utilities Commission of the WLS COL Application (ADAMS Accession Nos. ML112450014 and ML112450028).

In accordance with Section 182c of the Atomic Energy Policy Act of 1954, as amended, the staff also published a notice of application in the *Federal Register* on November 18, November 25, December 2, and December 9, 2011, 76 (FR 71608, 72725, 75566, and 77021).

Based on the staff's complete of notifications to regulatory agencies and the public notices described above, the staff concludes that, for the purposes of issuing COLs for WLS Units 1 and 2, any required notifications to other agencies or bodies have been duly made.

1.5.4 Evaluation of Departures and Exemption Associated with Application Organization and Numbering (10 CFR Part 52, Appendix D) and Exemption Associated with Special Nuclear Material Material Control and Accounting Program Description (10 CFR Part 70, Subpart D and 10 CFR Part 74 Subparts C, D, and E)

Evaluation of Departures and Exemption Associated with Organization and Numbering in the Application

In STD DEP 1.1-1 the applicant renumbered WLS COL FSAR Sections 2.1.1, 2.1.4, 2.2.1, 2.2.4, 2.4.1, 2.4.15, 2.5, 2.5.6, 9.2.11, 9.2.12, 9.2.13, 9.5.1.8, 9.5.1.9, 13.1, 13.1.4, 13.5, 13.5.3, 13.7, 17.5, 17.6, 17.7, and 17.8 to include content consistent with RG 1.206 and NUREG-0800. The departure and the exemption associated with the numbering scheme of the WLS COL FSAR are closely related.

Pursuant to 10 CFR 52.7, "Specific Exemptions," and 10 CFR 52.93, "Exemptions and Variances," the applicant requested an exemption from 10 CFR Part 52, Appendix D, Section IV.A.2.a, to include "a plant-specific DCD containing the same type of information and using the same organization and numbering as the generic DCD for the AP1000 design…." In WLS COL application Part 7, "Departures and Exemptions," the applicant stated that the exemption will not result in any significant departures from the expected organization and numbering of a typical FSAR, and the information is readily identifiable to facilitate an NRC review. The applicant states that the subject deviations are considered to be purely administrative to support a logical construction of the document. Further, the revised organization and numbering generally follows the guidance provided in RG 1.206 and NUREG-0800.

Pursuant to 10 CFR 52.7, the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR Part 52. 10 CFR 52.7 further states that the Commission's consideration will be governed by 10 CFR 50.12, "Specific exemptions," which states that an exemption may be granted when: (1) the exemptions are authorized by law, will not present an undue risk to public health or safety, and are consistent with the common defense and security; and (2) special circumstances are present. Special circumstances are present whenever, according to 10 CFR 50.12(a)(2)(ii), "Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule."

Before considering whether this numbering exemption should be granted, the staff needed to address a threshold question regarding the review standard applicable to the request. Under 10 CFR 52.93(a)(1), if a request for an exemption is from any part of a design certification rule, then the Commission may grant the exemption if the exemption complies with the appropriate change provision in the referenced design certification rule, or if there is no applicable change provision, if the exemption complies with 10 CFR 52.63. Here, there is no applicable change provision in the referenced design certification rule, so according to 10 CFR 52.93(a)(1), the exemption must meet 10 CFR 52.63. However, the standards of the appropriate provision of 10 CFR 52.63 applicable to requests for exemptions from a design certification rule in 10 CFR 52.63(b)(1), by their terms, also do not apply to this change. Specifically, 10 CFR 52.63(b)(1) applies to changes to "certification information," and not administrative or procedural design certification rule provisions such as this one under consideration. In the Statements of Consideration for 10 CFR 52.63, the Commission stated that it used the "phrase 'certification information' in order to distinguish the rule language in the DCRs from the design certification information (e.g., Tier 1 and Tier 2) that is incorporated by reference in the DCRs." (See 72 FR 49444). The exemption requested from the AP1000 DCD numbering scheme is an exemption from rule language, not Tier 1 or Tier 2 information; therefore, 10 CFR 52.63 should not be used to analyze this exemption.

Since there is not an applicable change provision in the referenced design certification, and because 10 CFR 52.63(b)(1) does not apply to this exemption, the exemption cannot comply with the plain language of 10 CFR 52.93(a)(1). In this situation, the language of 10 CFR 52.93(a)(1) does not appear to serve the underlying purpose of the regulation as described by the Commission in the Statements of Consideration to the rule, in which the Commission stated that only changes to certification information must meet 10 CFR 52.63. Instead, this exemption should have fallen under 10 CFR 52.93(a)(2), and, thus, be analyzed under the requirements in 10 CFR 52.7. Therefore, the staff finds that, pursuant to 10 CFR 52.7, an exemption to 10 CFR 52.93(a)(1) should be granted. This exemption is warranted because it meets the requirements in 10 CFR 50.12. First, because this is an administrative change regarding what exemption regulation applies, the exemption to 10 CFR 52.93(a)(1) is authorized by law, will not present an undue risk to public health or safety, and is consistent with the common defense and security. Additionally, application of the regulation in this case is not necessary to achieve the underlying purpose of the rule. The underlying purpose of the rule is to maintain the safety benefits of standardization by requiring any exemption from certification information to meet the requirements in 10 CFR 52.63(b)(1). This underlying purpose does not apply to this exemption, because the form and organization of the application does not affect the safety benefits of standardization of the certification information. Therefore, for the purpose of determining the standards applicable to the exemption related to STD DEP 1.1-1, the staff finds an exemption to 10 CFR 52.93(a)(1) acceptable for the review of the exemption related to STD DEP 1.1-1.

Pursuant to the exemption described above, the staff reviewed the exemption related to STD DEP 1.1-1 to determine whether the requirements in 10 CFR 52.7 are met. This exemption would allow the applicant to provide an FSAR with numbering and topics more closely related to NUREG-0800 and RG 1.206. The staff finds that this administrative change of minor renumbering will not present an undue risk to the public health and safety and is consistent with the common defense and security. In addition, this exemption is consistent with the Atomic Energy Act of 1954, as amended, and is authorized by law. Further, the application of the regulation in these particular circumstances is not necessary to achieve the underlying purpose of the rule. Therefore, the staff finds that the exemption to 10 CFR Part 52, Appendix D, Section IV.A.2.a is justified. Finally, for the same reasons the staff is granting the exemption request, the staff also finds the departure from the numbering scheme in the WLS COL FSAR acceptable.

Exemption Associated with Special Nuclear Material Material Control and Accounting Program

The applicant requested an exemption from the requirements of 10 CFR 70.22(b), 10 CFR 70.32(c) and, in turn, 10 CFR 74.31, 10 CFR 74.41, and 10 CFR 74.51. The provision of 10 CFR 70.22(b) requires an application for a license for SNM to include a full description of the applicant's program for MC&A of SNM under 10 CFR 74.31; 10 CFR 74.33, "Nuclear

material control and accounting for uranium enrichment facilities authorized to produce special nuclear material of low strategic significance"; 10 CFR 74.41; and 10 CFR 74.51[13].
10 CFR 70.32(c) requires a license authorizing the use of SNM to include and be subjected to a condition requiring the licensee to maintain and follow an SNM MC&A program. However, 10 CFR 70.22(b), 10 CFR 70.32(c), 10 CFR 74.31, 10 CFR 74.41, and 10 CFR 74.51 include exceptions for nuclear reactors licensed under 10 CFR Part 50. The regulations applicable to the MC&A of SNM for nuclear reactors licensed under 10 CFR Part 50 are provided in 10 CFR Part 74, Subpart B, 10 CFR 74.11 through 10 CFR 74.19, excluding 10 CFR 74.17. The purpose of this exemption request is to seek a similar exception for this COL under 10 CFR Part 52, such that the same regulations will be applied to the SNM MC&A program as nuclear reactors licensed under 10 CFR Part 50. In addition, the exemption request is evaluated under 10 CFR 52.7, which incorporates the requirements of 10 CFR 50.12. As stated previously, that section allows the Commission to grant an exemption if: 1) the exemption is authorized by law; will not present an undue risk to the public health and safety; and is consistent with the common defense and security; and 2) special circumstances are present as specified in 10 CFR 50.12(a)(2). The criteria in 10 CFR 50.12 encompass the criteria for an exemption in 10 CFR 70.17(a) and 10 CFR 74.7, the specific exemption requirements for 10 CFR Part 70 and 10 CFR Part 74, respectively. Therefore, by demonstrating that the exemption criteria in 10 CFR 50.12 are satisfied, the staff concludes that this request would also demonstrate that the exemption criteria in 10 CFR 52.7, 10 CFR 70.17(a), and 10 CFR 74.7 are satisfied.

The subject exemption would allow nuclear reactors licensed under 10 CFR Part 52 to be explicitly exempted from the requirements of 10 CFR 70.22(b), 10 CFR 70.32(c), 10 CFR 74.31, 10 CFR 74.41, and 10 CFR 74.51. There is no technical or regulatory basis to treat nuclear reactors licensed under 10 CFR Part 52 differently than reactors licensed under 10 CFR Part 50 with respect to the MC&A provisions in 10 CFR Part 74. As indicated in the Statement of Considerations for 10 CFR 52.0(b) (72 FR 49352, 49372, 49436 (August 28, 2007)), applicants and licensees under 10 CFR Part 52 are subject to all of the applicable requirements in 10 CFR Chapter I, whether or not those provisions explicitly mention a COL under 10 CFR Part 52. This regulation clearly indicates that plants licensed under 10 CFR Part 52 are to be treated no differently than plants licensed under 10 CFR Part 50 with respect to the substantive provisions in 10 CFR Chapter I (which includes 10 CFR Part 70 and 10 CFR Part 74). In particular, the exception for nuclear reactors licensed under 10 CFR Part 50, as in 10 CFR 70.22(b), 10 CFR 74.31, 10 CFR 74.41, or 10 CFR 74.51, should also be applied to reactors licensed under 10 CFR Part 52.

The staff agrees with the applicant's justification that nuclear reactors licensed under 10 CFR Part 52 should be treated the same as the reactors licensed under 10 CFR Part 50 regarding the MC&A for SNM.

Pursuant to 10 CFR 70.17(a), the Commission may, upon application of any interested person or upon its own initiative, grant such exemptions from the requirements of the regulations in this part as it determines are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest.

[13] While not including an explicit exception for 10 CFR Part 50 reactors, 10 CFR 74.33 applies only to uranium enrichment facilities and thus is not directly implicated in this exemption request.

In addition, pursuant to 10 CFR 74.7, "Specific exemptions," the Commission may, upon application of any interested person or upon its own initiative, grant such exemptions from the requirements of the regulations in this part as it determines are authorized by law and will not endanger life or property or the common defense and security, and are otherwise in the public interest.

Pursuant to 10 CFR 52.7, the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR Part 52. 10 CFR 52.7 further states that the Commission's consideration will be governed by 10 CFR 50.12, "Specific exemptions," which states that an exemption may be granted when: (1) the exemptions are authorized by law, will not present an undue risk to public health or safety, and are consistent with the common defense and security; and (2) when special circumstances are present. Special circumstances are present whenever, according to 10 CFR 50.12(a)(2)(ii), "Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule."

The staff reviewed the subject exemption, which will allow the applicant to have a similar exception for the COL under 10 CFR Part 52, such that the same regulations will be applied to the SNM MC&A program as nuclear reactors licensed under 10 CFR Part 50, and determined that this requested exemption will not present an undue risk to the public health and safety and is otherwise in the public interest. In addition, this exemption is consistent with the Atomic Energy Act of 1954, as amended, and is authorized by law. Therefore, granting this exemption will not adversely affect the common defense and security. Further, the application of the regulation in these particular circumstances is not necessary to achieve the underlying purpose of the rule. Since the exemption criteria in 10 CFR 50.12 are satisfied, the staff considers that this request also demonstrates that the exemption criteria in 10 CFR 52.7, 10 CFR 70.17(a), and 10 CFR 74.7 are satisfied. Therefore, the staff finds that the exemption from 10 CFR 70.22(b), 10 CFR 70.32(c) and, in turn, 10 CFR 74.31, 10 CFR 74.41, and 10 CFR 74.51, is justified.

1.5.5 Receipt, Possession, Use, and Transport of Source, Byproduct and Special Nuclear Material Authorized by 10 CFR Part 52 Combined Licenses

In DEC's Revision 4 of the COL application, dated August 9, 2011, including Part 1, "General and Financial Information," DEC requested material licenses for receipt, possession and use of source, byproduct and SNM in accordance with Commission regulations in 10 CFR Parts 30, 40, and 70. The reviews conducted for compliance with the requirements of 10 CFR Part 52 to support the issuance of the COLs encompass those necessary to support granting 10 CFR Parts 30, 40, and 70 licenses. In this respect, the 10 CFR Part 52 COLs for WLS will be consistent with the approach to 10 CFR Parts 30, 40, and 70 licensing followed for operating licenses for nuclear power plants licensed in accordance with 10 CFR Part 50. The staff considered the following proposed standard license provisions for the WLS COL as would relate to authorization pursuant to the regulations in 10 CFR Parts 30, 40, and 70[14].

[14] These proposed standard license conditions that the staff considered are based on similar license conditions found in other combined licenses.

Subject to the conditions and requirements incorporated herein, the Commission hereby licenses WLS:

(1) (a) pursuant to the Act and 10 CFR Part 70, to receive and possess at any time, special nuclear material as reactor fuel, in accordance with the limitations for storage and in amounts necessary for reactor operation, described in the FSAR, as supplemented and amended;

(b) pursuant to the Act and 10 CFR Part 70, to use special nuclear material as reactor fuel, after a Commission finding under 10 CFR 52.103(g) has been made, in accordance with the limitations for storage and amounts necessary for reactor operation, described in the FSAR, as supplemented and amended;

(2) (a) pursuant to the Act and 10 CFR Parts 30 and 70, to receive, possess, and use, at any time before a Commission finding under 10 CFR 52.103(g), such byproduct and special nuclear material (but not uranium hexafluoride) as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts not exceeding those specified in 10 CFR 30.35(d) and 10 CFR 70.25(d) for establishing decommissioning financial assurance, and not exceeding those specified in 10 CFR 30.72 and 10 CFR 70.22(i)(1);

(b) pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use, after a Commission finding under 10 CFR 52.103(g), any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as necessary;

(3) (a) pursuant to the Act and 10 CFR Parts 30 and 70, to receive, possess, and use, before a Commission finding under 10 CFR 52.103(g), any byproduct or special nuclear material (but not uranium hexafluoride) that is (1) in unsealed form; (2) on foils or plated surfaces, or (3) sealed in glass, for sample analysis or instrument calibration or other activity associated with radioactive apparatus or components, in amounts not exceeding those specified in 10 CFR 30.35(d) and 10 CFR 70.25(d) for establishing decommissioning financial assurance, and not exceeding those specified in 10 CFR 30.72 and 10 CFR 70.22(i)(1);

(b) pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use, after a Commission finding under 10 CFR 52.103(g), in amounts as necessary, any byproduct, source, or special nuclear material (but not uranium hexafluoride) without restriction as to chemical or physical form, for sample analysis or instrument calibration or other activity associated with radioactive apparatus or components; and

(4) pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

The staff notes that WLS COL FSAR Table 13.4-201 “Operational Programs Required by NRC Regulations,” provides milestones for the implementation of various operational programs. Important milestone dates for various operational programs that support issuance of the license and requirements relative to 10 CFR Parts 30, 40, and 70 include the following:

- Radiation Protection Program (including as low as is reasonably achievable [ALARA] principles) – prior to initial receipt of byproduct, source, or special nuclear materials (excluding exempt quantities as described in 10 CFR 30.18, “Exempt quantities”)

- Fire Protection Program – prior to initial receipt of byproduct, source, or special nuclear materials (excluding exempt quantities as described in 10 CFR 30.18, “Exempt quantities”)

- Security Program including physical security, safeguards contingency programs, training and qualification program – prior to receipt of fuel onsite (protected area)

- Security Program including physical security, safeguards contingency, and transportation programs – prior to transport or receipt of special nuclear material of low strategic significance

- Non-licensed plant staff training program associated with receipt of the radioactive material – prior to initial receipt of byproduct, source, or special nuclear materials (excluding exempt quantities as described in 10 CFR 30.18, “Exempt quantities”)

In WLS COL FSAR, Table 13.4-201, the applicant includes the appropriate milestones and requirements related to the SNM MC&A program. In addition, as documented in the following table WLS endorsed VEGP standard content letters related to this subject.

VEGP Letter Date	**VEGP Letters ADAMS Accession Nos.**	**WLS Endorsement Letter Date**	**WLS Letters ADAMS Accession Nos.**
July 29, 2009	ML092120064	December 18, 2009	ML093570280
July 9, 2010	ML101940025	January 6, 2011	ML110110399
October 15, 2010	ML102920120	April 25, 2011	ML11116A162
November 23, 2010	ML103300034	April 25, 2011	ML11116A162
March 16, 2011	ML110800088	April 25, 2011	ML11116A162

VEGP Letter Date	VEGP Letters ADAMS Accession Nos.	WLS Endorsement Letter Date	WLS Letters ADAMS Accession Nos.
March 3, 2011	ML110660153	April 25, 2011	ML11116A162
March 16, 2011[15]	ML110770137	April 25, 2011	ML11116A162
May 6, 2011	ML11129A155	May 18, 2011	ML11139A409
June 22, 2011	ML11175A169	July 28, 2011	ML11214A028

These letters identified the portions of the WLS COL application that demonstrate compliance with the requirements of 10 CFR Parts 30, 40, 70, and 74. The exemption request from the requirements of 10 CFR 70.22(b), 10 CFR 70.32(c) and, in turn, 10 CFR 74.31, 10 CFR 74.41, and 10 CFR 74.51 is addressed in Section 1.5.4 of this report.

Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff confirmed that the SNM Material and Control Accounting Program Description in Part 11 Enclosure D of the WLS COL application Revision 4 is identical to the November 23, 2010, VEGP submittal transmitting its SNM Material Control and Accounting Program Description. The only exceptions are that the title of the units are different and the identification that DEC and not Southern Nuclear Operating Company is responsible for implementation of the program is different.

- In an August 18, 2011, letter, the applicant concurred with the standard content of the VEGP SNMPPP description submitted in a March 16, 2011, letter. The staff confirmed that WLS SNMPPP description is identical with the only exception being the organization titles. However, there have been additional updates to the WLS SNMPPP in an October 16, 2014, letter from WLS. A specific review of the most recent WLS SNMPPP is discussed in Section 1.5.5 of this report.

[15] The March 16, 2011, letter from VEGP and the October 16, 2014, letter from WLS submitted the Special Nuclear Material Physical Protection Program (SNMPPP) Description for VEGP and WLS, respectively. Although the cover letters are publicly available, the SNMPP is considered safeguards information and is withheld from public disclosure.

- The staff confirmed that the VEGP new fuel shipping plan and the supplemental information in support of 10 CFR Part 70 special nuclear material found in Part 11 Enclosures E and F, respectively, of the VEGP COL application are identical to the material found in the WLS COL application Revision 8.

- The staff verified that the site-specific differences were not relevant and where the staff identified relevant differences, the staff performed additional review to determine the acceptability of the differences.

The staff completed its review and concluded that the evaluation performed for the standard content directly applicable to the WLS COL application, with the site-specific exceptions noted. This standard content material is identified in this report by use of italicized, double-indented formatting. Section 1.2.3 of this report provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the BLN Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 1.5.5:

> *In addition to the evaluation of the implementation milestones noted above, the staff's evaluation of the radiation protection program that supports the issuance of the 10 CFR Parts 30, 40, and 70 licenses is addressed in Chapter 12 of this SER. Additional staff evaluations that support the issuance of the 10 CFR Part 70 license are addressed in Chapter 9 of this SER (i.e., new fuel storage, spent fuel storage, and fire protection programs) and in the staff's evaluation of TVA's security program. The staff finds that the information in the Bellefonte COL application to support granting of the 10 CFR Part 70 license mentioned as part of the license above is sufficient, pending resolution of the open items in this report related to new and spent fuel, fire protection program, security program, and the implementation of the fire protection and security programs. However, TVA needs to provide a discussion of which parts of its COL application other than the reference to the radiation protection program provide sufficient information to support compliance with the applicable portions of 10 CFR Part 30 and 40, prior to the 10 CFR 52.103(g) finding. This is Open Item 1.5-1.*
>
> *<u>Resolution of Standard Content Open Item 1.5-1</u>*
>
> *In letters dated July 29, 2009, July 9, 2010, and October 15, 2010, the applicant provided additional information related to source, byproduct and SNM and its purposes, radiation safety personnel, personnel training, facilities and equipment, waste management, and the radiation safety program in general.*
>
> *Subsequent to the issuance of the SER with open items for the BLN application, the staff performed an additional review associated with granting the 10 CFR Parts 30, 40 and 70 licenses. For the 10 CFR Part 70 license, the staff considered SNM associated with the fuel (including security requirements) and SNM associated with non-fuel material (i.e., fission chambers). The staff also considered emergency plan requirements associated with SNM (fuel and non-fuel material). Based on these reviews, standard content Open Item 1.5-1 is resolved. These reviews are described below.*

Review of Parts 30 and 40 Materials

In a letter dated March 3, 2011, the applicant provided information regarding specific types of sources and byproduct material, the chemical or physical form, and the maximum amount at any time for the requested material licenses under 10 CFR Parts 30 and 40. The applicant also stated that SNM shall be in the form of reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the VEGP COL FSAR. Byproduct material and source material shall be in the form of sealed neutron sources for reactor startup and sealed sources for reactor instrumentation, radiation monitoring equipment, calibration, and fission detectors in amounts as required. The applicant also committed that no 10 CFR Part 40 specifically licensed source material, including natural uranium, depleted uranium and uranium hexafluoride will be received, possessed, or used during the period between issuance of the COL and the Commission's 10 CFR 52.103(g) finding for each of the VEGP Units 3 and 4. The applicant also stated that the quantity of any byproduct material with atomic numbers 1 through 93 would not exceed 100 millicuries for a single source and 5 Curies total. The maximum quantity for Americium 241 would not exceed 300 millicuries for single source and 500 millicuries total. Following the 10 CFR 52.103(g) finding for each of the VEGP Units 3 and 4, byproduct material, source material, and SNM in amounts as required, without restriction to chemical forms or physical form, would be used for the following:

- *Sample analysis,*
- *Instrument and equipment calibration, and*
- *Associated with radioactive apparatus or components.*

With respect to the requirements of 10 CFR Parts 30, 40, and 70 that are related to radiation protection (including administrative controls), the applicant provided information (in letters dated July 9, and November 23, 2010) on the purpose, storage and security of sources in VEGP COL FSAR Sections 12.2 and 12.5. Information related to the radiation protection program itself, including procedures for the use of these sources, is also described in VEGP COL FSAR Chapter 12. In addition, VEGP COL FSAR Section 13.4 states that the radiation protection program will be implemented according to the milestones listed in VEGP COL FSAR Table 13.4-201, Item 10. These milestones ensure that those portions of the program necessary to comply with the requirements of 10 CFR Parts 20, 30, 40, and 70, are implemented prior to the receipt of byproduct, source, SNM, or fuel, onsite.

The staff finds that the information provided by the applicant that describes the radiation protection measures (Chapter 12 of the VEGP COL FSAR) that will be implemented prior to receipt of byproduct, source or SNM, conforms to the applicable guidance in NUREG-1556, "Consolidated Guidance about Materials Licenses," and is, therefore, acceptable. The radiation protection program milestones included in the VEGP COL FSAR Table 13.4-201 are evaluated in Section 12.5 of this SER.

In a letter dated July 9, 2010, the applicant provided supplemental information relative to Item 14, Emergency Planning, in VEGP COL FSAR Table 13.4-201. In addition, the applicant proposed to revise the term 'portions applicable to SNM' to 'portions applicable to radioactive materials' for Item 14; Item 8, Fire Protection Program; Item 11, Non-Licensed Plant Staff Training Program; and Item 15, Physical Security Program. In addition, the applicant proposed to correct the references to regulatory citations of 10 CFR 30.32, "Application for specific licenses"; 10 CFR 40.31, "Application for specific licenses"; and 10 CFR 70.22, "Contents of applications." It also proposed to revise the "Requirements" column for Item 14 of the VEGP COL FSAR Table 13.4-201 to reference 10 CFR 30.32(i)(1), 10 CFR 40.31(j)(1), and 10 CFR 70.22(i)(1). It also proposed to revise Part 10 of the VEGP COL application, Proposed License Condition 3, "Operational Program Implementation," Section C, "Receipt of Materials," to include implementation of the portions of the emergency planning program applicable to SNM. In addition to the evaluation of the implementation milestones noted above, the staff's evaluation that supports the issuance of the 10 CFR Parts 30, and 40 licenses is addressed in Chapter 9 (the fire protection program).

The operational programs are specific programs that are required by regulations. VEGP COL FSAR Table 13.4-201 lists each operational program, the regulatory source for the program, the section of the FSAR in which the operational program is described, and the associated implementation milestone(s). The applicant proposed a license condition in Part 10, License Condition 3, Item C.3 of the VEGP COL application, which provides the milestones for implementing the portions of the non-licensed plant staff training program applicable to receipt of the radioactive material. However, Table 13.4-201 specifies implementation requirements (10 CFR 30.32(a), 10 CFR 40.31(a), and 10 CFR 70.22(a)) for the non-licensed plant staff training program associated with receipt of the radioactive material. Therefore, the staff determined that Item C.3 of proposed License Condition 3 is not needed because the implementation milestones for the non-licensed plant staff training program associated with receipt of radioactive material are governed by the applicable regulations.

The applicant proposed a license condition in Part 10 of the VEGP COL application to provide a schedule to support the NRC's inspection of operational programs, including the non-licensed plant staff training program applicable to receipt of the radioactive material. The proposed license condition is consistent with the policy established in SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," for operational programs and is acceptable.

In response to RAI 1.5-1, the applicant stated, in a letter dated October 15, 2010, that no byproduct material will be received, possessed, or used at AP1000 units of a physical form that is in unsealed form, on foils or plated sources, or sealed in glass, that exceeds the quantities in Schedule C of 10 CFR 30.72. Since the quantities do not exceed Schedule C, an emergency plan that meets the requirements of 10 CFR 30.32(i)(3) is not required. As such, the implementation of the emergency plan prior to the receipt of byproduct material will be removed from VEGP COL FSAR Table 13.4-201 and from Part 10 proposed License

Condition 3, Item C.4. The request for a 10 CFR Part 40 license does not involve authorization to receive, possess, or use uranium hexafluoride in excess of 50 kilograms in a single container or 1000 kilograms total. However, in a letter dated March 3, 2011, the applicant revised the request for a 10 CFR Part 40 license to state that no 10 CFR Part 40 specifically-licensed source material, including natural uranium, depleted uranium and uranium hexafluoride (UF_6), will be received, possessed, and used during the period between issuance of the COL and the Commission's 10 CFR 52.103(g) finding for each of the VEGP Units 3 and 4. Since the above quantities are not exceeded, an emergency plan for responding to the radiological hazards of an accidental release of source material and to any associated chemical hazards related to the material is not required. As such, the implementation of the emergency plan prior to the receipt of source material will be removed from VEGP COL FSAR Table 13.4-201. This applicant's proposal meets the requirements of 10 CFR 30.32 and 10 CFR 40.31 and is, therefore, acceptable. The incorporation of changes into a future revision of the VEGP COL FSAR is ***Confirmatory Item 1.5-1.***

Resolution of Standard Content Confirmatory Item 1.5-1

Confirmatory Item 1.5-1 is an applicant commitment to revise FSAR Table 13.4-201. The staff verified that the VEGP COL FSAR Table 13.4-201 was appropriately revised. As a result, Confirmatory Item 1.5- 1 is now closed.

The applicant also proposed an FSAR commitment to address the limitations during the period prior to the implementation of the emergency plan. In a letter dated March 16, 2011, the applicant stated that it has no plans to process UF_6 at the plant site at any time following the Commission's 10 CFR 52.103(g) finding, and consequently does not expect the requested 10 CFR Part 40 license to include receipt, storage, or use of UF_6 at the plant site. However, using the guidance of DC/COL-ISG-15, "Post-Combined License Commitments", the staff has determined that the commitment is not sufficient and instead the staff is proposing to add a restriction in the license condition related to 10 CFR Parts 30 and 40 (See License Condition 1-1.c(ii).

Review of Part 70 Materials

The staff reviewed information related to nuclear fuel as SNM included in the VEGP COL application including the AP1000 DCD against 10 CFR Part 70 requirements. Specifically, the staff's review included:

- *General information—financial qualification, site description, hydrology, geology, meteorology, the nearby population, and potential effects of natural phenomena (Part 1 of the application, FSAR Section 1.1 and Chapter 2, Section 4.1 and Table 4.1-1 of the AP1000 DCD against the requirements of 10 CFR 70.22(a)(1) through (a)(4));*

- *Organization and Administration—the responsibilities and associated resources for the receipt, possession, inspection, and storage of the SNM in the form of fresh fuel assemblies (Part 1 of the application, Quality Assurance Program included in Part 11 (Enclosure 11A) of the* *application, VEGP COL FSAR Section 13.1 for organization against the requirements of 10 CFR 70.22(a)(6) and (a)(8));*

- *Radiation Protection—Radiation protection program implementation, organization and personnel qualification, written procedures, ALARA, radiation survey and monitoring (AP1000 DCD Section 9.1 and Chapter 12 of VEGP COL FSAR against the requirements of 10 CFR 70.22(a)(6) through (a)(8));*

- *Nuclear Criticality Safety—use of area radiation monitors in lieu of criticality accident alarms (AP1000 DCD Sections 9.1.1.3 and 11.5.6 against the requirements of 10 CFR 70.22(a)(6) through (a)(8) and 10 CFR 50.68(b));*

- *Fire safety—fire protection program (VEGP COL FSAR Section 9.5.1 and Table 13.4-201 against the requirements of 10 CFR 70.22(a)(6) through (a)(8));*

- *Emergency Preparedness—emergency preparedness program for the VEGP site (VEGP COL FSAR Section 13.3 and Table 13.4-201 and the Emergency Plan against the requirements of 10 CFR 70.22(i));*

- *Environmental Protection—organization, procedures and controls that ensures that the environment is protected during the conduct of activities (i.e., receipt, possession, inspection, and storage of SNM) (VEGP COL FSAR Section 11.5 and AP1000 DCD Sections 9.1.1 and 11.5 against the requirements of 10 CFR 70.22(a)(7) and (a)(8)); and*

- *MC&A Program and Security (MC&A program included in the application against requirements of 10 CFR 70.22(b) and 10 CFR Part 74, and the Physical Security Plan (PSP) against the requirements of 10 CFR 73.67, "Licensee fixed site and in-transit requirements for the physical protection of special nuclear material of moderate and low strategic significance").*

As indicated above, the applicant's compliance with several applicable 10 CFR Part 70 requirements regarding radiation protection, nuclear criticality safety, and environmental protection is already encompassed by the design information incorporated by reference from the AP1000 DCD and evaluated by the staff as part of the design certification proceeding. As explained further below, with respect to other applicable 10 CFR Part 70 requirements to be addressed by the COL applicant, the staff finds that the information provided regarding general information, organization and administration, radiation protection, nuclear criticality safety, fire safety, emergency preparedness, and environmental protection to support receipt, storage, and possession of SNM, conforms to the applicable guidance in NUREG-1520 and NUREG-0800 and, therefore, is acceptable. First, however, the staff's review of information regarding the MC&A program (10 CFR 70.22(b) and 10 CFR Part 74) and the PSP (10 CFR 73.67) is provided below.

MC&A Program for SNM (Fuel)

In RAI 1.5-3, the staff requested the applicant to review the requirements of 10 CFR 70.22(b) for the program addressing the control and accounting of SNM and provide descriptions of how the applicable requirements for material accounting and controls under 10 CFR Part 74 will be met for the possession and storage of SNM during construction and prior to the operation of the nuclear power plant. In addition, the staff requested the applicant to provide a proposed license condition to clearly establish full implementation of the MC&A program meeting the applicable requirements of 10 CFR Part 74 prior to receipt of SNM, consistent and concurrent with the proposed license condition for implementing the applicable security (i.e., physical protection) requirements of 10 CFR Part 73.

In response to RAI 1.5-3, the applicant, in a letter dated November 23, 2010, stated that all non-irradiated SNM for the AP1000 units is identified as Category III, SNM of low strategic significance, as defined in 10 CFR 74.4, "Definitions." No SNM at an AP1000 nuclear facility will exceed an uranium-235 isotope enrichment of 10 percent. The quantity of SNM will be documented, controlled, and communicated to the NRC as required in 10 CFR 74.13, "Material status reports"; 10 CFR 74.15, "Nuclear material transaction reports"; and 10 CFR 74.19, "Recordkeeping."

Subsequent to the applicant's endorsement of the standard content response to RAI 01.05-3 stating that no SNM onsite will exceed a 10-percent uranium-235 isotope enrichment level in an August 18, 2011, letter, the applicant updated its COL application to include Part 11F, "Supplemental Information of 10 CFR Part 70 Special Nuclear Material License Application" acknowledging that WLS would possess uranium sources containing uranium enriched to 93 percent uranium-235 in a quantity meeting the criteria of SNM of low strategic significance.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 1.5.5:

In its response to RAI 1.5-3, the applicant also described the SNM MC&A program and stated that this program will be provided as an enclosure in the VEGP COL application, Part 11. The SNM MC&A program will be developed for control and accounting of SNM in accordance with the applicable requirements of 10 CFR Part 74, Subparts A and B. This program will be consistent with guidance of American National Standards Institute (ANSI) 15.8-2009, "Material Control Systems – Special Nuclear Material Control and Accounting Systems for Nuclear Power Plants." The SNM MC&A program will be implemented prior to receipt of SNM at the plant site and will remain in effect until the SNM is shipped from the plant site. The procedures constituting the SNM MC&A program will delineate the requirements, responsibilities, and methods of SNM control necessary to address the following programmatic elements:

1. *Establish, maintain, and follow written MC&A procedures to account for SNM.*

2. *Maintain adequate records of the initial receipt or current inventory of SNM, including records of isotopic content, material received, material*

shipped, and material lost (material balance reports and physical inventory listing reports).

3. *Develop adequate inventory procedures and maintain adequate perpetual inventory records.*

4. *Inventory SNM within the 12-month prescribed frequency.*

5. *Report SNM inventories on the applicable forms.*

6. *Establish an individual responsible for the control and accountability of SNM.*

7. *Report the loss of or inability to find SNM items in a timely manner.*

8. *Control access to SNM.*

9. *Control the shipping and transfer of SNM.*

The applicant proposed to add a new FSAR Section 13.5.2.2.9, which will summarize the use of plant procedures to address MC&A of SNM. The applicant also stated that VEGP COL FSAR Table 13.4-201 will be revised to provide information related to implementation of the SNM MC&A program.

In order to address the applicable 10 CFR Part 74 MC&A requirements prior to power operation, the applicant proposed a license condition that will require implementation of a MC&A program prior to receipt of SNM on site. Implementation of the SNM MC&A program prior to SNM receipt will also address the SNM possession and storage requirements during construction and prior to operation of the nuclear power plant.

The applicant's MC&A program for SNM is consistent with ANSI 15.8 and meets reporting and recordkeeping requirements of 10 CFR 74.11, "Reports of loss or theft or attempted theft or unauthorized production of special nuclear material"; 10 CFR 74.13; 10 CFR 74.15; and 10 CFR 74.19. The documentation, submitted by the applicant, for a program addressing the control and accounting of SNM provided descriptions of how the applicable requirements for material accounting and controls under 10 CFR Part 74 are met and, therefore, is acceptable, subject to the proposed revision to the VEGP COL application and the VEGP COL FSAR (this has been tracked as ***Confirmatory Item 1.5-2****). In addition, the proposed license condition includes a provision to provide a schedule to support the NRC's inspection of the MC&A program for the SNM. This is consistent with the policy established in SECY-05-0197 and is thus acceptable.*

Resolution of Standard Content Confirmatory Item 1.5-2

Confirmatory Item 1.5-2 is an applicant commitment to revise its FSAR Sections 13.4, 13.5 and Parts 7 and 11 (Enclosure 11D) of its application to address the SNM MC&A program. The staff verified that the VEGP COL FSAR and Parts 7 and 11 (Enclosure D) of its application were appropriately revised. As a result, Confirmatory Item 1.5-2 is now closed.

<u>*Security Review for 10 CFR Part 70 Materials*</u>

In accordance with 10 CFR 73.55(a)(4), current applicants for an operating license under 10 CFR Part 50, or a COL under 10 CFR Part 52 who have submitted their applications to the Commission prior to the effective date of this rule must amend their applications to include security plans consistent with this section.

The Commission worded 10 CFR 73.55(a)(4) to require implementation of 10 CFR 73.55, "Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage," "before fuel is allowed onsite (protected area)." The Commission explained this provision as follows:

> *This paragraph establishes when an applicant's physical protection program must be implemented. The receipt of special nuclear material (SNM) in the form of fuel assemblies onsite, (i.e., within the licensee's protected area) is the event that subjects a licensee or applicant to the requirements of this rule, and it is the responsibility of the applicant or licensee to complete the preliminary and preparatory actions required to implement an effective physical protection program at the time SNM is received onsite (within the protected area). 74 FR 13926, 13960 (Mar. 27, 2009).*

Further guidance is provided in the form of RGs to support implementation of this Rule. The following guidance is provided in RG 5.76, "Physical Protection Programs at Nuclear Power Reactors":

> *Except for mixed-oxide (MOX) fuel assemblies, the Commission requirements of 10 CFR 73.67, "Licensee Fixed Site and In-Transit Requirements for the Physical Protection of Special Nuclear Material of Moderate and Low Strategic Significance," apply and must be met until fuel assemblies are received inside an operational protected area. Consistent with 10 CFR 73.55(a)(4), applicants for an operating license under the provisions of 10 CFR Part 50, or holders of a COL under the provisions of 10 CFR Part 52, shall implement the requirements of 10 CFR 73.55 before special nuclear material (SNM) in the form of fuel assemblies are allowed on site (in the protected area).*

In a letter dated March 15, 2011, the NRC staff asked the applicant to provide its plan regarding the protection of new fuel as SNM at the VEGP Units 3 and 4 plant site prior to declaration of an operational protected area (PA) and implementation of the requirements of 10 CFR 73.55, as described in the SNM MC&A Program description. In addition, the staff also requested that the applicant consider the applicability of the substantive provisions of interim compensatory orders (ICMO) that were issued to Category III Fuel Cycle Facilities to ensure adequate protection when SNM is on site prior to the activation of the PA. In response to the staff's questions, in a letter dated March 16, 2011, the applicant provided a physical protection plan in accordance with 10 CFR 73.67(f) and (g). This plan was included as an annex to the PSP.

This plan includes transportation security provisions. The applicant also stated that once the PA is declared operational in accordance with 10 CFR 73.55(a)(4), the annex would no longer be required and could be removed in accordance with 10 CFR 50.54(p). Then, no separate transportation security provisions would be necessary for future new fuel shipments. The staff raised a question regarding the licensee's ability to receive new fuel and return new fuel rods/assemblies to the fuel manufacturer. In a letter dated May 6, 2011, the applicant proposed to revise its FSAR Section 13.5.2.2.8 to include the New Fuel Shipping Plan that addresses the applicable 10 CFR 73.67 requirements in the event that unirradiated new fuel assemblies or components are returned to the supplying fuel manufacturer(s) facility. The New Fuel Shipping Plan summarizes the procedures and the written agreement that the applicant will have in place prior to shipment of new fuel back to the fuel manufacturer and this plan will be included in Part 11, Enclosures of its application. The staff finds this New Fuel Shipping Plan acceptable because it meets the applicable requirements of 10 CFR 73.67(g). The staff verified that the VEGP FSAR Section 13.5 and Part 11 (Enclosure 11E) are appropriately updated.

In the RAI response dated March 16, 2011, the applicant addressed the Order imposing fingerprinting and criminal history records check requirements for unescorted access to radioactive material or other property dated April 30, 2007. In accordance with Section 5.4 of the PSP annex, the applicant committed to utilizing the access authorization program as outlined in Section 14.1 of the PSP. The access authorization program in Section 14.1 is in accordance with 10 CFR 73.56, "Personnel Access Authorization Requirements for Nuclear Power Plants," based on implementing guidance as provided by RG 5.66, "Access Authorization Program for Nuclear Power Plants," Revision 1 and Section 652 of the Energy Policy Act of 2005 (EP Act).

The applicant conducted a critical target area analysis (CTA), and determined that a CTA would not exist. Because there is no CTA at the facility, there is no need to address security issues related to CTAs. In addition, the applicant has adequately addressed security issues related to; security response procedures, coordination with local law enforcement for response support, storage of hazardous materials on-site, review of emergency shutdown/cool down procedures, supplementing of the Emergency Actions Levels, site accountability and evacuation strategies, emergency communications, evaluation of computer and communications networks for vulnerabilities, capabilities to provide fire suppression, evaluation of the need for offsite medical support, emergency support, and access to Federal support, and limiting public access to sensitive plant information. However, the staff has determined that the commitment included in the RAI responses is not sufficient and instead the staff is proposing to add a license condition to ensure adequate protection prior to implementation of the requirements of 10 CFR 73.55. This license condition (1-5) will preclude changes to the security plan provisions related to these issues without prior NRC approval until such matters fall under the new reactor security requirements of 10 CFR 73.55.

The staff's review of the applicant's PSP for the protection of SNM of low strategic significance (LSS) [Note: WLS refers to this plan as the SNMPPP] *includes information that has been marked as "Safeguards Information" by the* *applicant, pursuant to ~~10 CFR 2.390~~*[10 CFR 73.21 and 73.22]*. The NRC staff reviewed the applicant's PSP for fixed site physical protection of SNM- LSS and chemicals of concern. The methods and procedures outlined in the PSP satisfy the performance objectives, systems capabilities, and reporting requirements specified in 10 CFR 73.67. The PSP for the facility is acceptable and provides reasonable assurance that the requirements for the physical protection of SNM-LSS and chemicals of concern will be met. The staff also verified that the PSP is appropriately updated.*

Non-Fuel SNM

In a letter dated, June 22, 2011, the applicant provided information regarding the name, amount, and specifications (including the chemical and physical form and, where applicable, isotopic content) of the non-fuel SNM (Fission Chambers) the applicant proposes to use (10 CFR 70.22(a)(4)). The letter also provided information to confirm that the applicable design and programmatic elements provided in the licensing basis will satisfy the requirements in 10 CFR 70.22(a)(6) through (8) prior to receipt of non-fuel SNM.

10 CFR Part 70 Requirements - Other than MC&A (10 CFR 70.22(b) and 10 CFR Part 74) and Security (10 CFR 73.67) - for Fuel and Non-Fuel Material

As noted above, in addition to MC&A and security, the staff also examined the applicant's compliance with 10 CFR Part 70 requirements regarding general information, organization and administration, radiation protection, nuclear criticality safety, fire safety, emergency preparedness, and environmental protection to support receipt, storage, and possession of SNM.

The staff's analysis follows with respect to those other requirements not already resolved via the applicant's incorporation of the AP1000 DCD. For the reasons described in Section 1.4.4 of this FSER the staff agrees that the applicant is technically qualified to engage in the proposed activities associated with this license, based on the applicant's ongoing experience in the safe operation of nuclear power plants, as presented in Section 1.4.1 of the VEGP COL FSAR. Likewise, the applicant's financial qualifications and ownership structure meet the requirements of 10 CFR 70.22 for the same reasons described above in Section 1.5.1.

Note: WLS COL FSAR Section 1.4.1 has a similar discussion regarding DEC's operation of Catawba Units 1 and 2, McGuire Units 1 and 2, and Oconee Units 1, 2, and 3. The staff also concludes DEC is technically qualified to engage in the proposed activities associated with this license based on DEC's on-going experience with the safe operation of Catawba Units 1 and 2, McGuire Units 1 and 2, and Oconee Units 1, 2, and 3. In addition, Section 1.5.1 of this report finds that the financial qualifications and ownership structure for the WLS COL application acceptable.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 1.5.5:

Similarly, the applicant has explained the anticipated amounts, types, and uses of 10 CFR Part 70 materials at the site are consistent with the provisions of 10 CFR 70.22. The VEGP COL FSAR and Part 1 of the application provide *adequate description of the VEGP Units 3 and 4 facility and the proposed activities related to 10 CFR Parts 30, 40 and 70 material. In addition the VEGP COL FSAR provides information regarding regional hydrology, geology, meteorology, the nearby population, and potential effects of natural phenomena that could occur at the facility. The applicant has described the responsibilities and associated resources (see Part 1, "General and Administration Information," and Enclosure 11A, "Nuclear Development Quality Assurance Manual" of the application) for the receipt, possession, inspection, and storage of the 10 CFR Part 70 material (fuel and non-fuel). Therefore, it meets the requirements of 10 CFR 70.22(a)(1). Furthermore, as indicated in VEGP COL FSAR Table 13.4-201, applicable portions of the Radiation Protection Program will be implemented prior to initial receipt of byproduct, source, or SNMs. In accordance with VEGP COL FSAR Table 13.4-201, Item 10, Implementation Milestone #1, and the NRC-approved template, Nuclear Energy Institute (NEI) 07-03A, "Generic FSAR Template Guidance for Radiation Protection Program Description," which is incorporated by reference into VEGP COL FSAR Appendix 12AA (see SER Section 12.5), the appropriate radiation protection program elements associated with organization, facilities, instrumentation and equipment, procedures (e.g., procurement, receipt, inventory, labeling, leak testing, surveillance, control, transfer, disposal, storage, issuance, and use of radioactive sources), and training will be in place prior to initial receipt of byproduct, source, or special nuclear materials, thereby satisfying the requirements of 10 CFR 70.22(a)(4), (6), (7), and (8). VEGP COL FSAR Section 12.2 includes the requirements for written procedures that address leak-testing of radioactive sources. The leak-test will be consistent with 10 CFR 20.1501, "General," survey and monitoring requirements for evaluating the quantities of radioactive material and the potential radiological hazard of the radioactive source.*

The fission chambers will be disposed of consistent with the operating procedures that specify the processes to be followed to ship waste that complies with the waste acceptance criteria (WAC) of the disposal site, the waste classification and characteristics requirements of 10 CFR 61.55, "Waste classification," and 10 CFR 61.56, "Waste characteristics," and the requirements of third party waste processors as applicable. This process is identified in VEGP COL FSAR Section 11.4.6.1.

With respect to fire safety, prior to installation, the new fission chambers (along with the new fuel) will be stored in the Auxiliary Building fuel handling area, which is an area protected by the fire protection program and fire protection system, as discussed in the AP1000 DCD Section 9A.3.1.3.1.2. Temporary storage of these non-combustible sealed sources is not specifically addressed in the AP1000 fire protection analysis in DCD Appendix 9A; however, the approach to extinguishing fires and containing material releases associated with the fission chambers would be similar to, and bounded by, the approach considered for the fuel handling area in general. The fuel handling area has been evaluated and determined acceptable for the storage of SNM in a full core load of new fuel. The hazards imposed by the relatively small quantity of SNM associated with the fission chambers (less than 100 grams), is not expected to be a challenge to the existing fire protection analysis for the new fuel storage (see Section 9.5.1 of this SER). The VEGP COL FSAR Section 12.2 includes the requirements for written *procedures that address leak testing of radioactive sources (byproduct, source, and devices that contain SNM, as appropriate). Further, the fission chambers that contain the non-fuel SNM are sealed sources that are tested periodically to confirm their leak-tightness. Therefore, it is expected that the capabilities of the fire protection program and the fire protection equipment servicing this area are sufficient to meet the requirements of 10 CFR 70.22(a)(7) and 10 CFR 70.22(a)(8).*

Emergency Plan (SNM, Fuel and Non-Fuel)

The applicant will be storing the new fuel in the new fuel rack (stored dry) or in the spent fuel racks prior to loading into the reactor. The safety analysis included in AP1000 DCD Sections 9.1.1.3 and 9.1.2.3 provides safety analysis that indicates that: (1) the design of new fuel rack is such that K_{eff} remains less than or equal to 0.95 with full density unborated water and less than equal to 0.98 with optimum moderation and full reflection conditions; and (2) the design of spent fuel rack is such that K_{eff} remains less than or equal to 0.95 under design basis conditions. This criticality evaluation meets the requirements of 10 CFR 50.68(b). Therefore, a criticality accident alarm system to meet the requirements of 10 CFR 70.24, "Criticality accident requirements," is not required. As a result, an emergency plan (to receive and possess) pursuant to 10 CFR 70.22(i) is also not required. In addition, an emergency plan for the fission chambers (to receive and possess) pursuant to 10 CFR 70.22(i) is not required due to the small quantity of SNM (less than 100 grams) associated with the fission chambers.

1.5.5.1 *Physical Protection of Special Nuclear Material*

1.5.5.1.1 Introduction

This section addresses the physical protection of special nuclear material while possessed, used, and transported by the applicant, including during the period prior to implementation of the nuclear power reactor physical security plan (PSP). This review was performed by the Office of Nuclear Security and Incident Response (NSIR), Division of Security Policy (DSP), Fuel Cycle and Transportation Security Branch (FCTSB).

1.5.5.1.2 Summary of Application

The post September 11, 2001, security order for SNM of low strategic significance was sent to the applicant to be addressed in the form of a request for additional information. The letter conveying the order was sent on June 30, 2014 (ADAMS Accession No. ML14154A249) and the safeguards-information-containing order was sent under separate cover (Safeguards LAN Electronic Safe (SLES) Accession No. NS113121). In a letter dated October 16, 2014, the applicant provided a crosswalk that pointed out the text of the application that described the intent of meeting each element of the applicable portions of 10 CFR 73.67 (ADAMS Accession No. ML14290A523). In addition, the applicant submitted another letter dated October 16, 2014 that contained safeguards information and included a revised Special Nuclear Material Physical Protection Program Plan (SNMPPP) and a reviewer's aid matrix. The reviewer's aid matrix covered both the applicable 10 CFR 73.67 requirements and the applicable post September 11, 2001, security order.

1.5.5.1.3 Regulatory Basis

The regulatory requirements and guidance applicable to fixed site and in-transit physical protection are as follows:

- 10 CFR 73.67, "Licensee fixed site and in-transit requirements for the physical protection of special nuclear material of moderate and low strategic significance"
- Post September 11, 2001, Security Order for SNM of Low Strategic Significance
- 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 (1983)"
- RG 5.66, "Access Authorization Program for Nuclear Power Plants"
- RIS 2005-22, "Requirements for the Physical Protection During Transportation of Special Nuclear Material of Moderate and Low Strategic Significance: 10 CFR Part 73 vs. RG 5.59 (1983)"

1.5.5.1.4 Technical Evaluation

A technical evaluation of the DEC, Lee Units I and 2, COL FSAR, against applicable 10 CFR 73.67 fixed site and in-transit general performance objectives, general requirements and physical protection requirements for SNM of low strategic significance, was performed.

In addition the post September 11, 2001, security order for SNM of low strategic significance was sent to the applicant to be addressed in the form of a request for additional information. The letter conveying the order was sent on June 30, 2014 (ADAMS Accession No. ML14154A249) and the safeguards-information-containing order was sent under separate cover (SLES Accession No. NS113121). A technical evaluation of how the order was addressed was also performed. In a letter dated October 16, 2014, the applicant provided a crosswalk that pointed out the text of the application that described the intent of meeting each element of the applicable portions of 10 CFR 73.67 (ADAMS Accession No. ML14290A523). In addition, in a letter dated October 16, 2014, the applicant provided a revised SNMPPP and a reviewer's aid matrix (SLES Accession No. NS112457). The reviewer's aid matrix covered both the applicable 10 CFR 73.67 requirements and the applicable post September 11, 2001, security order.

1.5.5.1.4.1 *Fixed Site General Performance Objectives*

The applicable physical protection requirements specified in 10 CFR 73.67 titled, "Licensee fixed site and in-transit requirements for the physical protection of special nuclear material of moderate and low strategic significance," provide general performance objectives.

The physical protection requirements of 10 CFR 73.67(a)(1), states, "General performance objectives.

(1) Each licensee who possess, uses or transports special nuclear material of moderate or low strategic significance shall establish and maintain a physical protection system that will achieve the following objectives:

(i) Minimize the possibilities for unauthorized removal of special nuclear material consistent with the potential consequences of such actions; and

(ii) Facilitate the location and recovery of missing special nuclear material.

(2) To achieve these objectives, the physical protection system shall provide:

(i) Early detection and assessment of unauthorized access or activities by an external adversary within the controlled access area containing special nuclear material;

(ii) Early detection of removal of special nuclear material by an external adversary from a controlled access area;

(iii) Assure proper placement and transfer of custody of special nuclear material; and

(iv) Respond to indications of an unauthorized removal of special nuclear material and then notify the appropriate response forces of its removal in order to facilitate its recovery.

Therefore, the fixed site physical protection requirements of 10 CFR 73.67(a)(1) are applicable because of the manner in which SNM of low strategic significance was described in the WLS Units 1 and 2 COL application.

Applicable Requirement: 10 CFR 73.67(a)(1), "General performance objectives. (1) Each licensee who possesses, uses or transports special nuclear material of moderate or low strategic significance shall establish and maintain a physical protection system that will achieve the following objectives:…"

The applicant stated in "Table 13.4-201," (ADAMS Accession No. ML11229A602) under "Item 15," titled "Implementation Milestone," their commitment to meet the requirements of 10 CFR 73.67, "…prior to initial receipt of SNM." Establishment of the physical protection system is outlined in the SNMPPP, Revision 1, dated October 2014 (SLES Accession No. NS112457). Specifically, in Subsection 4.4.1, "Establishment of the Physical Protection System," there are six establishment elements described that pertain to: lighting, detection, alarm station status, communications, access control and physical barriers of the controlled access area. In addition, within Subsection 4.4.2, "Maintenance of the Physical Protection System," of the SNMPPP there is an explanation of the maintenance that will be applied to the physical protection system. The application as written stated that DEC will implement the applicable requirements of the physical protection program required by 10 CFR 73.67 before SNM is received on site.

DEC's application described that 10 CFR 73.67 will be fully implemented before SNM is on site. Also, the application outlined establishment and maintenance elements for the physical protection system. Therefore, the staff finds the requirement of 10 CFR 73.67(a)(1) to have a physical protection system established and maintained, would be met.

Applicable Requirement: 10 CFR 73.67(a)(1)(i), "General performance objectives. Each licensee who possesses, uses or transports special nuclear material of moderate or low strategic significance shall establish and maintain a physical protection system that will achieve the following objectives: (i) Minimize the possibilities for unauthorized removal

of special nuclear material consistent with the potential consequences of such actions. . .”

The applicant stated in “Table 13.4-201,” (ADAMS Accession No. ML11229A602) under “Item 15,” titled “Implementation Milestone,” their commitment to meet the requirements of 10 CFR 73.67, “…prior to initial receipt of SNM.” In addition, the SNMPPP describes in Subsection 5.3.1, “Monitoring SNM (Non-Fuel SNM-HEU Neutron Sources),” how this general performance objective will be met for the highly enriched uranium (HEU) sources by detailing adversary scenarios and how the physical protection system will work to meet the requirement. In addition, the SNMPPP in Subsection 5.3.2, “Monitoring SNM (New Fuel Assemblies),” describes adversary scenarios applied to SNM reactor fuel and how the physical protection system is designed to meet this requirement.

DEC’s application describes that 10 CFR 73.67 will be fully implemented before SNM is received. In addition, their SNMPPP describes how the possibilities for unauthorized removal are minimized consistent with the consequences of such actions. Therefore, the staff finds the requirement of 10 CFR 73.67(a)(1)(i) to have a physical protection system established and maintained that has the objective to minimize the possibilities for unauthorized removal of special nuclear material consistent with the potential consequences of such actions, would be met.

Applicable Requirement: 10 CFR 73.67(a)(1)(ii), “General performance objectives. Each licensee who possesses uses or transports special nuclear material of moderate or low strategic significance shall establish and maintain a physical protection system that will achieve the following objectives: “…(ii) Facilitate the location and recovery of missing special nuclear material.”

The applicant stated in “Table 13.4-201,” (ADAMS Accession No. ML11229A602) under “Item 15,” titled “Implementation Milestone,” their commitment to meet the requirements of 10 CFR 73.67, “…prior to initial receipt of SNM.” In addition, their SNMPPP in Section 5.10, “Contingency Response,” describes the detection assessment and response strategies of the physical protection system that would facilitate the location and recovery of missing special nuclear material.

DEC’s application describes that 10 CFR 73.67 will be fully implemented before SNM is received. In addition, their SNMPPP describes the detection, assessment and response attributes of the physical protection system that would facilitate the location and recovery of missing special nuclear material. Therefore, the staff finds the requirement, of 10 CFR 73.67(a)(1)(ii) to have a physical protection system established and maintained that has the objective to facilitate the location and recovery of missing special nuclear material, would be met.

Applicable Requirement: 10 CFR 73.67(a), “General performance objectives. (2) To achieve these objectives, the physical protection system shall provide: (i) Early detection and assessment of unauthorized access or activities by an external adversary within the controlled access area containing special nuclear material. . .”

The applicant stated in “Table 13.4-201,” (ADAMS Accession No. ML11229A602) under “Item 15,” titled “Implementation Milestone,” their commitment to meet the requirements of 10 CFR 73.67, “…prior to initial receipt of SNM.” In addition, their SNMPPP in Subsections 5.3.1 and 5.3.2 describes how the physical protection system provides for early detection and assessment of unauthorized access or activities by an external adversary within the controlled access area containing special nuclear material.

DEC's application describes that 10 CFR 73.67 will be fully implemented before SNM is received. In addition, their SNMPPP adequately describes the early detection and assessment physical protection strategies to address unauthorized access or activities by an external adversary within the controlled access area containing special nuclear material. Therefore, the staff finds the requirement, of 10 CFR 73.67(a)(2)(i) to have a physical protection system that provides early detection and assessment of unauthorized access or activities by an external adversary within the controlled access area containing special nuclear material, would be met.

Applicable Requirement: 10 CFR 73.67(a)(2)(ii), "General performance objectives. To achieve these objectives, the physical protection system shall provide: (ii) Early detection of removal of special nuclear material by an external adversary from a controlled access area. . ."

The applicant stated in "Table 13.4-201," (ADAMS Accession No. ML11229A602) under "Item 15," titled "Implementation Milestone," their commitment to meet the requirements of 10 CFR 73.67, "…prior to initial receipt of SNM." In addition, their SNMPPP in Subsections 5.3.1 and 5.3.2 describes how the physical protection system provides for early detection of removal of special nuclear material by an external adversary from a controlled access area.

DEC's application describes that 10 CFR73.67 will be fully implemented before SNM is received. In addition, their SNMPPP describes the early detection and assessment physical protection strategies to address removal of special nuclear material by an external adversary from a controlled access area. Therefore, the staff finds the requirement, of 10 CFR 73.67(a)(2)(ii) to have a physical protection system that provides early detection of removal of special nuclear material by an external adversary from a controlled access area, would be met.

Applicable Requirement: 10 CFR 73.67(a)(2)(iii), "General performance objectives. To achieve these objectives, the physical protection system shall: …(iii) Assure proper placement and transfer of custody of special nuclear material; and…"

The applicant stated in "Table 13.4-201," (ADAMS Accession No. ML11229A602) under "Item 22," titled "Implementation Milestone," their commitment to meet the requirements of 10 CFR 74, (i.e., Material Control and Accounting of Special Nuclear Material) "Prior to receipt of special nuclear material" as a "license condition." Also, the applicant stated in "FSAR Part 11D," "Special Nuclear Material (SNM) Material Control and Accounting Program Description," that the applicant will establish, a "…SNM control and accounting system,"…" including internal control, physical inventory and shipment of SNM."

In addition, the applicant stated in their SNMPPP in Section 5.1, "Receipt of SNM," within Subsection 5.1.1 (pertaining to non-fuel SNM), Subsection 5.1.2 (pertaining to fuel SNM), and in Section 5.8, "Internal Transfers," material control and accounting (MC&A) measures specific to the non-fuel and fuel SNM, respectively.

DEC's application describes that the appropriate provisions of 10 CFR 74 will be fully implemented before SNM is received. In addition, the applicant has described in the SNMPPP how specific MC&A measures apply to meet this general performance objective, therefore, the staff finds the requirement, of 10 CFR 73.67(a)2(iii), to assure proper placement and transfer of custody of special nuclear material, would be met.

Applicable Requirement: 10 CFR 73.67(a)(2)(iv), "General performance objectives. To achieve these objectives, the physical protection system shall: … (iv) Respond to indications of an unauthorized removal of special nuclear material and then notify the appropriate response forces of its removal in order to facilitate its recovery."

The applicant stated in "Table 13.4-201," (ADAMS Accession No. ML11229A602) under "Item 15," titled "Implementation Milestone," their commitment to meet the requirements of 10 CFR 73.67, "…prior to initial receipt of SNM." In addition, their SNMPP in Section 5.10 describes the detection, assessment and response measures that would provide indications of missing or stolen SNM and subsequent recovery thereof. The appropriate response from offsite (i.e., the specifically coordinated with local law enforcement agency (LLEA), etc.) was pointed out in the SNMPPP by referencing Section 8 of the power reactor physical security plan (PSP) (Revision 3, dated April 10, 2013, (SLES Accession No. NS112930)), and Sections 5.6, 5.7 and 5.8 of the power reactor contingency plan (CP) (Revision 3, dated April 10, 2013, (SLES Accession No. NS112930)).

DEC's application describes that 10 CFR 73.67 will be fully implemented before SNM is received. In addition, their SNMPPP adequately describes the early detection, assessment and response physical protection strategies that would facilitate recovery of missing or stolen SNM; therefore, the staff finds the requirement, of 10 CFR 73.67(a)(2)(iv) to have a physical protection system that shall respond to indications of an unauthorized removal of special nuclear material and then notify the appropriate response forces of its removal in order to facilitate its recovery, would be met.

1.5.5.1.4.2 *Fixed Site General Requirements*

The applicable requirements specified in 10 CFR 73.67, "Licensee fixed site and in-transit requirements for the physical protection of special nuclear material of moderate and low strategic significance," include the following general requirements for fixed sites.

> "(c) Each licensee who possesses, uses, transports, or delivers to a carrier for transport special nuclear material of moderate strategic significance, or 10 kg or more of special nuclear material of low strategic significance shall:
>
> (1) Submit a security plan or an amended security plan describing how the licensee will comply with all the requirements of paragraphs (d), (e), (f), and (g) of this section, as appropriate, including schedules of implementation. The licensee shall retain a copy of the effective security plan as a record for 3 years after the close of period for which the licensee possesses the special nuclear material under each license for which the original plan was submitted. Copies of superseded material must be retained for 3 years after each change.
>
> (2) Within 30 days after the plan submitted pursuant to paragraph (c)(1) of this section is approved, or when specified by the NRC in writing, implement the approved security plan."

Applicable Requirement: 10 CFR 73.67(c)(1), "Submit a security plan…including schedules for implementation…shall retain a copy …for 3 years..." … "Copies of the superseded material must be retained for 3 years after each change."

The applicant stated in Section 5.7, “Audits and Records,” of their SNMPPP that the security plan (i.e., the SNMPPP) would be retained for 3 years and that copies of superseded material will be retained for 3 years after each change.

The application stated that DEC will not receive SNM of low strategic significance (both fuel and non-fuel) on site until implementation of the physical protection system required by 10 CFR 73.67 is accomplished.

In addition, DEC’s SNMPPP adequately describes the required retention parameters for the SNMPPP and changes to it.

Therefore, the staff finds the requirement, of 10 CFR 73.67(c)(1) to submit a security plan, retain the security plan for 3 years after the specific type of SNM has been removed from the site, and to retain superseded security plan change(s) for 3 years after each change, would be met.

Applicable Requirement: 10 CFR 73.67(c)(2), “Within 30 days after the plan submitted pursuant to paragraph (c)(1) of this section is approved, or when specified by the NRC in writing, implement the approved security plan.”

The applicant stated in their revised SNMPPP, in Section 1, “Scope,” thereof, that: “The NRC will be notified 180 days prior to the establishment of the CAA for SNM receipt.” However, this leaves the NRC staff without provisions specific for implementation of the complete SNMPPP and the required provisions of SNM of low strategic significance for transport, therefore a license condition will be configured to meet the requirement. This requirement would be met through a license condition that states, (or conveys the same meaning with different wording):

1. DEC will notify the NRC 120 days prior to shipment of special nuclear material of low strategic significance to, or receipt of special nuclear material of low strategic significance at, the Lee Nuclear Plant Unit 1 or Unit 2 owner controlled area, and

2. DEC will implement the Special Nuclear Material Physical Protection Plan that describes the physical protection strategies for special nuclear material of low strategic significance, 120 days prior to the shipment of special nuclear material of low strategic significance to, or receipt of special nuclear material of low strategic significance at, the Lee Nuclear Plant Unit 1 or Unit 2 owner controlled area.”

1.5.5.1.4.3 *Fixed Site Physical Protection Requirements*

The applicable requirements specified in 10 CFR 73.67 titled, “Licensee fixed site and in-transit requirements for the physical protection of special nuclear material of moderate and low strategic significance,” provide fixed site physical protection requirements for SNM of low strategic significance.

The physical protection requirements of 10 CFR 73.67(f), state, “Fixed site requirements for special nuclear material of low strategic significance. Each licensee who possesses, stores, or uses special nuclear material of low strategic significance at a fixed site or contiguous sites, except those who are licensed to operate a nuclear power reactor pursuant to Part 50, shall:

(1) Store or use the material only within a controlled access area,

(2) Monitor with an intrusion alarm or other device or procedures the controlled access areas to detect unauthorized penetrations or activities,

(3) Assure that a watchman or offsite response force will respond to all unauthorized penetrations or activities, and

(4) Establish and maintain response procedures for dealing with threats of thefts or thefts of this material. The licensee shall retain a copy of the current response procedures as a record for 3 years after the close of period for which the licensee possesses the special nuclear material under each license for which the procedures were established. Copies of superseded material must be retained for 3 years after each change."

The fixed site physical protection requirements of 10 CFR 73.67(f) are applicable because of the manner in which SNM of low strategic significance was described in the WLS Units 1 and 2 COL application.

Applicable Requirement: 10 CFR 73.67(f)(1), "Fixed site requirements for special nuclear material of low strategic significance. Each licensee who possesses, stores, or uses special nuclear material of low strategic significance at a fixed site or contiguous sites, except those who are licensed to operate a nuclear power reactor pursuant to Part 50, shall: (1) Store or use the material only within a controlled access area…"

The applicant stated in "Table 13.4-201," (ADAMS Accession No. ML11229A602) under "Item 15," titled "Implementation Milestone," their commitment to meet the requirements of 10 CFR 73.67, "…prior to initial receipt of SNM." In addition, their SNMPPP in Sections: 5.2 "Storage," 5.8 and in Figures 1 through13, describe the physical characteristics of the controlled access area.

DEC's application describes that 10 CFR 73.67 will be fully implemented before SNM is received. In addition, their SNMPPP adequately describes the characteristics of their planned-for controlled access area; therefore, the staff finds the requirement, of 10 CFR 73.67(f)(1) to store or use the material only within a controlled access area, would be met.

Applicable Requirement: 10 CFR 73.67(f)(2), "Fixed site requirements for special nuclear material of low strategic significance. Each licensee who possesses, stores, or uses special nuclear material of low strategic significance at a fixed site or contiguous sites, except those who are licensed to operate a nuclear power reactor pursuant to Part 50, shall: (2) Monitor with an intrusion alarm or other device or procedures the controlled access areas to detect unauthorized penetrations or activities. . ."

The applicant stated in "Table 13.4-201," (ADAMS Accession No. ML11229A602) under "Item 15," titled "Implementation Milestone," their commitment to meet the requirements of 10 CFR 73.67, "…prior to initial receipt of SNM." In addition, their SNMPPP in Subsections 5.3.1 and 5.3.2 describes the detection processes that would result in recognition of unauthorized penetrations or activities in the locations of SNM of low strategic significance and the controlled access area.

DEC's application describes that 10 CFR 73.67 will be fully implemented before SNM is received. In addition, their SNMPPP describes the detection processes that would result in recognition of unauthorized penetrations or activities in the locations of SNM of low strategic significance and the controlled access area. Therefore, the staff finds the requirement, of 10 CFR 73.67(f)(2) to monitor with an intrusion alarm or other device or procedures the controlled access areas to detect unauthorized penetrations or activities, would be met.

Applicable Requirement: 10 CFR 73.67(f)(3), "Fixed site requirements for special nuclear material of low strategic significance. Each licensee who possesses, stores, or uses special nuclear material of low strategic significance at a fixed site or contiguous sites, except those who are licensed to operate a nuclear power reactor pursuant to part 50, shall: (3) Assure that a watchman or offsite response force will respond to all unauthorized penetrations or activities. . ."

The applicant stated in "Table 13.4-201," under "Item 15," titled "Implementation Milestone," their commitment to meet the requirements of 10 CFR 73.67, "…prior to initial receipt of SNM." In addition, their SNMPPP in Subsections: 5.3.1, 5.3.2, and in Section 5.10, describes the detection, assessment and response measures for the physical protection of the material. Furthermore, the appropriate response from offsite (i.e., the specifically coordinated with local law enforcement agency (LLEA), etc.) was pointed out by referencing Section 8 of the reactor physical security plan (PSP) (Revision 3, dated April 10, 2013, (SLES Accession No. NS112930)) and Sections 5.6, 5.7 and 5.8 of their reactor contingency plan (CP) (Revision 3, dated April 10, 2013, (SLES Accession No. NS112930)).

DEC's application describes that 10 CFR 73.67 will be fully implemented before SNM is received. In addition, their SNMPPP and other information referenced in the SNMPPP describes the detection, assessment and response measures for the physical protection of the material; therefore, the staff finds the requirement, of 10 CFR 73.67(f)(3) to assure that a watchman or offsite response force will respond to all unauthorized penetrations or activities, would be met.

Applicable Requirement: 10 CFR 73.67(f)(4), "Fixed site requirements for special nuclear material of low strategic significance. Each licensee who possesses, stores, or uses special nuclear material of low strategic significance at a fixed site or contiguous sites, except those who are licensed to operate a nuclear power reactor pursuant to Part 50, shall: (4) Establish and maintain response procedures for dealing with threats of thefts or thefts of this material. The licensee shall retain a copy of the current response procedures as a record for 3 years after the close of period for which the licensee possesses the special nuclear material under each license for which the procedures were established. Copies of superseded material must be retained for 3 years after each change."

The applicant stated in "Table 13.4-201," (ADAMS Accession No. ML11229A602) under "Item 15," titled "Implementation Milestone," their commitment to meet the requirements of 10 CFR 73.67, "…prior to initial receipt of SNM." In addition, their SNMPPP in Section 4.1 "Procedures," Subsection 5.3.1, Subsection 5.3.2, Section 5.7, and in Section 5.10, describes the framework of and details to the development of response procedures. In addition, in Section 5.7 of the SNMPPP the retention of 3 years for response procedures and changes thereof, are noted.

DEC's application describes that 10 CFR 73.67 will be fully implemented before SNM is received. In addition, their SNMPPP describes the framework of the response procedures, details on the development of response procedures and retention actions of 3 years of the response procedures; therefore, the staff finds the requirement, of 10 CFR 73.67(f)(4), to establish and maintain response procedures, would be met.

1.5.5.1.4.4 *In-Transit General Performance Objectives*

The applicable requirements specified in 10 CFR 73.67, "Licensee fixed site and in-transit requirements for the physical protection of special nuclear material of moderate and low strategic significance," include general performance objectives.

The physical protection requirements of 10 CFR 73.67(a), state the following, "General performance objectives":

(1) Each licensee who possesses, uses, or transports special nuclear material of moderate or low strategic significance shall establish and maintain a physical protection system that will achieve the following objectives:

   (i) Minimize the possibilities for unauthorized removal of special nuclear material consistent with the potential consequences of such actions; and

   (ii) Facilitate the location and recovery of missing special nuclear material.

(2) To achieve these objectives, the physical protection system shall provide:

   (i) Early detection and assessment of unauthorized access or activities by an external adversary within the controlled access area containing special nuclear material;

   (ii) Early detection of removal of special nuclear material by an external adversary from a controlled access area;

   (iii) Assure proper placement and transfer of custody of special nuclear material; and

   (iv) Respond to indications of an unauthorized removal of special nuclear material and then notify the appropriate response forces of its removal in order to facilitate its recovery.

The in-transit physical protection requirements of 10 CFR 73.67(a) are applicable because of the manner in which SNM of low strategic significance was described in the WLS Units 1 and 2 COL application.

Applicable Requirement: 10 CFR 73.67(a), "General performance objectives. (1) Each licensee who possesses, uses or transports special nuclear material of moderate or low strategic significance shall establish and maintain a physical protection system that will achieve the following objectives:…"

The applicant included a description of how it intended to meet the in-transit physical protection requirements of 10 CFR 73.67(g) in Section 6 of their SNMPPP. It is stated in the SNMPPP that a SNM-qualified licensed shipper; other than DEC, will be used for transport of SNM of low strategic significance both to and from the site. In addition, it is stated in Section 6, “Shipment,” of the SNMPPP that DEC will confirm that the licensee used for transport of SNM has “…plans and procedures…” that are developed and implemented in such a manner that each general performance objective of 10 CFR 73.67 will be met. Because DEC will be using a SNM-qualified licensee to perform the shipment of SNM of low strategic significance and will confirm that such a licensee has the physical protection measures in place to meet each general performance objective, subsequently that SNM-qualified licensee will have the ability to meet the requirement to establish and maintain a physical protection system.

The DEC application stated that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance, and that DEC will confirm that the licensed shipper has physical protection measures in place to meet each general performance objective; therefore, the staff finds the requirement, of 10 CFR 73.67(a)(1) to establish and maintain a physical protection system, would be met.

Applicable Requirement: 10 CFR 73.67(a)(1)(i), “General performance objectives. Each licensee who possesses, uses or transports special nuclear material of moderate or low strategic significance shall establish and maintain a physical protection system that will achieve the following objectives: (i) Minimize the possibilities for unauthorized removal of special nuclear material consistent with the potential consequences of such actions. . .”

The applicant included a description of how it intended to meet the in-transit physical protection requirements of 10 CFR 73.67(g) in Section 6 of their SNMPPP. It is stated in the SNMPPP that a SNM-qualified licensed shipper, other than DEC, will be used for transport of SNM of low strategic significance both to and from the site. In addition, it is stated in Section 6 of the SNMPPP that DEC will confirm that the licensee used for transport of SNM has “…plans and procedures…” that are developed and implemented in such a manner that each general performance objective of 10 CFR 73.67 will be met. Because DEC will be using a SNM-qualified licensee to perform the shipment of SNM of low strategic significance and will confirm that such a licensee has the physical protection measures in place to meet each general performance objective, subsequently that SNM-qualified licensee will have the ability to meet the requirement to establish and maintain a physical protection system that has the capability to minimize the possibilities for unauthorized removal of special nuclear material consistent with the potential consequences of such actions.

The DEC application stated that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance, and that DEC will confirm that the licensed shipper has physical protection measures in place to meet each general performance objective. Therefore, the staff finds the requirement, of 10 CFR 73.67(a)(1)(i) to minimize the possibilities for unauthorized removal of special nuclear material consistent with the potential consequences of such actions, would be met.

Applicable Requirement: 10 CFR 73.67(a)(1)(ii), “General performance objectives. Each licensee who possesses uses or transports special nuclear material of moderate or low strategic significance shall establish and maintain a physical protection system that will achieve the following objectives: “…(ii) Facilitate the location and recovery of missing special nuclear material.”

The applicant included a description of how it intended to meet the in-transit physical protection requirements of 10 CFR 73.67(g) in Section 6 of their SNMPPP. It is stated in the SNMPPP that a SNM-qualified licensed shipper, other than DEC, will be used for transport of SNM of low strategic significance both to and from the site. In addition, it is stated in Section 6 of the SNMPPP that DEC will confirm that the licensee used for transport of SNM has "…plans and procedures…" that are developed and implemented in such a manner that each general performance objective of 10 CFR 73.67 will be met. Because DEC will be using a SNM-qualified licensee to perform the shipment of SNM of low strategic significance and will confirm that such a licensee has the physical protection measures in place to meet each general performance objective, subsequently that SNM-qualified licensee will have the ability to meet the requirement of establishing and maintaining a physical protection system that has the capability to facilitate the location and recovery of missing special nuclear material.

The DEC application stated that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance, and that DEC will confirm that the licensed shipper has physical protection measures in place to meet each general performance objective. Therefore, the staff finds the requirement, of 10 CFR 73.67(a)(1)(ii) to "(i) Minimize …; and," (ii) Facilitate the location and recovery of missing special nuclear material," would be met.

Applicable Requirement: 10 CFR 73.67(a), "General performance objectives. (2) To achieve these objectives, the physical protection system shall provide: (i) Early detection and assessment of unauthorized access or activities by an external adversary within the controlled access area containing special nuclear material. . ."

The applicant included a description of how it intended to meet the in-transit physical protection requirements of 10 CFR 73.67(g) in Section 6 of their SNMPPP. It is stated in the SNMPPP that a SNM-qualified licensed shipper, other than DEC, will be used for transport of SNM of low strategic significance both to and from the site. In addition, it is stated in Section 6 of the SNMPPP that DEC will confirm that the licensee used for transport of SNM has "…plans and procedures…" that are developed and implemented in such a manner that each general performance objective of 10 CFR 73.67 will be met. Because DEC will be using a SNM-qualified licensee to perform the shipment of SNM of low strategic significance and will confirm that such a licensee has the physical protection measures in place to meet each general performance objective-subsequently that SNM-qualified licensee will have the ability to meet the requirement of establishing and maintaining a physical protection system that has the capability to provide for early detection and assessment of unauthorized access or activities by an external adversary within the controlled access area containing special nuclear material.

The DEC application stated that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance, and that DEC will confirm that the licensed shipper has physical protection measures in place to meet each general performance objective. Therefore, the staff finds the requirement, of 10 CFR 73.67(2)(i) to provide "Early detection and assessment of unauthorized access or activities by an external adversary within the controlled access area containing special nuclear material...," would be met.

Applicable Requirement: 10 CFR 73.67(a)(2)(ii), "General performance objectives. To achieve these objectives, the physical protection system shall provide: (ii) Early detection of removal of special nuclear material by an external adversary from a controlled access area. . ."

The applicant included a description of how it intended to meet the in-transit physical protection requirements of 10 CFR 73.67(g) in Section 6 of their SNMPPP. It is stated in the SNMPPP that a SNM-qualified licensed shipper, other than DEC, will be used for transport of SNM of low strategic significance both to and from the site. In addition, it is stated in Section 6 of the SNMPPP that DEC will confirm that the licensee used for transport of SNM has "…plans and procedures…" that are developed and implemented in such a manner that each general performance objective of 10 CFR 73.67 will be met. Because DEC will be using a SNM-qualified licensee to perform the shipment of SNM of low strategic significance and will confirm that such a licensee has the physical protection measures in place to meet each general performance objective-subsequently that SNM-qualified licensee will have the ability to meet the requirement of establishing and maintaining a physical protection system that has the capability to provide for early detection of removal of special nuclear material by an external adversary from a controlled access area.

The DEC application stated that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance, and that DEC will confirm that the licensed shipper has physical protection measures in place to meet each general performance objective. Therefore, the staff finds the requirement, of 10 CFR 73.67(2)(ii) to provide "Early detection of removal of special nuclear material by an external adversary from a controlled access area...," would be met.

Applicable Requirement: 10 CFR 73.67(a)(2)(iii), "General performance objectives. To achieve these objectives, the physical protection system shall: (iii) Assure proper placement and transfer of custody of special nuclear material; and…"

The applicant included a description of how it intended to meet the in- transit physical protection requirements of 10 CFR 73.67(g) in Section 6 of their SNMPPP. It is stated in the SNMPPP that a SNM-qualified licensed shipper, other than DEC, will be used for transport of SNM of low strategic significance both to and from the site. In addition, it is stated in Section 6 of the SNMPPP that DEC will confirm that the licensee used for transport of SNM has "…plans and procedures…" that are developed and implemented in such a manner that each general performance objective of 10 CFR 73.67 will be met. Also, DEC, in their SNMPPP, describes in Section 5.1, "Receipt of SNM," within Subsections 5.1.1 (for non-fuel SNM) and 5.1.2 (for fuel SNM), the process for receiving and placing SNM. Furthermore, SNM to be transported from the site or received at the site will have an MC&A program applied to it as described in Part 11D of the application. Because DEC will be using a SNM-qualified licensee to perform the shipment of SNM of low strategic significance, will confirm that such a licensee has the physical protection measures in place to meet each general performance objective, has procedures for receipt/placement of SNM and has an MC&A program that will apply to SNM; subsequently that SNM-qualified licensed shipper and DEC will have the ability to meet the requirement of establishing and maintaining a physical protection system that has the capability to assure proper placement and transfer of custody of special nuclear material.

The DEC application stated that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance, and that DEC will confirm that the licensed- shipper has physical protection measures in place to meet each general performance objective. In addition, DEC has a described process for receiving and placing SNM and will have a MC&A program applied to SNM to be shipped or received. Therefore, the staff finds the requirement, of 10 CFR 73.67(2)(iii) to assure proper placement and transfer of custody of special nuclear material, would be met.

Applicable Requirement: 10 CFR 73.67(a)(2)(iv), "General performance objectives. To achieve these objectives, the physical protection system shall: (iv) Respond to indications of an unauthorized removal of special nuclear material and then notify the appropriate response forces of its removal in order to facilitate its recovery."

The applicant included a description of how it intended to meet the in-transit physical protection requirements of 10 CFR 73.67(g) in Section 6 of their SNMPPP. It is stated in the SNMPPP that a SNM-qualified licensed shipper, other than DEC, will be used for transport of SNM of low strategic significance both to and from the site. In addition, it is stated in Section 6 of the SNMPPP that DEC will confirm that the licensee used for transport of SNM has "…plans and procedures…" that are developed and implemented in such a manner that each general performance objective of 10 CFR 73.67 will be met. Because DEC will be using a SNM-qualified licensee to perform the shipment of SNM of low strategic significance and will confirm that such a licensee has the physical protection measures in place to meet each general performance objective, subsequently that SNM-qualified licensee will have the ability to meet the requirement of responding to indications of an unauthorized removal of special nuclear material and then notify the appropriate response forces of its removal in order to facilitate its recovery.

The DEC application stated that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance, and that DEC will confirm that the licensed shipper has physical protection measures in place to meet each general performance objective. Therefore, the staff finds the requirement, of 10 CFR 73.67(a)(2)(iv) to respond to indications of an unauthorized removal of special nuclear material and then notify the appropriate response forces of its removal in order to facilitate its recovery, would be met.

1.5.5.1.4.5 *In-Transit General Requirements*

The applicable requirements specified in 10 CFR 73.67, "Licensee fixed site and in-transit requirements for the physical protection of special nuclear material of moderate and low strategic significance," include the following general requirements.

> (c) Each licensee who possesses, uses, transports, or delivers to a carrier for transport special nuclear material of moderate strategic significance, or 10 kg or more of special nuclear material of low strategic significance shall:
>
> (1) Submit a security plan or an amended security plan describing how the licensee will comply with all the requirements of paragraphs (d), (e), (f), and (g) of this section, as appropriate, including schedules of implementation. The licensee shall retain a copy of the effective security plan as a record for 3 years after the close of period for which the licensee possesses the special nuclear material under each license for which the original plan was submitted. Copies of superseded material must be retained for 3 years after each change.
>
> (2) Within 30 days after the plan submitted pursuant to paragraph (c)(1) of this section is approved, or when specified by the NRC in writing, implement the approved security plan."

Applicable Requirement: 10 CFR 73.67(c)(1), "Submit a security plan including schedules for implementation. . . shall retain a copy. . . for 3 years. . ." "Copies of the superseded material must be retained for 3 years after each change."

The applicant included a description of how it intended to meet the in-transit physical protection requirements of 10 CFR 73.67(g) in Section 6 of their SNMPPP. It is stated in the SNMPPP that a SNM-qualified licensed shipper, other than DEC, will be used for transport of SNM of low strategic significance both to and from the site. In addition, it is stated in Section 6 of the SNMPPP that DEC will confirm that the licensee used for transport of SNM has "…plans and procedures…" that are developed and implemented in such a manner that 10 CFR 73.67(c)(1) would be met.

DEC's application describes that 10CFR73.67 will be fully implemented before SNM is received. In addition, their SNMPPP describes the required retention parameters for the SNMPPP and changes to it; therefore, the requirement of 10 CFR 73.67(c)(1) to retain the security plan for 3 years after the specific type of SNM has been removed from the site, and superseded security plan change(s) shall be retained for 3 years after each change, would be met.

The DEC application stated that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance, and that DEC will confirm that the licensed shipper has provisions in place to meet 10 CFR 73.67(c)(1); therefore, the staff finds the requirement, of 10 CFR 73.67(c)(1), to "Submit a security plan or as amended security plan describing how the licensee will comply with all the requirements of paragraphs (d), (e), (f), and (g) of this section, as appropriate, including schedules of implementation. The licensee shall retain a copy of the effective security plan as a record for 3 years after the close of period for which the licensee possesses the special nuclear material under each license for which the original plan was submitted. Copies of superseded material must be retained for 3 years after each change…," would be met.

Applicable Requirement: 10 CFR 73.67(c)(2), "Within 30 days after the plan submitted pursuant to paragraph (c)(1) of this section is approved, or when specified by the NRC in writing, implement the approved security plan."

The applicant included a description of how it intended to meet the in-transit physical protection requirements of 10 CFR 73.67(g) in Section 6 of its SNMPPP. The SNMPPP states that a SNM-qualified licensed shipper, other than DEC, will be used for transport of SNM of low strategic significance both to and from the site. In addition, Section 6 of the SNMPPP states that DEC will confirm that the licensee used for transport of SNM has "…plans and procedures…" that are developed and implemented in such a manner that 10 CFR 73.67(c)(2), would be met.

1.5.5.1.4.6 *In-Transit Physical Protection Requirements*

The applicable requirements specified in 10 CFR 73.67, "Licensee fixed site and in-transit requirements for the physical protection of special nuclear material of moderate and low strategic significance," describes in-transit physical protection requirements.

The physical protection requirements of 10 CFR 73.67(g) state, "In-transit requirements for special nuclear material of low strategic significance.

(1) Each licensee who transports or who delivers to a carrier for transport special nuclear material of low strategic significance shall:

   (i) Provide advance notification to the receiver of any planned shipments specifying the mode of transport, estimated time of arrival, location of the nuclear material transfer point, name of carrier and transport identification,

(ii) Receive confirmation from the receiver prior to commencement of the planned shipment that the receiver will be ready to accept the shipment at the planned time and location and acknowledges the specified mode of transport,
(iii) Transport the material in a tamper indicating sealed container,
(iv) Check the integrity of the containers and seals prior to shipment, and
(v) Arrange for the in-transit physical protection of the material in accordance with the requirements of Section 73.67(g)(3) of this part, unless the receiver is a licensee and has agreed in writing to arrange for the in-transit physical protection.

(2) Each licensee who receives quantities and types of special nuclear material of low strategic significance shall:

(i) Check the integrity of the containers and seals upon receipt of the shipment,
(ii) Notify the shipper of receipt of the material as required in Section 74.15 of this chapter, and
(iii) Arrange for the in-transit physical protection of the material in accordance with the requirements of Section 73.67(g)(3) of this part, unless the shipper is a licensee and has agreed in writing to arrange for the in-transit physical protection.

(3) Each licensee, either shipper or receiver, who arranges for the physical protection of special nuclear material of low strategic significance while in transit or who takes delivery of such material free on board (f.o.b.) the point at which it is delivered to a carrier for transport shall:

(i) Establish and maintain response procedures for dealing with threats or thefts of this material. The licensee shall retain a copy of the current response procedures as a record for 3 years after the close of period for which the licensee possesses the special nuclear material under each license for which the procedures were established. Copies of superseded material must be retained for 3 years after each change.

(ii) Make arrangements to be notified immediately of the arrival of the shipment at its destination, or of any such shipment that is lost or unaccounted for after the estimated time of arrival at its destination, and

(iii) Conduct immediately a trace investigation of any shipment that is lost or unaccounted for after the estimated arrival time and notify the NRC Operations Center within one hour after the discovery of the loss of the shipment and within one hour after recovery of or accounting for such lost shipment in accordance with the provisions of Section 73.71 of this part."

The in-transit physical protection requirements of 10 CFR 73.67(g) are applicable because of the manner in which SNM of low strategic significance was described in the WLS Units I and 2 COL application.

Applicable Requirement: 10 CFR 73.67(g), "In-transit requirements for special nuclear material of low strategic significance. (1) Each licensee who transports or who delivers

to a carrier for transport special nuclear material of low strategic significance shall: (i) Provide advance notification to the receiver of any planned shipments specifying the mode of transport, estimated time of arrival, location of the nuclear material transfer point, name of carrier and transport identification. . .”

The applicant included a description of how it was intended to meet the in-transit physical protection requirements of 10 CFR 73.67(g) in Section 6 of their SNMPPP. It is stated in the SNMPPP that a SNM-qualified licensed shipper, other than DEC, will be used for transport of SNM of low strategic significance both to and from the site. In addition, it is stated in Section 6 of the SNMPPP that DEC will confirm that the licensee used for transport of SNM has “…plans and procedures…” that are developed and implemented in such a manner that 10 CFR 73.67(g)1)(i) will be met. DEC will be using a SNM-qualified licensee to perform the shipment of SNM of low strategic significance and will confirm that such a licensee has the physical protection measures in place to meet 10 CFR 73.67(g)1)(i). Therefore, that SNM-qualified licensee will have the ability to meet the requirement of providing advance notification to the receiver of any planned shipments specifying the mode of transport, estimated time of arrival, location of the nuclear material transfer point, name of carrier and transport identification.

The DEC application stated that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance, and that DEC will confirm that the licensed shipper has physical protection measures in place to meet 10 CFR 73.67(g)(1)(i). Therefore, the staff finds the requirement, of 10 CFR 73.67(g)(1)(i) to provide advance notification to the receiver of any planned shipments specifying the mode of transport, estimated time of arrival, location of the nuclear material transfer point, name of carrier and transport identification, would be met.

Applicable Requirement: 10 CFR 73.67(g)(1)(ii), “In-transit requirements for special nuclear material of low strategic significance. (1) Each licensee who transports or who delivers to a carrier for transport special nuclear material of low strategic significance shall: (ii) Receive confirmation from the receiver prior to commencement of the planned shipment that the receiver will be ready to accept the shipment at the planned time and location and acknowledges the specified mode of transport. . .”

The applicant included a description of how it intended to meet the in-transit physical protection requirements of 10 CFR 73.67(g) in Section 6 of their SNMPPP. It stated in the SNMPPP that a SNM-qualified licensed shipper, other than DEC, will be used to transport SNM of low strategic significance both to and from the site. In addition, it is stated in Section 6 of the SNMPPP that DEC will confirm that the licensee used for transport of SNM has “…plans and procedures…” that are developed and implemented in such a manner that 10 CFR 73.67(g)(1)(ii) will be met. Because DEC will be using a SNM-qualified licensee to perform the shipment of SNM of low strategic significance and will confirm that such a licensee has the physical protection measures in place to meet 10 CFR 73.67(g)(1)(ii), subsequently that SNM-qualified licensee will have the ability to meet the requirement of receiving confirmation from the receiver prior to commencement of the planned shipment that the receiver will be ready to accept the shipment at the planned time and location and acknowledges the specified mode of transport.

The DEC application stated that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance, and that DEC will confirm that the licensed shipper has physical protection measures in place to meet 10 CFR 73.67(g)(1)(ii). Therefore, the staff finds the requirement, of 10 CFR 73.67(g)(1)(ii) to receive confirmation from the receiver prior to commencement of the planned shipment that the receiver will be ready to accept the shipment at the planned time and location and acknowledges the specified mode of transport, would be met.

Applicable Requirement: 10 CFR 73.67(g)(1)(iii), "In-transit requirements for special nuclear material of low strategic significance. (1) Each licensee who transports or who delivers to a carrier for transport special nuclear material of low strategic significance shall: (iii) Transport the material in a tamper indicating sealed container. . ."

The applicant included a description of how it intended to meet the in-transit physical protection requirements of 10 CFR 73.67(g) in Section 6 of their SNMPPP. It is stated in the SNMPPP that a SNM-qualified licensed shipper, other than DEC, will be used for transport of SNM of low strategic significance both to and from the site. In addition, it is stated in Section 6 of the SNMPPP that DEC will confirm that the licensee used for transport of SNM has "…plans and procedures…" that are developed and implemented in such a manner that 10 CFR 73.67(g)(1)(iii) will be met.

The DEC application stated that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance, and that DEC will confirm that the licensed shipper has physical protection measures in place to meet 10 CFR 73.67(g)(1)(iii). Therefore, the staff finds the requirement, of 10 CFR 73.67(g)(1)(iii) to transport the material in a tamper indicating sealed container, would be met.

Applicable Requirement: 10 CFR 73.67(g)(2)(i), "In-transit requirements for special nuclear material of low strategic significance. (2) Each licensee who receives quantities and types of special nuclear material of low strategic significance shall: (i) Check the integrity of the containers and seals upon receipt of the shipment,…"

The applicant included a description of how it intended to meet the in-transit physical protection requirements of 10 CFR 73.67(g) in Section 6 of their SNMPPP. Specifically, in Section 5.1, within Subsections 5.1.1.4 (for non-fuel SNM) and 5.1.2.4 (for fuel SNM), it is described that the integrity of both shipping containers and tamper-seals will be checked.

The DEC application described that shipment containers and tamper-seals applied to those containers would be checked upon receipt; therefore, the staff finds the requirement, of 10 CFR 73.67(g)(2)(i) to check the integrity of the containers and seals upon receipt of the shipment, would be met.

Applicable Requirement: 10 CFR 73.67(g)(2)(ii), "In-transit requirements for special nuclear material of low strategic significance. (2) Each licensee who receives quantities and types of special nuclear material of low strategic significance shall: (ii) Notify the shipper of receipt of the material as required in Section 74.15 of this chapter. . ."

In the DEC SNMPPP in Section 5.1, within Subsections 5.1.1.1 (for non-fuel SNM) and 5.1.2.1 (for fuel SNM), it is described that the shipper would be notified in accordance with 10 CFR 74.15. In addition, the development of procedures for "Receiving and shipping SNM" was described in the SNMPP within Section 4.1.

The DEC application described that the shipper would be notified in accordance with 10 CFR 74.15 for both non-fuel and fuel SNM; therefore, the staff finds the requirement, of 10 CFR 73.67(g)(2)(ii) to notify the shipper of receipt of SNM, as required per 10 CFR 74.15, would be met.

Applicable Requirement: 10 CFR 73.67(g)(2)(iii), "Arrange for the in-transit physical protection of the material in accordance with the requirements of Section 73.67(g)(3) of this part, unless the shipper is a licensee and has agreed in writing to arrange for the in-transit physical protection."

The applicant included a description of how it intended to meet the in-transit physical protection requirements of 10 CFR 73.67(g) in Section 6 of their SNMPPP. It is stated in the SNMPPP that a SNM-qualified licensed shipper, other than DEC, will be used for transport of SNM of low strategic significance both to and from the site. In addition, it is stated in Section 6 of the SNMPPP that DEC will confirm that the licensee used for transport of SNM has "…plans and procedures…" that are developed and implemented in such a manner that 10 CFR 73.67(g)(2)(iii) will be met.

The DEC application stated that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance, and that DEC will confirm that the licensed shipper has provisions in place to meet10 CFR 73.67(g)(2)(iii). Therefore, the staff finds the requirement, of 10 CFR 73.67(g)(2)(iii) to arrange for the in-transit physical protection of the material in accordance with the requirements of Section 73.67(g)(3) of this part, unless the shipper is a licensee and has agreed in writing to arrange for the in-transit physical protection, would be met.

Applicable Requirement: 10 CFR 73.67(g)(3), "Each licensee, either shipper or receiver, who arranges for the physical protection of special nuclear material of low strategic significance while in transit or who takes delivery of such material free on board (f.o.b.) the point at which it is delivered to a carrier for transport shall: (i) Establish and maintain response procedures for dealing with threats or thefts of this material. The licensee shall retain a copy of the current response procedures as a record for 3 years after the close of period for which the licensee possesses the special nuclear material under each license for which the procedures were established. Copies of superseded material must be retained for 3 years after each change."

The applicant included a description of how the how it was intended to meet the in-transit physical protection requirements of 10 CFR 73.67(g) in Section 6 of their SNMPPP. It is stated in the SNMPPP that a SNM-qualified licensed shipper, other than DEC, will be used for transport of SNM of low strategic significance both to and from the site. In addition, it is stated in Section 6 of the SNMPPP that DEC will confirm that the licensee used for transport of SNM has "…plans and procedures…" that are developed and implemented in such a manner that 10 CFR 73.67(g)(3)(i) will be met.

The DEC application stated that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance, and that DEC will confirm that the licensed shipper has provisions in place to meet 10 CFR 73.67(g)(3)(i). Therefore, the staff finds the requirement, of 10 CFR 73.67(g)(3)(i) to, "Establish and maintain response procedures …," would be met.

Applicable Requirement: 10 CFR 73.67(g)(3), "Each licensee, either shipper or receiver, who arranges for the physical protection of special nuclear material of low strategic significance while in transit or who takes delivery of such material free on board (f.o.b.) the point at which it is delivered to a carrier for transport shall: (ii) Make arrangements to be notified immediately of the arrival of the shipment at its destination point, or of any

shipment that is lost or unaccounted for after the estimated time of arrival at its destination."

The applicant included a description of how it intended to meet the in-transit physical protection requirements of 10 CFR 73.67(g) in Section 6 of their SNMPPP. It is stated in the SNMPPP that a SNM-qualified licensed shipper, other than DEC, will be used for transport of SNM of low strategic significance both to and from the site. In addition, it is stated in Section 6 of the SNMPPP that DEC will confirm that the licensee used for transport of SNM has "…plans and procedures…" that are developed and implemented in such a manner that 10 CFR 73.67(g)(3)(ii) will be met. The SNMPP stated that DEC will use an SNM licensed shipper and that DEC will verify that the shipper will be able to meet the requirement.

The DEC application stated that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance, and that DEC will confirm that the licensed shipper has provisions in place to meet10 CFR 73.67(g)(3)(ii). Therefore, the staff finds the requirement, of 10 CFR 73.67(g)(3)(ii) to, "make arrangements to be notified immediately of the arrival of the shipment at its destination point, or of any shipment that is lost or unaccounted for after the estimated time of arrival at its destination," would be met.

Applicable Requirement: 10 CFR 73.67(g)(3), "Each licensee, either shipper or receiver, who arranges for the physical protection of special nuclear material of low strategic significance while in transit or who takes delivery of such material free on board (f.o.b.) the point at which it is delivered to a carrier for transport shall: (iii) Conduct immediately a trace investigation of any shipment that is lost or unaccounted for after the estimated arrival time and notify the NRC Operations Center within one hour after the discovery of the loss of the shipment and within one hour after recovery of or accounting for such lost shipment in accordance with the provisions of Section 73.71 of this part."

The applicant included a description of how it intended to meet the in-transit physical protection requirements of 10 CFR 73.67(g) in Section 6 of their SNMPPP. It is stated in the SNMPPP that a SNM-qualified licensed shipper, other than DEC, will be used for transport of SNM of low strategic significance both to and from the site. In addition, it is stated in Section 6 of the SNMPPP that DEC will confirm that the licensee used for transport of SNM has "…plans and procedures…" that are developed and implemented in such a manner that 10 CFR 73.67(g)(3)(iii) will be met. DEC has committed to meeting the requirement in their SNMPPP in Section 5.1, within Subsections 5.1.1.1 (for non-fuel SNM) and 5.1.2.1 (for fuel SNM). Furthermore, DEC noted that a procedure would be developed for notification process in Section 4.1 of the SNMPPP.

The DEC application stated that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance, and that DEC will confirm that the licensed shipper has provisions in place to meet 10 CFR 73.67(g)(3)(iii). In addition, DEC has committed to meeting the 10 CFR 73.67(g)(3)(iii) trace investigation/notification requirement. Therefore, the staff finds the requirement, of 10 CFR 73.67(g)(3)(iii) to, "conduct immediately a trace investigation of any shipment that is lost or unaccounted for after the estimated arrival time and notify the NRC Operations Center within one hour after the discovery of the loss of the shipment and within one hour after recovery of or accounting for such lost shipment in accordance with the provisions of Section 73.71 of this part," would be met.

Applicable Requirement: 10 CFR 73.67(g)(4), "Each licensee who exports special nuclear material of low strategic significance shall comply with the appropriate requirements

specified in paragraphs (c) and (g) (1) and (3) of this section. The licensee shall retain each record required by these sections for 3 years after the close of period for which the licensee possesses the special nuclear material under each license that authorizes the licensee to export this material. Copies of superseded material must be retained for 3 years after each change."

How the requirements of 10 CFR 73.67(c) would be met by the applicant are described in Section 1.5.5.1.4.5, "In-Transit General Requirements," of this safety evaluation report. Also, the applicant included a description of how it intended to meet the in- transit physical protection requirements of 10 CFR 73.67(g) in Section 6 of their SNMPPP. It is stated in the SNMPPP that a SNM-qualified licensed shipper, other than DEC, will be used for transport of SNM of low strategic significance both to and from the site. In addition, it is stated in Section 6 of the SNMPPP that DEC will confirm that the licensee used for transport of SNM has "…plans and procedures…" that are developed and implemented in such a manner that 10 CFR 73.67(g)(4) will be met.

The DEC application stated that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance, and that DEC will confirm that the licensed- shipper has provisions in place to meet 10 CFR 73.67(c) requirements, as specified in the SNMPPP Section 6.1. How the requirements of 10 CFR 73.67(g)(1) and (3) would be met are detailed in this safety evaluation report in Section 1.5.5.1.4.6, "In-Transit Physical Protection Requirements." Therefore, the staff finds the requirement, of 10 CFR 73.67(g)(4), "Each licensee who exports special nuclear material of low strategic significance shall comply with the appropriate requirements specified in paragraphs (c) and (g) (1) and (3) of this section. The licensee shall retain each record required by these sections for 3 years after the close of period for which the licensee possesses the special nuclear material under each license that authorizes the licensee to export this material. Copies of superseded material must be retained for 3 years after each change…," would be met.

Applicable Requirement: 10 CFR 73.67(g)(5)(i), "Each licensee who imports special nuclear material of low strategic significance shall: (i) Comply with the requirements specified in paragraphs (c) and (g) (2) and (3) of this section and retain each record required by these paragraphs for 3 years after the close of period for which the licensee possesses the special nuclear material under each license that authorizes the licensee to import this material. Copies of superseded material must be retained for 3 years after each change."

The applicant included a description of how it intended to meet the in- transit physical protection requirements of 10 CFR 73.67(g) in Section 6 of their SNMPPP. It is stated in the SNMPPP that a SNM-qualified licensed shipper, other than DEC, will be used for transport of SNM of low strategic significance both to and from the site. In addition, it is stated in Section 6 of the SNMPPP that DEC will confirm that the licensee used for transport of SNM has "…plans and procedures…" that are developed and implemented in such a manner that 10 CFR 73.67(g)(4) will be met.

The DEC application stated that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance, and that DEC will confirm that the licensed shipper has provisions in place to meet 10 CFR 73.67(c) requirements, as specified in the SNMPPP Section 6.1. How the requirements of 10 CFR 73.67(c) would be met by the applicant are described in Section 1.5.5.1.4.5, "In-Transit General Requirements," of this safety evaluation report. In addition, how the requirements of 10 CFR 73.67(g)(2) and (3) would be met are detailed in this safety evaluation report in Section 1.5.5.1.4.6, "In-Transit Physical Protection Requirements." Therefore, the staff finds the requirement, of 10 CFR 73.67(g)(5), "Each licensee who exports special nuclear material of low strategic significance shall comply with the appropriate requirements specified in paragraphs (c) and (g) (2) and (3) of this section. The licensee shall retain each record required by these sections for 3 years after the close of period for which the licensee possesses the special nuclear material under each license that authorizes the licensee to export this material. Copies of superseded material must be retained for 3 years after each change," would be met.

Applicable Requirement: 10 CFR 73.67(g)(5)(ii), "Each licensee who imports special nuclear material of low strategic significance shall: (ii) Notify the person who delivered the material to a carrier for transport of the arrival of such material."

The applicant included a description of how it intended to meet the in- transit physical protection requirements of 10 CFR 73.67(g) in Section 6 of their SNMPPP. Specifically, in the DEC SNMPPP in Section 5.1, within Subsections 5.1.1.1 (for non-fuel SNM) and 5.1.2.1 (for fuel SNM), it is described that the shipper would be notified upon receipt of SNM. In addition, the development of procedures for "Receiving and shipping SNM" was described in Section 4.1 of the SNMPPP.

The staff finds that because DEC has described: 1) notification actions to be made upon the receipt of SNM in their SNMPPP, and 2) the development of procedures that would pertain to "Receiving and shipping SNM" the requirement, of 10 CFR 73.67(g)(5)(ii) to "notify the person who delivered the material to a carrier for transport of the arrival of such material," would be met.

1.5.5.1.4.7 *WLS COL FSAR Section 13.5.2.2.8*

The applicant included in WLS COL FSAR Section 13.5.2.2.8, in general terms, the correct manner in which the requirements of 10 CFR 73.67 must be applied to the non-fuel HEU sources that are SNM of low strategic significance, that the applicant proposes to possess, transport and use at the Lee site. Therefore, the staff finds the application of the correct physical protection measures, as stated in 10 CFR 73.67, to all types of SNM of low strategic significance, would be met.

1.5.5.1.4.8 *Post-September 11, 2001, Security Orders for SNM of Low Strategic Significance*

Applicable Requirement: "General Performance Objectives and Requirements" Analysis required per the order.

The applicant considered the order and assessed that only parts C and D of the order must be addressed. The discussion of the analysis that justified only part C and D of the order needed to be addressed was in the SNMPPP within Section 1. Therefore, the analysis requirement presented in the beginning of the order, would be met.

Part C of the Order "Response"

Applicable Requirement: Part C.1. of the order "Develop security response procedures…"

The applicant described the procedures that would be developed in Section 4.1 of the SNMPPP. Those procedures listed to be developed; included response procedures.

Because the applicant committed to develop response implementing procedures, the order requirement of C.1., would be met.

Applicable Requirement: Part C.2. of the order (Part C.2. contains safeguards information and is not described here).

The applicant addressed Part C.2. in Section 5.10 of the SNMPPP.

Because the applicant described the response attributes that aligned with Part C.2. of the order, the order requirement of C.2., would be met.

Part D of the Order "General"

Applicable Requirement: Part D.1. of the order "…hexafluoride…"

This part of the order was associated with uranium hexafluoride. The applicant addressed this order requirement in Section 1 of the SNMPPP.

Because the applicant described the conditions associated with uranium hexafluoride with the Lee site, Part D.1., of the order, would be met.

Applicable Requirement: Part D.2. of the order "…hazardous material…" This part of the order was associated with hazardous material.

The applicant addressed this order requirement in Section 5.9 "Chemicals and Hazardous Materials." In addition, a procedure to implement the strategy outlined in Section 5.9 of the SNMPPP was committed to be developed in Section 4.1 of the SNMPPP.

Because the applicant described a strategy to address Part D.2. of the order, and committed to the development of a procedure to implement that strategy, Part D.2. of the order would be met.

Applicable Requirement: Part D.3. of the order "Supplement the Emergency Action Levels…"

The applicant addressed Part D.3 of the order in section 5.11 "Emergency Response" of the SNMPPP.

Because the applicant described how the requirement of Part D.3. of the order would be addressed, Part D.3. of the order, would be met.

Applicable Requirement: Part D.4. of the order "Evaluate computer and communications…"

The applicant addressed Part D.4. of the order in Section 5.11 "Emergency Response" of the SNMPPP.

Because the applicant described how the requirement of Part D.4. of the order would be addressed, Part D.4. of the order, would be met.

Applicable Requirement: Part D.5. of the order "Evaluate capabilities…fire suppression…"

The applicant addressed Part D.5. of the order in Section 5.12 "Fire Response" of the SNMPPP.

Because the applicant described how the requirement of Part D.5, of the order would be addressed, Part D.5 of the order would be met.

Applicable Requirement: Part D.6. of the order "Evaluate…medical…"

The applicant addressed Part D.6. of the order in Section 5.13 "Medical Response" of the SNMPPP.

Because the applicant described how the requirement of Part D.6, of the order would be addressed, Part D.6 of the order would be met.

Applicable Requirement: Part D.7. of the order "Limit…access…"

The applicant discussed in Section 5.7 how the order requirement D.7 would be addressed.

Because the applicant described how the requirement of Part D.7, of the order would be addressed, Part D.7 of the order would be met.

Part 3 of the Order "Access Control and Badging"

The applicant stated in Section 5.4 "Access Control and Badging" of the SNMPPP that those persons afforded access to the controlled access area would be under the access authorization program as presented in Section 14.1 of their power reactor PSP. In Section 14.1 of the PSP, Rev. 3, dated April 10, 2013, (SLES No. NS112930), the RG 5.66 "Access Authorization Program for Nuclear Power Plants" was the applicable access authorization program. The access authorization program as described in RG 5.66 includes fingerprinting and an overall more-stringent access authorization program than that described in Part 3 of the order.

In addition, individuals not under the subject access authorization program would be escorted into, out of, and within the controlled access area in accordance with Section 14.4.6 of the PSP which describes escort methodologies developed for the Lee power reactors.

The applicant described that RG 5.66 would be applied to meet Part 3 of the order and it is recognized that in doing so a more stringent access authorization process would be utilized than that described in Part 3 of the order. Therefore, Part 3 of the order, which includes fingerprinting and other access authorization provisions, would be met.

Conclusion

The NRC staff reviewed DEC, WLS Units I and 2, Docket Nos. 52-018 and 52-019, application (ADAMS Accession No. ML073510494), and finds that the applicable requirements specified in 10 CFR 73.67, "Licensee fixed site and in-transit requirements for the physical protection of SNM of moderate and low strategic significance" and the post September 11, 2001, security order for SNM of low strategic significance, would be met.

1.5.5.2 *Conclusion and Post Combined License Activities*

The NRC staff reviewed DEC, WLS Units I and 2, Docket Nos. 52-018 and 52-019, application (ADAMS Accession No. ML073510494), and finds that the applicable requirements specified in 10 CFR 73.67, "Licensee fixed site and in-transit requirements for the physical protection of SNM of moderate and low strategic significance" and the post September 11, 2001, security order for SNM of low strategic significance, would be met.

With respect to the applicable fixed site and in-transit physical protection requirements for SNM of low strategic significance specified in 10 CFR 73.67 and the post September 11, 2001, security order for the fixed site possession and use of SNM of low strategic significance, the NRC staff reviewed application and concludes that the relevant information in the application is acceptable because it meets the pertinent recommendations stated in RG 5.59.

The license condition language in this section has been modified, per a letter from the applicant dated March 22, 2016 (ADAMS Accession No. ML16084A099), confirming the acceptability of the following license conditions proposed by the staff. These changes do not affect the staff's above analysis of the conditions, and therefore, for the reasons discussed in the technical evaluation section above, the staff finds the following license conditions acceptable:

- License Condition (1-3) – Subject to the conditions and requirements incorporated herein, the Commission hereby licenses DEC:

    (1) (a) pursuant to the Act and 10 CFR Part 70, to receive and possess at any time, special nuclear material as reactor fuel, in accordance with the limitations for storage and in amounts necessary for reactor operation, described in the FSAR, as supplemented and amended;

    (b) pursuant to the Act and 10 CFR Part 70, to use special nuclear material as reactor fuel, after a Commission finding under 10 CFR 52.103(g) has been made, in accordance with the limitations for storage and amounts necessary for reactor operation, described in the FSAR, as supplemented and amended;

    (2) (a) pursuant to the Act and 10 CFR Parts 30 and 70, to receive, possess, and use, at any time before a Commission finding under 10 CFR 52.103(g), such byproduct and special nuclear material (but not uranium hexafluoride) as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts not exceeding those specified in 10 CFR 30.35(d) and 10 CFR 70.25(d) for establishing decommissioning financial assurance, and not exceeding those specified in 10 CFR 30.72 and 10 CFR 70.22(i)(1);

    (b) pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use, after a Commission finding under 10 CFR 52.103(g), any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as necessary;

    (3) (a) pursuant to the Act and 10 CFR Parts 30 and 70, to receive, possess, and use, before a Commission finding under 10 CFR 52.103(g), any byproduct or special nuclear material (but not uranium hexafluoride) that is (1) in unsealed form; (2) on foils or plated surfaces, or (3) sealed in glass, for sample analysis or instrument calibration or other activity associated with radioactive apparatus or components, in amounts not exceeding those specified

in 10 CFR 30.35(d) and 10 CFR 70.25(d) for establishing decommissioning financial assurance, and not exceeding those specified in 10 CFR 30.72 and 10 CFR 70.22(i)(1);

(b) pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use, after a Commission finding under 10 CFR 52.103(g), in amounts as necessary, any byproduct, source, or special nuclear material (but not uranium hexafluoride) without restriction as to chemical or physical form, for sample analysis or instrument calibration or other activity associated with radioactive apparatus or components but not uranium hexafluoride; and

(4) pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

- License Condition (1-4) - Prior to initial receipt of special nuclear materials onsite, the licensee shall implement the Special Nuclear Material Control and Accounting Program. No later than 12 months after issuance of the COL the licensee shall submit to the Director of the Office of New Reactors a schedule that supports planning for and conduct of NRC inspections of the Special Nuclear Material Control and Accounting Program. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until the Special Nuclear Material Control and Accounting Program has been fully implemented.

- License Condition (1-5) – No later than 12 months after issuance of the COL, the licensee shall submit to the Director of the Office of New Reactors a schedule that supports planning for and conduct of NRC inspection of the non-licensed plant staff training program. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until the non-licensed plant staff training program has been fully implemented.

- License Condition (1-6) – Prior to initial receipt of special nuclear material on site, the licensee shall implement the Special Nuclear Material Physical Protection Program. No later than 12 months after issuance of the COL, the licensee shall submit to the Director of the Office of New Reactors a schedule that supports planning for and conduct of NRC inspection of the Special Nuclear Material Physical Protection Program. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until the Special Nuclear Material Physical Protection Program has been fully implemented.

1.5.6 Physical Protection of Category 1 and Category 2 Quantities of Radioactive Material

On March 19, 2013, a new 10 CFR Part 37 rule was published in the FR in which the NRC amended its regulations to establish security requirements for the use and transport of Category 1 and Category 2 quantities of radioactive material. The NRC considers these quantities to be risk significant and, therefore, to warrant additional protection. Category 1 and Category 2 thresholds are based on the quantities established by the International Atomic Energy Agency (IAEA) in its Code of Conduct on the Safety and Security of Radioactive Sources, which the NRC endorses. The objective of the 10 CFR Part 37, “Physical Protection of Category 1 and Category 2 Quantities of Radioactive Material," regulation is to provide reasonable assurance of preventing the theft or diversion of Category 1 and Category 2 quantities of radioactive material. The regulations also include security requirements for the transportation of irradiated reactor fuel that weighs 100 grams or less in net weight of irradiated fuel.

2 SITE CHARACTERISTICS

Chapter 2, "Site Characteristics," of the William States Lee III Nuclear Station (WLS) Final Safety Analysis Report (FSAR) addresses the geological, seismological, hydrological, and meteorological characteristics of the site and vicinity, in conjunction with present and projected population distribution and land use, and site activities and controls.

2.0 Introduction

2.0.1 Introduction

The site characteristics are reviewed by the U.S. Nuclear Regulatory Commission (NRC) staff (the staff) to determine whether the applicant has accurately described the site characteristics and site parameters together with site-related design parameters and design characteristics in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, "Licenses, certifications, and approvals for nuclear power plants." The review is focused on the site characteristics and site-related design characteristics needed to enable the staff to reach a conclusion on all safety matters related to siting of WLS Units 1 and 2. Since this combined license (COL) application references the AP1000 design certification (DC), this section focuses on the applicant's demonstration that the characteristics of the site fall within the site parameters specified in the DC rule or, if outside the site parameters, that the design satisfies the requirements imposed by the specific site characteristics and conforms to the design commitments and acceptance criteria described in the AP1000 Design Control Document (DCD).

2.0.2 Summary of Application

WLS COL FSAR, Revision 11, Section 2.0 incorporates by reference AP1000 DCD, Revision 19, Chapter 2, including Section 2.0. In this section, the applicant provided WLS COL FSAR Table 2.0-201, which provides a comparison of the AP1000 DCD site parameters against WLS site parameters. In addition, in WLS COL FSAR Section 2.0, the applicant provided the following:

Departures

- WLS DEP 2.0-1

The applicant departed from the AP1000 standard plant certified seismic design response spectra (CSDRS) that is addressed in AP1000 DCD, Section 3.7.1.1. The application deviated from the CSDRS due to a new seismic model for the local area described in NUREG-2115, "Central and Eastern United States Seismic Source Characterization for Nuclear Facilities." Considering this new seismic model and updated seismic hazards and site-specific foundation response spectra resulted in exceeding the AP1000 CSDRS and the need for a departure from the AP1000 DCD. The sections affected by this departure are WLS COL FSAR Table 2.0-201, Sections 3.7.1.1.1, 3.7.2.8.4, and 3.7.2.15, Appendix 3I, and Section 19.55.6.3. Sections in other chapters of the WLS COL FSAR will be discussed in the appropriate chapters of this report.

Supplemental Information

- WLS SUP 2.0-1

The applicant provided supplemental information in WLS COL FSAR Section 2.0, "Site Characteristics," which describes the characteristics and site-related design parameters of WLS.

Interface Items

These items are listed in the AP1000 DCD as plant interfaces.

- Interface Item 2.1

The WLS COL FSAR addresses the envelope of AP1000 plant site-related parameters in Tables 2.0-201 and 2.0-202.

- Interface Item 2.2

The WLS COL FSAR addresses external missiles from manmade hazards and accidents in Table 2.0-201, Sections 2.2.3.1.1 and 3.5.

- Interface Item 2.3

The WLS COL FSAR addresses maximum loads from manmade hazards and accidents in Section 2.2.3.

- Interface Item 2.4

The WLS COL FSAR addresses limiting meteorological parameters for accidents, releases, and extreme conditions, related to the design of systems and components exposed to the environment. This information is found in WLS COL FSAR Tables 2.0-201 and 2.0-202.

- Interface Item 2.5

The WLS COL FSAR addresses tornado and operating basis wind loadings in Table 2.0-201.

- Interface Item 2.6

The WLS COL FSAR addresses external missiles generated by natural phenomena in Table 2.0-201.

- Interface Item 2.7

The WLS COL FSAR addresses snow, ice, and rain loads in Table 2.0-201.

- Interface Item 2.8

The WLS COL FSAR addresses ambient air temperatures in Table 2.0-201.

- Interface Item 2.9

The WLS COL FSAR addresses onsite meteorological measurement program in Section 2.3.3.

- Interface Item 2.10

The WLS COL FSAR addresses Flood and groundwater elevations in Table 2.0-201.

- Interface Item 2.11

The WLS COL FSAR addresses Hydrostatic loads on systems, components, and structures in Table 2.0-201 and Section 2.4.12.5.

- Interface Item 2.12

The WLS COL FSAR addresses seismic parameters such as peak ground acceleration, response spectra, and shear wave velocity, in Table 2.0-201.

- Interface Item 2.13

The WLS COL FSAR addresses required bearing capacity of foundation materials in Table 2.0-201.

2.0.3 Regulatory Basis

The information in the WLS COL FSAR that is incorporated by reference, the regulatory basis is addressed in NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," and its supplements. The applicable regulatory requirements for site characteristics are as follows:

- 10 CFR 52.79(a)(1)(i) through (vi), as it relates to the requirements for the site-related contents of the application.

- 10 CFR 52.79(d)(1), as it relates to information sufficient to demonstrate that the characteristics of the site fall within the site parameters specified in the DC.

- 10 CFR Part 100, "Reactor site criteria," as it relates to the siting factors and criteria for determining an acceptable site.

The related acceptance criteria associated with these site characteristic requirements are given in NUREG 0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," Section 2.0. The related acceptance criteria are as follows:

- The acceptance criteria associated with specific site characteristics/parameters and site-related design characteristics/parameters are addressed in the related Chapter 2 or other referenced sections of NUREG-0800.

- Acceptance is based on the applicant's demonstration that the characteristics of the site fall within the site parameters of the certified design. If the actual site characteristics do not fall within the certified standard design site parameters, the COL applicant provides sufficient justification (e.g., by request for exemption or amendment from the DC) that the proposed facility is acceptable at the proposed site.

The regulatory requirements associated with the Tier 1 and 2 departures and the exemption request are as follows:

- 10 CFR 50.12(a), “Specific exemptions.”

  > (a) The Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of the regulations of this part, which are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. The Commission will not consider granting an exemption unless special circumstances are present.

- 10 CFR Part 52, “Licenses, certifications, and approvals for nuclear power plants,” Appendix D, “Design certification rule for the AP1000 design,” Section VIII, “Processes for changes and departures,” Item B.5.

- 10 CFR Part 52, Appendix D, Section IV.A.2.d.

  > An applicant for a combined license that wishes to reference this appendix shall…comply with the following requirements: Include, as part of its application…Information demonstrating compliance with the site parameters and interface requirements.

- 10 CFR 52, Appendix D, Section VIII.A.4. This section states that exemptions from Tier 1 material are governed by 10 CFR 52.63(b)(1). 10 CFR 52.63(b)(1) references 10 CFR 52.7, “Specific exemptions.”

- 10 CFR 52.7. This section states that the Commission may grant exemptions from the requirements of the regulations of this part as governed by 10 CFR 50.12, “Specific exemptions,” of this chapter.

2.0.4 Technical Evaluation

The staff reviewed WLS COL FSAR Section 2.0 and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The staff’s review confirmed that the information in the application and incorporated by reference addresses the required information relating to site

[1] See Section 1.2.2 of the staff’s Safety Evaluation Report for a discussion of the staff’s review related to verification of the scope of information to be included in a COL application that references a DC.

characteristics. The results of the staff's evaluation of the information incorporated by reference in the application are documented in NUREG-1793 and its supplements. The staff reviewed the following information in the WLS COL FSAR:

Supplemental Information

- WLS SUP 2.0-1

The staff reviewed supplemental information WLS SUP 2.0-1 in WLS COL FSAR Section 2.0 describing the characteristics and site-related design parameters of WLS Units 1 and 2. In the WLS COL FSAR, the site parameters in AP1000 DCD Table 2-1 are compared to the site-specific site characteristics in WLS COL FSAR Table 2.0-201. In addition, control room atmospheric dispersion factors for accident dose analysis are presented in WLS COL FSAR Table 2.0-201.

The staff reviewed and compared the site-specific characteristics included in WLS COL FSAR Table 2.0-201 against AP1000 DCD Table 2-1. The staff's evaluation of the site characteristics associated with air temperature, precipitation, wind speed, atmospheric dispersion values, and control room atmospheric dispersion values is addressed in Section 2.3 of this report.
The staff's evaluation of site characteristics associated with flood level, ground water level, and plant grade elevation is addressed in Section 2.4 of this report. The staff's evaluation of seismic and soil site characteristics is addressed in Section 2.5 of this report. The staff's evaluation of site characteristics associated with missiles is addressed in Section 3.5 of this report.

2.0.5 Post Combined License Activities

There are no post COL activities related to this section.

2.0.6 Conclusion

The staff reviewed the WLS COL application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to site characteristics, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the application are documented in NUREG-1793 and its supplements.

As set forth above, the staff reviewed the application to ensure that sufficient information was presented in WLS SUP 2.0-1 to demonstrate that the characteristics of the site fall within the site parameters specified in the DC.

2.1 Geography and Demography

2.1.1 Site Location and Description

2.1.1.1 *Introduction*

The descriptions of the site area and reactor location are used to assess the acceptability of the reactor site. The review covers the following specific areas: (1) specification of reactor location with respect to latitude and longitude, political subdivisions; and prominent natural and manmade features of the area; (2) site area map to determine the distance from the reactor to the boundary lines of the exclusion area, including consideration of the location, distance, and orientation of plant structures with respect to highways, railroads, and waterways that traverse or lie adjacent to the exclusion area; and (3) any additional information requirements prescribed in the "Contents of application" sections of the applicable subparts to 10 CFR Part 52. The purpose of the review is to ascertain the accuracy of the applicant's description for use in independent evaluations of the exclusion area authority and control, the surrounding population, and nearby manmade hazards.

2.1.1.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 2.1, incorporates by reference AP1000 DC, Revision 19, Section 2.1. In addition, in WLS COL FSAR Section 2.1, the applicant provided the following:

Departures

- STD DEP 1.1-1

The applicant proposed the following Tier 2 standard departure from the AP1000 DCD. WLS COL FSAR Section 2.1.1, "Site Location and Description," identifies instances where the WLS COL FSAR sections are renumbered to include content consistent with Regulatory Guide (RG) 1.206, as well as NUREG-0800. WLS COL FSAR Section 2.1.4, "Combined License Information for Geography and Demography," references WLS COL 2.1-1 as discussed in WLS COL FSAR Sections 2.1.1, 2.1.2, and 2.1.3.

AP1000 COL Information Item

- WLS COL 2.1-1

The applicant provided additional information to resolve WLS COL 2.1-1, which addresses the provision of site-specific information related to site location and description, including the site exclusion area and its control, population, natural and manmade features, highways, railways, waterways, and other significant features of the area. WLS COL 2.1-1 is discussed in WLS COL FSAR Sections 2.1.1, 2.1.2, and 2.1.3.

2.1.1.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. In addition, the acceptance criteria associated with the relevant requirements of NRC regulations for the site location and description are given in NUREG-0800, Section 2.1.1.

The applicable regulatory requirements for identifying site location and description are:

- 10 CFR 50.34(a)(1) and 10 CFR 52.79(a)(1),as they relate to the inclusion in the safety analysis report (SAR) of a detailed description and safety assessment of the site on which the facility is to be located, with appropriate attention to features affecting facility design.
- 10 CFR Part 100, as it relates to the following: (1) defining an exclusion area and setting forth requirements regarding activities in that area (10 CFR 100.3, "Definitions)); (2) addressing and evaluating factors that are used in determining the acceptability of the site as identified in 10 CFR 100.20(b); (3) determining an exclusion area such that certain dose limits would not be exceeded in the event of a postulated fission product release as identified in 10 CFR 50.34(a)(1), as it relates to site evaluation factors identified in 10 CFR Part 100; and (4) requiring that the site location and the engineered features included as safeguards against the hazardous consequences of an accident, should one occur, would ensure a low risk of public exposure.

The related acceptance criteria from NUREG-0800, Section 2.1.1 are as follows:

- Specification of Location: The information submitted by the applicant is adequate and meets the requirements of 10 CFR 50.34(a)(1) and 10 CFR 52.79(a)(1) if it describes highways, railroads, and waterways that traverse the exclusion area in sufficient detail to allow the reviewer to determine that the applicant has met the requirements in 10 CFR 100.3.
- Site Area Map: The information submitted by the applicant is adequate and meets the requirements of 10 CFR 50.34(a)(1) and 10 CFR 52.79(a)(1) if it describes the site location, including the exclusion area and the location of the plant within the area, in sufficient detail to enable the reviewer to evaluate the applicant's analysis of a postulated fission product release, thereby allowing the reviewer to determine (in Sections 2.1.2 and 2.1.3, and Chapter 15 of this report) that the applicant has met the requirements of 10 CFR 50.34(a)(1) and 10 CFR Part 100.

2.1.1.4 *Technical Evaluation*

The staff reviewed WLS COL FSAR Section 2.1 and checked the referenced DCD to ensure that the combination of the DCD and the WLS COL application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the site location and description. The results of the staff's evaluation of the information incorporated by reference in the application are documented in NUREG-1793 and its supplements. The staff reviewed the following information in the WLS COL FSAR:

Departures

- STD DEP 1.1-1

The applicant proposed the following Tier 2 standard departure from the AP1000 DCD. WLS COL FSAR Section 2.1.1, "Site Location and Description," identifies instances where the WLS COL FSAR sections are renumbered to include content consistent with RG 1.206, as well as NUREG-0800. This change is acceptable because it does not alter the information required to be provided.

AP1000 COL Information Item

- WLS COL 2.1-1

The staff reviewed WLS COL 2.1-1 related to site location and description, including political subdivisions, natural and manmade features, population, highways, railways, waterways, and other significant features of the area included in WLS COL FSAR Section 2.1.1. COL Information Item 2.1-1 in AP1000 DCD, Section 2.1.1 states:

> Combined License applicants referencing the AP1000 certified design will provide site-specific information related to site location and description, exclusion area authority and control, and population distribution. Site-specific information on the site and its location will include political subdivisions, natural and man-made features, population, highways, railways, waterways, and other significant features of the area.

Using publicly available maps, the staff independently verified the applicant-supplied latitude and longitude. The staff then converted this latitude and longitude to Universal Transverse Mercator (UTM) coordinates for the proposed WLS Units 1 and 2 and used the calculated values to verify the UTM coordinates provided in the WLS COL FSAR. The staff then verified the UTM coordinates of the WLS units.

The staff reviewed the site area map provided in the WLS COL FSAR for the proposed Units 1 and 2 to verify that the distance from the reactor to the boundary line of the exclusion area meets the guidance in NUREG-0800, Section 2.1.1. On the basis of the staff's review of the information in the WLS COL FSAR, and also the staff's confirmatory review of pertinent information, such as the location, site area map, effluent release limit boundaries, and prominent natural and manmade features of the area as described in publicly available documentation, as well as information collected during a site visit, the staff determined the information provided by the applicant with regard to the site location and description is considered adequate and acceptable.

2.1.1.5 *Post Combined License Activities*

There are no post COL activities related to this section.

2.1.1.6 *Conclusion*

The staff reviewed the WLS COL application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to site location and description, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the application are documented in NUREG-1793 and its supplements.

As set forth above, the applicant presented and substantiated information to establish the site location and description. The staff reviewed COL 2.1-1, and for the reasons given above, concludes that it is sufficient for the staff to evaluate compliance with the siting evaluation factors in 10 CFR 100.3, as well as with the radiological consequence evaluation factors in 10 CFR 52.79(a)(1). The staff further concludes that the applicant provided sufficient details about the site location and site description to allow the staff to evaluate, as documented in Sections 2.1.2, 2.1.3, and Chapter 11, Section 13.3, and Chapter 15 of this report, whether the applicant has met the relevant requirements of 10 CFR Part 52.79(a)(1) and 10 CFR Part 100 with respect to determining the acceptability of the site.

2.1.2 Exclusion Area Authority and Control

2.1.2.1 *Introduction*

The descriptions of exclusion area authority and control are used to verify the applicant's legal authority to determine and control activities within the designated exclusion area, as provided in the application, are sufficient to enable the reviewer to assess the acceptability of the reactor site. The review covers the following specific areas: (1) establishment of the applicant's legal authority to determine all activities within the designated exclusion area; (2) the applicant's authority and control in excluding or removing personnel and property in the event of an emergency; (3) establish that proposed or permitted activities in the exclusion area unrelated to operation of the reactor do not result in a significant hazard to public health and safety; and (4) any additional information requirements prescribed within the "Contents of Application" sections of the applicable subparts to 10 CFR Part 52.

2.1.2.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 2.1, incorporates by reference AP1000 DCD, Revision 19, Section 2.1. In addition, in WLS COL FSAR Section 2.1.2, the applicant provided the following:

AP1000 COL Information Item

- WLS COL 2.1-1

The applicant provided additional information to resolve COL Information Item 2.1-1, which addresses the provision of site-specific information related to exclusion area authority and control, including size of the area, and activities that may be permitted within the designated exclusion area.

2.1.2.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. In addition, the acceptance criteria associated with the relevant requirements of NRC regulations for the exclusion area authority and control are given in NUREG-0800, Section 2.1.2.

The applicable regulatory requirements for verifying exclusion area authority and control are:

- 10 CFR 50.34(a)(1), and 10 CFR 52.79(a)(1), as it relates to the inclusion in the FSAR of a detailed description and safety assessment of the site on which the facility is to be located, with appropriate attention to features affecting facility design (10 CFR 50.34(a)(1), and 10 CFR 52.79(a)(1)).

- 10 CFR Part 100, as it relates to the following: (1) defining an exclusion area and setting forth requirements regarding activities in that area (10 CFR 100.3); (2) addressing and evaluating factors that are used in determining the acceptability of the site as identified in 10 CFR 100.20(b); and (3) determining an exclusion area such that certain dose limits would not be exceeded in the event of a postulated fission product release as identified in 10 CFR 50.34(a)(1) as it relates to site evaluation factors identified in 10 CFR Part 100.

The related acceptance criteria from NUREG-0800, Section 2.1.2 are as follows:

- Establishment of Authority for the Exclusion or Removal of Personnel and Property: The information submitted by the applicant is adequate and meets the requirements of 10 CFR 50.33, “Contents of applications; general information”; 10 CFR 50.34(a)(1); 10 CFR 52.79, “Contents of applications; technical information in final safety analysis report”; and 10 CFR Part 100 if it provides sufficient detail to enable the staff to evaluate the applicant’s legal authority for the exclusion or removal of personnel or property from the exclusion area.

- Proposed and Permitted Activities: The information submitted by the applicant is adequate and meets the requirements of 10 CFR 50.33, 10 CFR 50.34(a)(1), 10 CFR 52.79, and 10 CFR Part 100 if it provides sufficient detail to enable the staff to evaluate the applicant’s legal authority over all activities within the designated exclusion area.

2.1.2.4 *Technical Evaluation*

The staff reviewed WLS COL FSAR Section 2.1.2 and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The staff’s review confirmed that the information in the application and incorporated by reference addresses the required information relating to the exclusion area authority and control. The results of the staff’s evaluation of the information incorporated by reference in the application are documented in NUREG-1793 and its supplements. The staff reviewed the following information in the WLS COL FSAR:

AP1000 COL Information Item

- WLS COL 2.1-1

The staff reviewed WLS COL 2.1-1 related to the exclusion area authority and control, including the exclusion area boundary, and activities that may be permitted within the designated exclusion area included in WLS COL FSAR Section 2.1.2. COL Information Item in AP1000 DCD, Section 2.1.1 states:

> Combined License applicants referencing the AP1000 certified design will provide site-specific information related to site location and description, exclusion area authority and control, and population distribution. Site-specific information on the exclusion area will include the size of the area and the exclusion area authority and control. Activity that may be permitted within the exclusion area will be included in the discussion.

The applicant supplied the following information: There are no residences and only limited commercial and recreational activities within the Units 1 and 2 exclusion area. No public highways or active railroads traverse the exclusion area. There are four historical cemeteries within the site boundary area. Access of members of the public to these cemeteries will be controlled by site security personnel.

The staff also verified for consistency that the exclusion area boundary (EAB) is the same as that being considered by the applicant for the radiological consequences in WLS COL FSAR Section 13.3 and Chapter 15. The acceptance criteria of NUREG-0800, Section 2.1.2 states, “Absolute ownership of all lands within the exclusion area, including mineral rights, is considered to carry with it the required authority to determine all activities on this land and is acceptable.” Thus, the staff concludes that the applicant has the required authority to determine and control all activities within the designated exclusion area.

The staff reviewed the resolution to the site-specific items related to the exclusion area authority and control included under WLS COL FSAR Section 2.1. The staff verified the applicant’s description of the exclusion area as well as the authority under which all activities within the exclusion area can be controlled. The staff also verified for consistency that the EAB is the same as that being considered for the radiological consequences by the applicant in WLS COL FSAR Chapters 15 and 13.3. The staff used publicly available maps and satellite pictures, a site visit, and the area map provided in the Unit 1 and 2 WLS COL FSAR to verify that no publicly used transportation mode crosses the EAB; therefore, no arrangements for the control of traffic in the event of an emergency are required. The staff also verified that no public roads cross the exclusion area; therefore, neither relocation nor abandonment of roads is needed. The staff concludes that the applicant has acquired authority to control all activities within the designated exclusion area.

2.1.2.5 *Post Combined License Activities*

There are no post COL activities related to this section.

2.1.2.6 *Conclusion*

The staff reviewed the WLS COL application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to the exclusion area authority and control, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the application are documented in NUREG-1793 and its supplements.

As set forth above, the applicant provided and substantiated information concerning its legal authority and control of all activities within the designated exclusion area. The staff reviewed WLS COL 2.1-1 and, for the reasons given above, concludes that the applicant's exclusion area is acceptable to meet the requirements of 10 CFR 50.34(a)(1), 10 CFR 52.79(a)(1), 10 CFR Part 100, and 10 CFR 100.3 with respect to determining the acceptability of the site. This conclusion is based on the applicant having appropriately described the plant exclusion area, the authority under which all activities within the exclusion area can be controlled, the need for relocation or abandonment of public roads, and the methods by which access and occupancy of the exclusion area can be controlled during normal operation and in the event of an emergency situation. In addition, the applicant has the required authority to control activities within the designated exclusion area, including the admission, exclusion, and removal of persons and property, and has established acceptable methods for control of the designated exclusion area. The staff finds that the application adequately addresses COL Information Item 2.1-1. The staff finds that the applicant has provided sufficient information to satisfy 10 CFR Part 52.79(a)(1) and 10 CFR Part 100 and, therefore, is acceptable.

2.1.3 Population Distribution

2.1.3.1 *Introduction*

The description of population distributions addresses the need for information about: (1) population in the site vicinity, including transient populations; (2) population in the exclusion area; (3) whether special protective measures should be taken on behalf of the populace in the specified low-population zone (LPZ) in the event of a serious accident; (4) whether the nearest boundary of the closest population center containing 25,000 or more residents is at least one and one-third times the distance from the reactor to the outer boundary of the LPZ; (5) whether the population density in the site vicinity is consistent with the guidelines given in RG 4.7, "General Site Suitability Criteria for Nuclear Power Stations;" Regulatory Position C.4 and (6) any additional information requirements prescribed in the "Contents of Application," sections of the applicable subparts to 10 CFR Part 52.

2.1.3.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 2.1, incorporates by reference AP1000 DCD, Revision 19, Section 2.1. In addition, in WLS COL FSAR Section 2.1.3, the applicant provided the following:

AP1000 COL Information Item

- WLS COL 2.1-1

The applicant provided additional information resolve COL Information Item 2.1-1, which addresses the provision of site-specific information related to the site environment, specifically, population centers and distribution up to 80 km (50 mi) from the WLS site.

2.1.3.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. In addition, the acceptance criteria associated with the relevant requirements of NRC regulations for population distribution are given in NUREG-0800, Section 2.1.3. The applicable regulatory requirements for identifying site location and description are:

- 10 CFR 50.34(a)(1), as it relates to consideration of the site evaluation factors identified in 10 CFR 100.3, 10 CFR Part 100 (including consideration of population density), and 10 CFR 52.79, as they relate to provision by the applicant in the SAR of the existing and projected future population profile of the area surrounding the site.

- 10 CFR 100.20 and 10 CFR 100.21, as they relate to determining the acceptability of a site for a power reactor. In 10 CFR 100.3, 10 CFR 100.20(a), and 10 CFR 100.21(b), the NRC provides definitions and other requirements for determining an exclusion area, LPZ, and population center distance.

The related acceptance criteria from Section 2.1.3 of NUREG-0800 are as follows:

- Population Data: The population data supplied by the applicant in the FSAR is acceptable under the following conditions: (1) the FSAR includes population data from the latest census and projected population at the year of plant approval and 5 years thereafter, in the geographical format given in RG 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition),” Section 2.1.3, Revision 3, and in accordance with DG-1145, “Combined License Applications for Nuclear Power Plants (LWR Edition)”; (2) the FSAR describes the methodology and sources used to obtain the population data, including the projections; and (3) the FSAR includes information on transient populations in the site vicinity.

- Exclusion Area: The exclusion area should either not have any residents, or such residents should be subject to ready removal if necessary.

- Low-Population Zone: The specified LPZ is acceptable if it is determined that appropriate protective measures could be taken on behalf of the enclosed populace in the event of a serious accident.

- Nearest Population Center Boundary: The nearest boundary of the closest population center containing 25,000 or more residents is at least one and one-third times the distance from the reactor to the outer boundary of the LPZ.

- Population Density: If the population density exceeds the guidelines given in RG 4.7, Regulatory Position C.4, the applicant must give special attention to the consideration of alternative sites with lower population densities.

2.1.3.4 *Technical Evaluation*

The staff reviewed WLS COL FSAR Section 2.1.3 and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to population distribution. The results of the staff's evaluation of the information incorporated by reference in the application are documented in NUREG-1793 and its supplements. The staff reviewed the information in the WLS COL FSAR:

AP1000 COL Information Item

- WLS COL 2.1-1

The staff reviewed WLS COL 2.1-1 related to the population distribution around the site environs included in WLS COL FSAR Section 2.1.3. COL information item in AP1000 DCD, Section 2.1.1 states:

> Combined License applicants referencing the AP1000 certified design will provide site-specific information related to site location and description, exclusion area authority and control, and population distribution. Site-specific information will be included on population distribution.

The staff reviewed the data on the population in the site environment to determine whether the exclusion area, LPZ, and population center distance for the proposed WLS site comply with the requirements of 10 CFR Part 100. The staff also evaluated whether, consistent with RG 4.7, Regulatory Position C.4, the applicant should consider alternative sites with lower population densities. The staff also reviewed whether appropriate protective measures could be taken on behalf of the enclosed populace within the emergency planning zone (EPZ), which encompasses the LPZ, in the event of a serious accident. The staff compared and verified the applicant's population data against U.S. Census Bureau data available on the Internet. Transient population estimates were based on evaluations of seasonal transient business, hotels, motels, recreation, schools, hospitals, nursing homes, correctional facilities, festivals, and migrant workers populations. The staff reviewed the projected population data provided by the applicant, including the weighted transient population for 2016, 2021, 2026, 2036, 2046, and 2056. The staff reviewed the extensive transient population data provided by the applicant. Based on this information, the staff finds that the applicant's estimate of the population and the population projections are acceptable.

The staff verified the distances to the nearest population centers are well in excess of the minimum population center distance of 4.3 km (2.7 miles) (1 1/3 times the distance from center point to the outer boundary of the LPZ). The LPZ is defined as a two-mile radius from the site center point. The center point is defined as the midway point between Unit 1 and Unit 2. The nearest population center, as defined by 10 CFR 100.3, is Gastonia, NC. The distance to

Gastonia's urban boundary, as defined by US Census files, is 25 km (16 mi) northeast from the center point between the two reactors. Using the county's population projection ratios for both transient and permanent population, results in Gaffney, SC having a total population number greater than 25,000 people. Gaffney's closest boundary, defined by the US Census Bureau, is 10 km (6 mi) northwest from the center point between the two reactors. Both of these distances are greater than one and one-third times the distance from the reactor center point to the boundary of the low population zone as required by NUREG-0800 and comply with the guidance provided by RG 4.7, "General Site Suitability Criteria for Nuclear Power Stations." Therefore, the staff concludes that the proposed site meets the population center distance requirement as defined in 10 CFR Part 100, Subpart B.

The staff evaluated the site population density against the criterion in RG 4.7, Regulatory Position C.4, Revision 2, regarding whether it is necessary to consider alternative sites with lower population densities. The evaluation included the review and verification of whether the population densities at the time of initial site approval (assumed 2016) and 5 years thereafter, would not exceed the criteria of 500 persons per square mile averaged over a radial distance of 32 km (20 mi) (cumulative population at a distance divided by the area at that distance). The staff independently verified the applicant's calculated population density for the years 2016 and 2021 based on the staff's linear calculation projection estimate, and concludes that the population density for the WLS site is well below the criterion. Therefore, the staff finds that the WLS site conforms to RG 4.7, Regulatory Position C.4, Revision 2.

2.1.3.5 *Post Combined License Activities*

There are no post COL activities related to this section.

2.1.3.6 *Conclusion*

The applicant initially estimated the commercial operation of WLS to be 2016 and revised it to 2024. Due to this change, the applicant has reviewed the impact and determined it is not significant. As discussed in NUREG-1793 and its supplements, the applicant provided an acceptable description of current and projected population densities in and around the site. The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to population distribution, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the application are documented in NUREG-1793 and its supplements.

The staff reviewed WLS COL 2.1-1 and, for the reasons given above, concludes that the population data meet the requirements of 10 CFR 50.34(a)(1), 10 CFR 52.79(a)(1), 10 CFR 100.20(a), 10 CFR 100.20(b), 10 CFR Part 100, and 10 CFR 100.3. This conclusion is based on the applicant having provided an acceptable description and safety assessment of the site, which includes present and projected population densities that are within the guidelines of RG 4.7, Regulatory Position C.4, and properly specified the LPZ and population center distance. In addition, the staff reviewed and confirmed, by comparison with independently obtained population data, the applicant's estimates of the present and projected populations surrounding the site, including transients. The applicant also calculated the radiological consequences of design-basis accidents (DBAs) at the outer boundary of the LPZ (NUREG-0800, Chapter 15) and has provided reasonable assurance that appropriate protective measures can be taken within the LPZ to protect the population in the event of a radiological emergency. The staff notes this addresses and resolves COL Information Item 2.1-1 regarding population distribution. The staff finds that the applicant has provided sufficient information to satisfy the requirements in 10 CFR Part 50, 10 CFR Part 52, and 10 CFR Part 100.

2.2 Nearby Industrial, Transportation, and Military Facilities

2.2.1 Locations and Routes

This section of the report discusses the application and the staff review of nearby industrial facilities, transportation routes, storage tanks, military and other facilities with potential to impact the safe operation of WLS.

2.2.1.1 *Introduction*

The description of locations and routes refers to potential external hazards or hazardous materials that are present or may reasonably be expected to be present during the projected lifetime of the proposed plant. The purpose is to evaluate the sufficiency of information concerning the presence and magnitude of potential external hazards so that the reviews and evaluations described in NUREG-0800, Sections 2.2.3, 3.5.1.5, and 3.5.1.6, can be performed. The review covers the following specific areas: (1) the locations of, and separation distances to, transportation facilities and routes including airports and airways, roadways, railways, pipelines, and navigable bodies of water; (2) the presence of military and industrial facilities, such as fixed manufacturing, processing, and storage facilities; and (3) any additional information relevant to meeting the requirements prescribed in the “Contents of Application,” sections of the applicable subparts to 10 CFR Part 52.

2.2.1.2 *Summary of Application*

WLS COL FSAR Revision 11, Section 2.2, incorporates by reference AP1000 DCD, Revision 19, Section 2.2. In addition, in WLS COL FSAR Section 2.2, the applicant provided the following:

Departures

- STD DEP 1.1-1

The applicant proposed the following AP1000 DCD Tier 2 standard departure. WLS COL FSAR Section 2.2.1, “Locations and Routes,” identifies instances where the WLS COL FSAR sections are renumbered to include content consistent with RG 1.206, as well as NUREG-0800. Here, AP1000 DCD, Section 2.2.1 is renumbered as Section 2.2.4.

In addition, this WLS COL FSAR section addresses the following COL-specific information identified in AP1000 DCD, Section 2.

AP1000 COL Information Item

This WLS COL FSAR section addresses the following COL-specific information identified in AP1000 DCD, Chapter 2.

- WLS COL 2.2-1

The applicant provided additional information in WLS COL 2.2-1 to resolve COL Information Item 2.2-1, which addresses information about industrial, military, and transportation facilities and routes to establish the presence and magnitude of potential external hazards.

Interface Items

- Interface Item 2.2

WLS COL FSAR Table 2.0-201, Sections 2.2.3.1.1 and 3.5 address Interface Item 2.2, “External missiles from manmade hazards and accidents.”

- Interface Item 2.3

WLS COL FSAR Section 2.2.3 addresses Interface Item 2.3, “Maximum loads from manmade hazards and accidents.”

2.2.1.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. In addition, the acceptance criteria associated with the relevant requirements of NRC regulations for the nearby industrial, transportation, and military facilities are given in NUREG-0800, Sections 2.2.1-2.2.2. The applicable regulatory requirements for identifying locations and routes are as follows:

- 10 CFR 52.79(a)(1)(iv), as it relates to the evaluation of sites, which requires the location and description of industrial, military, or transportation facilities and routes, and of 10 CFR 52.79(a)(1)(vi) as it relates to the compliance with 10 CFR Part 100.

- 10 CFR 100.20(b), as it relates to the requirement that the nature and proximity of man related hazards (e.g., airports, dams, transportation routes, military and chemical facilities) be evaluated to establish site parameters for use in determining whether plant design can accommodate commonly occurring hazards, and whether the risk of other hazards is very low.

- In addition, in accordance with 10 CFR Part 52, Appendix D, Section VIII, the applicant identified a Tier 2 departure, which does not require prior NRC approval. This departure is subject to the requirements in Section VIII, which are similar to the requirements in 10 CFR 50.59, “Changes, tests and experiments.”

The related acceptance criteria from NUREG-0800, Section 2.2.1-2.2.2 are as follows:

- Data in the FSAR adequately describes the locations and distances from the plant for nearby industrial, military, and transportation facilities and that such data are in agreement with data obtained from other sources, when available.
- Descriptions of the nature and extent of activities conducted at the site and in its vicinity, including the products and materials likely to be processed, stored, used, or transported, are adequate to permit identification of the possible hazards cited in NUREG-0800, Sections 2.2.1-2.2, Section III.
- Sufficient statistical data with respect to hazardous materials are provided to establish a basis for evaluating the potential hazards to the plant or plants considered at the site.

2.2.1.4 *Technical Evaluation*

The staff reviewed WLS COL FSAR Section 2.2 and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to nearby industrial, transportation, and military facilities. The results of the staff's evaluation of the information incorporated by reference in the application are documented in NUREG-1793 and its supplements.

Departures

- STD DEP 1.1-1

The applicant's evaluation in accordance with 10 CFR Part 52, Appendix D, Section VIII, Item B.5 determined that this departure did not require prior NRC approval. The staff finds that it is reasonable that the departure does not require prior NRC approval because the numbering system proposed by the applicant does not alter the information required to be provided.

AP1000 COL Information Item

- WLS COL 2.2-1

The staff reviewed WLS COL 2.2-1 related to information about industrial, military, and transportation facilities and routes to establish the presence and magnitude of potential external hazards included in WLS COL FSAR Section 2.2. AP1000 DCD, Section 2.2.1, COL information item 2.2-1 states:

> Combined License applicants referencing the AP1000 certified design will provide site-specific information related to the identification of potential hazards within the site vicinity, including an evaluation of potential accidents and verify that the frequency of site-specific potential hazards is consistent with the criteria outlined in Section 2.2. The site-specific information will provide a review of

aircraft hazards, information on nearby transportation routes, and information on potential industrial and military hazards.

The staff reviewed the WLS COL FSAR using the review procedures described in NUREG-0800, Section 2.2.1-2.2.2. This section of the report evaluates the applicant's identification of industrial, transportation, mining, and military installations in the WLS area. The evaluation of potential effects on the safe operation of the nuclear facility is described in Section 2.2.3 of this report.

Locations and Routes

The applicant identified and provided information regarding potential external hazard facilities and operations within an 8 km (5 mi) radius of the WLS site. Within the 8 km (5 mi) radius of the site, there are four major industrial facilities, a railroad, and four State and one Federal highway.

Dams

The Ninety-Nine Islands hydroelectric dam is on the Broad River adjacent to the WLS site boundary, approximately 1.8 km (1.1 mi) south of the WLS center point.

Quarrying or Mining Facilities

The staff verified that there are three permitted quarrying or mining facilities are located within the 8 km (5 mi) radius of the WLS site. The closest permitted mine is 1.6 km (1 mi) north of the site operated by Thomas Sand Company, called Blacksburg Plant, is a sand mine. Cunningham Brick Company operates a mine 5 km (3.2 mi) north of the site to mine mica. The Kings Creek Mine, operated by Industrial Materials, Inc., is 7.9 km (4.9 mi) northeast of the site and also mines mica. No explosives are used in these mining operations.

Storage Tanks

The applicant provided information from a database of registered storage tanks in South Carolina. The database includes all underground storage tanks for regulated substances greater than 416 liter (L) (110 gallon (gal)) capacity. Within an 8 km (5 mi) radius there are, registered aboveground storage tanks. At the WLS site, liquid hydrogen will be stored according to the AP1000 design, as well as other compressed gasses, including compressed hydrogen, liquid nitrogen, and liquid carbon dioxide.

Oil and Gas Pipelines

Nine major pipelines are located within 8 km (5 mi) of the WLS site. The information about whether these pipelines could, in the future, carry different and potentially more hazardous products is considered to be sensitive information. In addition to local major pipelines, there are branch distribution lines to local residential, commercial, and industrial facilities ranging from 15 to 2.5 cm (6 to 1 in.) diameter.

Military Facilities

There are no military facilities within 8 km (5 mi) of the WLS site.

Description of Railroads

The staff verified that no active railroads are located within the 8 km (5 mi) radius of the WLS site. Two railroad lines, an abandoned track and an active line, are located within 16 km (10 mi) of the WLS site.

Manufacturing and Storage of Hazardous Materials

The staff verified that no manufacturing facilities that use or store hazardous products are located within the 8 km (5 mi) radius of the WLS site (WLS COL FSAR Figure 2.2.2-201).

Description of Waterways

The WLS site is approximately 1.5 km west and 0.7 km south (4800 ft west and 2400 ft south) of the Broad River and 1.8 km (1.1 mi) north of the Ninety-Nine Islands hydroelectric station.

Upstream of WLS, the Broad River is shallow and generally unnavigable and south of the Ninety-Nine Islands hydroelectric station to the Pacolet River is considered “navigable waters” under the laws of South Carolina. There are public access points on the nearby Broad River. The access points are the Ninety-Nine Islands boat landing operated by Duke Energy and the Cherokee Landing, located across the river from the Ninety-Nine Islands boat landing. The intake structure for the WLS site is also located on the Broad River.

Description of Highways

The staff verified that the major highway located near the WLS site is U.S. 29, passing 7.4 km (4.6 mi) northwest, at its closest point. In addition, segments of South Carolina Routes 5, 97, 105 and 329 are located within 8 km (5 mi) of WLS at their closest point. The application included estimate average daily traffic counts.

Description of Railways

The staff verified that two railroad lines are located within 16 km (10 mi) of the WLS site. The Norfolk Southern Railroad Company (NSRC) owns a spur line that passes 7.5 km (4.7 mi) from the site and averages two trains per day (one round-trip) with the speed limit of 40 km/hr (25 mi/hr) and 16 km/hr (10 mi/hr) on curves, carrying freight only. The NSRC owns a major rail line located 8.8 km (5.5 mi) from WLS. This line averages 22 trains per day at an average speed of 80 km/hr (50 mi/hr). Future activities include a proposed Southeast High-Speed Rail Corridor running in the current rail right-of-way expected to reach a maximum speed of 180 km/hr (110 mi/hr). In accordance with RG 1.206, further analysis of the NSRC rail line is not required because it is outside of the 8 km (5 mi) radius of the WLS site. A railroad spur is expected to be constructed that will join the WLS site to the NSRC line. It will be owned and maintained by the applicant. This spur will be generally limited to shipments, and supporting activities, for the WLS site only.

Description of Airports

There are two commercial airports within 80 km (50 mi) of the WLS site. Greenville-Spartanburg International Airport is 66 km (41.3 mi) west and Charlotte Douglas

International Airport is 55 km (34.4 mi) northeast of WLS. The approach and departure paths of these commercial airports do not align with the WLS site. These airports within 80 km (50 mi) of the WLS site do not meet or exceed the acceptance criteria of NUREG-0800, Section 3.5.1.6 and RG 1.206, Part I, Subsections C.I.2.2.2.7 and C.I.3.5.6. There are no airports located within 16 km (10 mi) of WLS. At 24 km (14.7 mi) east of the site is York Airport. This airport is used exclusively by single-engine private aircraft of which there are 12 based at the field. However there is one heliport, with one helipad, located approximately 10 km (6 mi) north of the site. The staff verified that no active military facilities are within 8 km (5 mi) of the WLS site. The staff verified that no airports are within the 8 km (5 mi) radius of the WLS.

WLS COL FSAR Tables 2.2-205 and 2.2-207 describes the historical air traffic for passenger-aircraft-associated airports within the region. The applicant stated that neither of these airports has approach or departure paths aligned with WLS.

<u>Airways</u>

The applicant addressed and evaluated potential aircraft hazards following the approach and methods outlined in NUREG-0800, Section 3.5.1.6, "Aircraft Hazards," and determined an aircraft crash into the effective plant areas of the safety-related structures on the site met the acceptance criteria. Due to the close proximity of airways to the proposed WLS site, the applicant provided additional evaluation of hazards from air traffic in WLS COL FSAR Section 3.5.1.6. There are two low-altitude (below 5.5 km (18,000 ft)) Federal air routes within 24 km (15 mi) of the site. One air route is 6.4 km (4 mi) north of the site and the other is 16 km (10 mi) southwest of the site and is used by general aviation aircraft. Two high altitude (5.5 - 13.8 km (18,000 - 45,000 ft)) Federal air routes are located within 24 km (15 mi) of the site. One is 15 km (9 mi) southeast and the other is 20 km (12.5 mi) northeast of WLS. These air routes are primarily used by commercial air carriers, military, and high performance general aviation aircraft.

<u>Projections of Industrial Growth</u>

There are no commercial industrial parks within 8 km (5 mi) of the WLS site. However, there are two industrial companies within 8 km (5 mi); the Broad River Energy Center and Herbie's Famous Fireworks. There is no planned industrial growth within an 8 km (5 mi) area.

2.2.1.5 *Post Combined License Activities*

There are no post COL activities related to this section.

2.2.1.6 *Conclusion*

The staff reviewed the WLS COL application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to nearby industrial, transportation, military facilities, and other places of interest, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the application are documented in NUREG-1793 and its supplements.

As set forth above, the applicant has presented and substantiated information to establish an identification of potential hazards in the site vicinity. The staff reviewed WLS COL 2.2-1, and for the reasons given above, concludes that the applicant has provided information with respect to identification of potential hazards in accordance with the requirements of 10 CFR 52.79(a)(1)(iv) and 10 CFR 52.79(a)(1)(vi). The nature and extent of activities involving potentially hazardous materials that are conducted at nearby industrial, military, and transportation facilities have been evaluated to identify any such activities that have the potential for adversely affecting plant safety-related structures. Based on an evaluation of information in the WLS COL FSAR, as well as information that the staff independently obtained, the staff concluded that all potentially hazardous activities on site and in the vicinity of the plant have been identified. The hazards associated with these activities have been reviewed and are discussed in Sections 2.2.3, 3.5.1.5, and 3.5.1.6 of this report.

2.2.2 Descriptions

The staff's review of WLS COL FSAR Section 2.2.2, "Descriptions," is addressed in Section 2.2.1, "Locations and Routes," of this report.

2.2.3 Evaluation of Potential Accidents

2.2.3.1 *Introduction*

The evaluation of potential accidents considers the applicant's probability analyses of potential accidents involving hazardous materials or activities on site and in the vicinity of the proposed site to confirm that appropriate data and analytical models have been used. The review covers the following specific areas: (1) hazards associated with nearby industrial activities, such as manufacturing, processing, or storage facilities; (2) hazards associated with nearby military activities, such as military bases, training areas, or aircraft flights; and (3) hazards associated with nearby transportation routes (aircraft routes, highways, railways, navigable waters, and pipelines). Each hazard review area includes consideration of the following principal types of hazards: (1) toxic vapors or gases and their potential for incapacitating nuclear plant control room operators; (2) overpressure resulting from explosions or detonations involving materials such as munitions, industrial explosives, or explosive vapor clouds resulting from the atmospheric release of gases (such as propane and natural gas or any other gas) with a potential for ignition and explosion; (3) missile effects attributable to mechanical impacts, such as aircraft impacts, explosion debris, and impacts from waterborne items such as barges; and (4) thermal effects attributable to fires.

2.2.3.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 2.2, incorporates by reference AP1000 DCD, Revision 19, Section 2.2. In addition, in WLS COL FSAR Section 2.2, the applicant provided the following:

<u>*AP1000 COL Information Item*</u>

- WLS COL 2.2-1

The applicant provided additional information in WLS COL 2.2-1 to resolve COL Information Item 2.2-1, which addresses the provision of information about industrial, military, and transportation facilities and routes to establish the presence and magnitude of potential external hazards, including the following accident categories: explosions, flammable vapor clouds (delayed ignition), toxic chemicals, fires, and airplane crashes.

2.2.3.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. In addition, the acceptance criteria associated with the relevant requirements of NRC regulations for the evaluation of potential accidents are given in NUREG-0800, Section 2.2.3. The applicable regulatory requirements for evaluation of potential accidents are as follows:

- 10 CFR 52.79(a)(1)(iv), as it relates to the evaluation of sites, including the location and description of industrial, military, or transportation facilities and routes, and the requirements of 10 CFR 52.79(a)(1)(vi) as it relates to compliance with 10 CFR Part 100.

- 10 CFR 100.20(b), as it relates to the requirement that the nature and proximity of manmade related hazards (e.g., airports, dams, transportation routes, military and chemical facilities) be evaluated to establish site parameters for use in determining whether plant design can accommodate commonly occurring hazards, and whether the risk of other hazards is very low.

The related guidance and acceptance criteria from NUREG-0800, Section 2.2.3 are as follows:

- Event Probability: The identification of design-basis events (DBEs) resulting from the presence of hazardous materials or activities in the vicinity of the plant or plants of specified type is acceptable if all postulated types of accidents are included for which the expected rate of occurrence of potential exposures resulting in radiological dose in excess of the 10 CFR 50.34(a)(1) limits as it relates to the requirements of 10 CFR Part 100 is estimated to exceed the staff's objective of an order of magnitude of 10^{-7} per year.

- Design-Basis Events: The effects of DBEs have been adequately considered, in accordance with 10 CFR 100.20(b), if analyses of the effects of those accidents on the safety-related features of the plant or plants of specified type have been performed and measures have been taken (e.g., hardening, fire protection) to mitigate the consequences of such events.

In addition, the toxic gas evaluations should be consistent with appropriate sections from RG 1.78, "Evaluating the Habitability of a Nuclear Power Plant Control Room during a Postulated Hazardous Chemical Release," Revision 1.

2.2.3.4 *Technical Evaluation*

The staff reviewed WLS COL FSAR Section 2.2 and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the evaluation of potential accidents. The results of the staff's evaluation of the information incorporated by reference in the application are documented in NUREG-1793 and its supplements. The staff reviewed the information in the WLS COL FSAR:

AP1000 COL Information Item

- WLS COL 2.2-1

The staff reviewed WLS COL 2.2-1 related to information regarding industrial, military, and transportation facilities and routes to establish the presence and magnitude of potential external hazards, including the following accident categories: explosions, flammable vapor clouds (delayed ignition), toxic chemicals, fires, and airplane crashes included in WLS COL FSAR Section 2.2.3. COL information item in AP1000 DCD, Section 2.2 states:

> Combined License applicants referencing the AP1000 certified design will provide site-specific information related to the identification of potential hazards within the site vicinity, including an evaluation of potential accidents and verify that the frequency of site-specific potential hazards is consistent with the criteria outlined in Section 2.2. The site-specific information will provide a review of aircraft hazards information on nearby transportation routes, and information on potential industrial and military hazards.

Explosions

The applicant considered hazards involving potential explosions that could result in blast overpressure because of detonation of explosives, chemicals, liquid fuels, and gaseous fuels for facilities and activities either onsite or within the site vicinity of the proposed units. The applicant evaluated potential explosions from nearby highways, railways, or facilities using 0.07 bar (1 psi) overpressure as a criterion for adversely effecting plant operation or preventing safe shutdown of the plant. In accordance with RG 1.91, "Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants," peak positive incident overpressures below 0.07 bar (1 psi) are considered to cause no significant damage.

The applicant determined a minimum safe standoff distance of 0.8 km (0.52 mi) for truck transport using conservative assumptions and RG 1.91 methods. By comparison, the distance to the closest highway is 7 km (4.24 mi) from the nearest safety-related structure. The applicant also calculated the minimum safe standoff distance of 2.8 km (1.76 mi) for a railroad accident, using conservative assumptions and RG 1.91 methods. The staff performed independent calculations, which confirmed the applicant's results. Therefore, the staff concludes the applicant's assumptions and methods are adequate.

The nearest and largest nearby natural gas pipelines are owned by Colonial Pipeline and by Plantation Pipeline are located 5.2 km (3.24 mi) from the WLS site, at the closest point. The 1 m (40 in) diameter pipeline, buried 1-1.2 m, (3-4 ft) underground, was analyzed using conservative assumptions and RG 1.91 methods. The analysis concluded that the minimum safe standoff distance is 4.6 km (2.83 mi) and the distance to the closest point on the site boundary is 5.2 km (3.24 mi). In WLS COL FSAR Section 2.2.3.1.1.2, the applicant stated that unconfined vapor explosions of natural gas are not considered credible events. The applicant also stated that deflagration of a natural gas/air mixture is the limiting case, assuming that a mixture within the flammable limits is not present near the safety-related structures. In WLS COL FSAR Section 2.2.3.1.2, a delayed flammable cloud ignition is discounted on the basis of insufficient gas concentrations at the WLS site. In WLS COL FSAR Section 2.2.3.1.1.3, other nearby industrial facilities identified were reviewed for their minimum standoff distance from the site for explosive hazard with similar results to those discussed for pipelines. In WLS COL FSAR Section 2.2.3.1.1.4, the hazard from onsite chemicals was reviewed. Except as noted in the application, the hazardous chemicals listed by the WLS COL FSAR are bounded by the AP1000 design. The staff finds that this WLS COL FSAR discussion satisfies the requirement in Interface Item 2.2 to address external missiles from manmade hazards.

Toxic Chemicals

As previously noted, there is no major barge traffic within 8 km (5 mi) of the WLS site. The highway with heavy commercial traffic, nearest the WLS site, is U.S. Highway 29, passing about 7.2 km (4.5 mi) northwest of the site. South Carolina Highways 5, 97, 105, and 329 are located within 8 km (5 mi) of the site. Annual average traffic data for 2005 was used to estimate traffic volume for the hazards analysis. The result of the analysis was that the release frequencies are higher than the acceptance criteria of RG 1.78 (1E-6/yr for mobile sources). The additional analysis is discussed in WLS COL FSAR Section 2.2.3. After independently reviewing available information on the internet from local, State and Federal agencies, the staff concluded that the applicant's review is adequate.

Regarding the onsite storage of chemicals, the applicant stated that the chemicals stored on site are bounded by the standard chemicals identified in AP1000 DCD, Table 6.4-1. These chemicals were assessed by Westinghouse as part of the main control room habitability hazard analysis supporting the application. The Westinghouse analysis concluded that the chemicals listed in AP1000 DCD Table 6.4-1 did not present a hazard to the control room operators or to safety-related systems, structures, or components (SSCs). The applicant stated that the analysis was applicable because chemicals stored on site are bounded by the types, locations, and quantities of chemicals specified in the AP1000 DCD. Therefore, no further analysis is required. However, the staff independently reviewed the Westinghouse analysis and the site-specific analysis and concluded that the immediate danger to life and health (IDLH) is still exceeded outside the control room for carbon dioxide and hydrazine. The effect of these two gases on the habitability of the control room is evaluated in Section 6.4 of this report. The staff concurs with the applicant's conclusions on the other chemicals listed in AP1000 DCD, Table 6.4-201.

Fire, Smoke, and Heat Fluxes

The distances between any fires originating from facilities or transportation accidents and the WLS facility is at least 3.7 km (2.31 mi), and do not have the potential to affect the safe operation of WLS. The WLS main control room heating, ventilation, and air conditioning (HVAC) system is described in the application as follows: "continuously monitors the outside air using smoke monitors located at the outside air intake plenum and monitors the return air for smoke upstream of the supply air handling units (DCD Section 9.4.1.2.3.1). If a high concentration of smoke is detected in the outside air intake, an alarm is initiated in the main control room and the main control room/technical support center HVAC subsystem is manually realigned to the recirculation mode by closing the outside air and toilet exhaust duct isolation valves. Therefore, any potential heavy smoke problems at the main control room air intakes would not affect the WLS operators." The staff reviewed and verified the above information and concluded that the applicant's determination is adequate.

Collision with the Intake Structure

This section is not applicable, as the WLS intake structure is not located on a navigable portion of the Broad River waterway, collision with commercial traffic is not considered credible. Neither is collision with any of the site make-up ponds.

Liquid Spills

There is no safety-related equipment at the WLS intake structure. Liquid spill of petroleum or similar products or corrosive liquids would collect or disperse depending on its particular attributes. Spills that float on the river would not be drawn into the WLS water supply because of the location of the intake pipes. Spills that sink would be drawn into the intake structure and would be sent to Makeup Pond A and would likely be diluted in the process. Failure of the raw water system would not prevent essential functions of WLS safety-related systems. The staff reviewed and verified the information given above and concurs with the applicant's conclusion.

2.2.3.5 *Post Combined License Activities*

There are no post COL activities related to this section.

2.2.3.6 *Conclusion*

The staff reviewed the WLS COL application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to evaluation of potential accidents, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the application are documented in NUREG-1793 and its supplements.

As discussed above, the applicant presented and substantiated information to identify potential hazards in the site vicinity. The staff reviewed the information provided and, for the reasons given above, concludes that the applicant provided sufficient information with respect to identification of potential hazards in accordance with the requirements of 10 CFR 52.79(a)(1)(iv) and 10 CFR 52.79(a)(1)(vi). The nature and extent of activities involving potentially hazardous materials that are conducted at nearby industrial, military, and transportation facilities have been evaluated to identify any such activities that have the potential for adversely affecting plant safety-related structures. Based on an evaluation of information in the WLS COL FSAR as well as information that the staff independently evaluated, the staff concludes that potentially hazardous activities on site and in the vicinity of the plant have been identified. This addresses and resolves COL Information Item 2.2-1 and Interface Item 2.2. In addition, the staff finds that the applicant has provided sufficient information to satisfy the requirements of 10 CFR Part 50, 10 CFR Part 52, and 10 CFR Part 100.

2.3 Meteorology

To ensure that a nuclear power plant or plants can be designed, constructed, and operated on an applicant's proposed site in compliance with NRC regulations, the staff evaluated regional and local climatological information, including climate extremes and severe weather occurrences that may affect the design and siting of a nuclear plant. The staff reviewed information on the atmospheric dispersion characteristics of a nuclear power plant site to determine whether the radioactive effluents from postulated accidental releases, as well as routine operational releases, comply with NRC regulations. The staff prepared Sections 2.3.1 through 2.3.5 of this safety evaluation report (SER) in accordance with the review procedures described in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," using information presented in WLS COL FSAR Section 2.3 (which references the AP1000 DCD), responses to staff requests for additional information (RAIs), and generally available reference materials (as cited in applicable sections of NUREG-0800).

2.3.1 Regional Climatology

2.3.1.1 *Introduction*

WLS COL FSAR Section 2.3.1, "Regional Climatology," addresses averages and measured and probabilistic extremes of climatic conditions and regional meteorological phenomena that could affect the safe design and siting of the plant. This includes information describing the general climate of the region, seasonal and annual frequencies of severe weather phenomena, and other climatological conditions to be used for design and operating-basis considerations. This section of the report also addresses the supplemental information in WLS COL FSAR Section 2.3.6 related to regional climatology.

2.3.1.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 2.3, incorporates by reference AP1000 DCD, Revision 19, Section 2.3. In WLS COL FSAR Section 2.3.1, the applicant addressed the following:

<u>AP1000 COL Information Item</u>

- WLS COL 2.3-1

The applicant provided additional information in WLS COL 2.3-1 to address COL Information Item 2.3-1. WLS COL 2.3-1 addresses site-specific information related to regional climatology.

In addition, this WLS COL FSAR section addresses Interface Item 2.4 related to extreme meteorological conditions for the design of systems and components exposed to the environment, Interface Item 2.5 related to tornado and operating basis wind loadings, Interface

Item 2.7 related to snow, ice and rain loads, and Interface Item 2.8 related to ambient air temperatures.

2.3.1.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed NUREG-1793 and its supplements. The acceptance criteria and information needed to evaluate regional climatological and meteorological characteristics are based on meeting the relevant requirements of NRC regulations under 10 CFR Part 52 and 10 CFR Part 100. The staff considered the following regulatory requirements in reviewing the applicant's descriptions of the regional climatological and meteorological characteristics around the proposed WLS site.

- 10 CFR 52.79(a)(1)(iii), as it relates to identifying the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and time in which the historical data have been accumulated.

- 10 CFR Part 100, 10 CFR 100.20(c)(2), and 10 CFR 100.21(d) with respect to the consideration given to the regional meteorological characteristics of the site.

NUREG-0800, Section 2.3.1, recommends the following acceptance criteria be provided by a COL applicant:

- The description of the general climate of the region should be based on standard climatic summaries compiled by the National Oceanic and Atmospheric Administration (NOAA). In addition, consideration of the relationships between regional, synoptic scale atmospheric processes and local (site) meteorological conditions should be based on appropriate meteorological data.

- Data on severe weather phenomena should be based on standard meteorological records from nearby representative National Weather Service (NWS), military, or other stations recognized as standard installations that have long periods of data on record. The applicability of these data to represent site conditions during the expected period of reactor operation should be substantiated.

- Tornado parameters to be used in establishing pressure and tornado missile loadings on SSCs important to safety should be based on RG 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," Revision 1. Alternatively, an applicant may specify any tornado parameters that are appropriately justified, provided that a technical evaluation of site-specific data is conducted.

- The basic (straight-line) 100-year return period 3-second gust wind speed to be used in establishing wind loadings on plant structures should be based on appropriate standards, with suitable corrections for local conditions.

- The ultimate heat sink (UHS) meteorological data that would result in the maximum evaporation and drift loss of water and minimum water cooling should be based on long-period regional records that represent site conditions in accordance with RG 1.27, "Ultimate Heat Sink for Nuclear Power Plants," Revision 2. If applicable, the potential for water freezing in the UHS water storage facility should also be analyzed. (Note: Not applicable to a Passive Containment Cooling System design that does not use a cooling tower or cooling pond, such as the AP1000, where the UHS is the atmosphere.)

- The weight of the 100-year return period snowpack or snowfall and the weight of the 48-hour probable maximum winter precipitation (PMWP) are to be used in determining the weight of snow and ice on the roofs of safety-related structures. Consistent with the staff's branch position on winter precipitation loads, the winter precipitation loads to be included in the combination of normal live loads should be based on the weight of the 100-year snowpack or snowfall, whichever is greater, recorded at ground level. The weight of the 100-year return period snowpack or snowfall should be based on data recorded at nearby representative climatic stations or obtained from appropriate standards with suitable corrections for local conditions. The winter precipitation loads to be included in the combination of extreme live loads should be based on the weight of the 100-year snowpack or snowfall at ground level plus the weight of the 48-hour PMWP of which should be determined in accordance with reports published by NOAA's Hydrometeorological Design Studies Center.

- Ambient temperature and humidity statistics for use in establishing heat loads for the design of normal plant heat sink systems, post-accident containment heat removal systems, and plant heating, ventilating, and air conditioning systems should be derived from data recorded at nearby representative climatic stations or obtained from appropriate standards with suitable corrections for local conditions.

- High air pollution potential information should be based on U.S. Environmental Protection Agency (EPA) studies.

- All other meteorological and air quality conditions identified by the applicant as climate site characteristics or used as design and operating bases should be documented and substantiated.

Generally, the information should be presented and substantiated in accordance with acceptable practice and data as issued by NOAA, industry standards, and regulatory guides. The following regulatory guides and other related guidance documents were also taken into consideration in the staff's review of WLS COL FSAR Section 2.3.1 and Appendix 2CC (as applicable):

- RG 1.27, "Ultimate Heat Sink for Nuclear Power Plants," which provides criteria for selecting the UHS meteorological data that would result in the maximum evaporation and drift loss of water and minimum water cooling.

- RG 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," Revision 1, which includes criteria for selecting design-basis tornado parameters for locations within the contiguous United States (U.S.);

- RG 1.206, "Combined License Applications for Nuclear Power Plants," which summarizes the types of regional climatological and meteorological information, identified in NUREG-0800, Section 2.3.1, that an applicant should provide in FSAR Section 2.3.1 when describing the general climatic conditions of the site area and region, and characterizing the meteorological conditions against which the plant design and operating bases will be evaluated;

- RG 1.221, “Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants,” which provides new guidance for selecting the design-basis hurricane wind speed and hurricane-generated missiles to be addressed in permit, license, and design certification applications for new nuclear reactors proposed to be located in the contiguous U.S., and, where applicable, for comparison against corresponding design-basis tornado wind speeds and tornado missiles to determine the controlling severe weather phenomenon;

- Interim Staff Guidance (ISG) document DC/COL-ISG-7, “Interim Staff Guidance on Assessment of Normal and Extreme Winter Precipitation Loads on the Roofs of Seismic Category I Structures,” which was issued subsequent to the publication of NUREG-0800, Section 2.3.1 to clarify the staff’s position on identifying winter precipitation events as site characteristics and site parameters for determining normal and extreme winter precipitation loads on the roofs of Seismic Category I structures.

2.3.1.4 *Technical Evaluation*

The staff reviewed WLS COL FSAR Section 2.3.1 and checked the applicable site parameters in AP1000 DCD Tier 1 (Table 5.0-1) and Tier 2 (Table 2-1) to ensure that the combination of the DCD and the information in the COL application represent the complete scope of information relating to this review topic. The staff’s review confirmed that the information contained in the application and incorporated by reference (as related to site parameters) addresses the required information relating to regional climatology. The staff reviewed AP1000 DCD, Section 2.3.1 under Docket Number 52-006. The staff’s technical evaluation of the applicable site parameters incorporated by reference related to regional climatology will be documented in the staff’s separate SER on the DC application for the AP1000 design. In addition, the staff reviewed the information in the WLS COL FSAR corresponding to the following:

AP1000 COL Information Item

- WLS COL 2.3-1

The staff reviewed information related to regional climatological conditions included by the applicant under WLS COL FSAR Section 2.3.1 in response to COL 2.3-1. The specific text of this COL Information Item in AP1000 DCD Tier 2, Section 2.3.6.1 states:

> Combined License applicants referencing the AP1000 certified design will address site-specific information related to regional climatology.

The staff relied upon the review procedures presented in NUREG-0800, Section 2.3.1, to independently assess the technical sufficiency of the information presented by the applicant.

2.3.1.4.1 General Climate

The applicant provided descriptions of the general climate of both South Carolina (including the Piedmont region in which the proposed WLS site is located) and for North Carolina. The discussion for South Carolina is primarily based on a climate summary available from the State Climatology Office[2]. The following discussion briefly summarizes the information provided by the applicant in WLS COL FSAR Section 2.3.1.1 and the staff's evaluation of this information.

The State's proximity to the Atlantic Ocean, the proximity to the Appalachian Mountains, and local elevation all influence the local climate. As indicated above, the proposed WLS site is located in the Piedmont region of South Carolina. In RAI 446, Question 02.03.01-1, the staff requested that the applicant identify the National Climatic Data Center's (NCDC) State climatic division for the proposed WLS site or an alternative. In an October 10, 2008, response to RAI 446, Question 02.03.01-1 and in accordance with NUREG-0800, the applicant updated Revision 0 of WLS COL FSAR Section 2.3.1.1 to indicate the State climatic division in which the proposed WLS site is located (i.e., South Carolina Climate Division 2, Northwest[3]).

The climate of South Carolina is humid, subtropical, and characterized by a short cold season and a relatively long warm season. During the summer, a maritime tropical air mass, known as the Bermuda High (where anticyclonic, or clockwise, motion brings additional moisture into the state) dominates South Carolina's weather. The State's annual average temperature varies around 12 °Centigrade (C)) (mid-50 (Fahrenheit (F)) in the mountains to the low-15 °C (60s F) along the coast.

Precipitation in South Carolina is ample and well distributed with two maxima and two minima throughout the year. The maxima occur around March and August, and the minima occur around April and November. Precipitation varies across the State, averaging from about 101 - 203 cm (40 to 80 inches (in.)) annually[1].

The staff notes the following with respect to this information:

- the proposed WLS site is located about 8 km (5 mi) west of York County, SC, one of several counties referred to as South Carolina Climate Division 3 (or North Central)

- average temperatures are similar between South Carolina Climate Divisions 2 and 3 (monthly and annual means are only about 0.55 ° C (1 to 1.5 °F) cooler in Climate Division 2)

- annual average total precipitation is about 10 percent higher (by about 11 to 13 cm) (4.5 to 5 in.) in Climate Division 2 than in Climate Division 3; being similar during the

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[2] South Carolina State Climatology Office, "Climate," http://www.dnr.sc.gov/climate/sco/ClimateData/cli_sc_climate.php Accessed September 15, 2014
[3] National Climatic Data Center, "US Climate Divisions," http://www.ncdc.noaa.gov/monitoring-references/maps/us-climate-divisions.php Accessed September 15, 2014

summer months and generally about 20 percent higher in Climate Division 2 from autumn through early spring[4]

Measurable snowfall may occur from one to three times a winter in all areas of the State, except in the Low Country. The greatest snowfall measured at the nearby Ninety-Nine Islands cooperative observing station, approximately one mile northeast of the site, was 33 cm (13 in.) on January 7, 1988. Overall, however, the greatest observed 24-hour snowfall total in the State (60 cm) (24 in.)) was recorded at Rimini, SC in February of 1973[5]. The staff notes that the Rimini cooperative observing station is located in the south-central part of the state in the Midlands region (i.e., between the Piedmont and Coastal Plain) about 177 km (110 mi) southeast of the proposed WLS site.

The occurrence of sleet and freezing rain varies across the state, averaging from 0.75 to 3.75 events per year. The highest monthly frequency occurs in January and ranges from approximately 0.25 to 1.5 events across the state[1].

Hail-producing storms are infrequent events that can occur at any time of the year but are generally associated with thunderstorm activity during the spring. The incidence of such events is higher in the Midlands, Piedmont, and Foothills regions (averaging about 1 to 2 days annually), decreasing to less than one day per year (on average) in the Low Country[1].

Average surface wind speeds across the state for all months range between 10 to 16 km (6 to 10 mi/hr). The Appalachian Mountains exert a strong influence on the surface wind direction. Prevailing winds tend to have a southwesterly orientation in the spring and summer and a northeasterly orientation in the autumn and winter. However, the staff notes that the undulating topography that characterizes the Piedmont region, which includes the site vicinity, also affects local airflow, especially under low wind speed conditions (such as that which occurs diurnally during the overnight and early morning hours, particularly during the autumn and summer seasons)[1].

The applicant characterized the relative potential for air pollution (rather, for prolonged periods of poor atmospheric dispersion) based on the frequency of air stagnation conditions in the site region. These conditions typically occur during an extended summer season from about May to October, with a minimum during the month of July. In South Carolina, air stagnation conditions generally occur about 10 to 20 days per year.

An air stagnation case is an episode of poor dispersion that persists for at least 4 days (as defined in the applicant's reference material). The applicant suggested that in the eastern U.S., "six cases" occur during the spring, "14 cases" in the summer, and "11 cases" in the fall, based

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[4] National Climatic Data Center, "Climate at a Glance," http://www.ncdc.noaa.gov/cag/ Accessed October 31, 2014

[5] South Carolina State Climatology Office, "South Carolina Maximum 24-Hour Snow, 1890-2007," http://www.dnr.sc.gov/climate/sco/ClimateData/cli_table_24hr_max_snow.php Accessed September 17, 2014

on a regionally averaged mean annual cycle. Figure 5 of Wang and Angell[6], as cited by the applicant, shows that all mean annual stagnation case frequencies are less than one for each of the months through the period of May through October, with the highest frequency being in September (0.75 case per month). Since this information is not a direct input to any of the analyses under WLS COL FSAR Section 2.3, the staff did not evaluate this discussion further.

The applicant briefly described some of the effects that tropical cyclones (which include hurricanes and tropical storms) have on the weather and climate of South Carolina, primarily focusing on the coast but also noting that these events have a significant influence over other areas inland by enhancing rainfall during the summer and fall months. The applicant also provided a graphic plot of tropical cyclone tracks based on a 156-year period of record (POR), whose paths at one point crossed within a 120 km (75 mi) radius of the Greenville/Spartanburg International Airport. While the center of this radial area lies about 64 km (40 mi) west of the proposed WLS site, the plot nevertheless shows that tropical cyclone storm intensities decrease as they move inland from the coast (Hurricane Hugo, despite its diminishing strength, retained its hurricane status as it traversed this area).

Finally, the applicant summarized tornado occurrences on a statewide basis indicating that such events were observed, on average, about 11 times per year based on a 40-year POR. Most of these storms occur between February and September, with May and August being the peak months. The peak in May is attributed primarily to the passage of squall lines and cold fronts; the August peak is said to often be associated with the passage of tropical cyclones.

Consistent with NUREG-0800, Section 2.3.1, except as noted above, the applicant provided an acceptable description of the general climate of South Carolina based on a recent summary prepared by the State Climatology Office.

2.3.1.4.2 Regional Meteorological Conditions for Design and Operating Bases

2.3.1.4.2.1 *Hurricanes*

In WLS COL FSAR Section 2.3.1.2.1, Revision 0, the applicant provided data on historical hurricane events that have affected North and/or South Carolina (i.e., having made landfall in either one or both states) using NOAA Technical Memorandum NWS SR-206.[7] The staff was unable to identify the same number of hurricane events as presented by the applicant. Therefore, in RAI 446, Question 02.03.01-2, the staff requested that the applicant explain their data sources.

In an October 10, 2008, response RAI 446, Question 02.03.01-2, the applicant revised the original hurricane statistics by creating a composite dataset, which was based on data provided by the NOAA Coastal Services Center (CSC)[7] and in NOAA Technical Memorandum NWS

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[6] J. X. L. Wang and J. K. Angell, “Air Stagnation Climatology for the United States (1948-1998),” NOAA Air Resources Laboratory Atlas No. 1, Air Resources Laboratory, Environmental Research Laboratories, Office of Oceanic and Atmospheric Research, Silver Spring, MD, April 1999.
[7] NOAA Coastal Services Center, “Historical Hurricane Tracks,” http://coast.noaa.gov/hurricanes/# Accessed November 3, 2014.

SR-206.[8] The applicant stated that where there were inconsistencies noted between the two sources of information, the most conservative storm category was chosen. Table 2.3.1-1 of this report contains the applicant's (original and revised) event counts.

Table 2.3.1-1 Historical Hurricane Data for North and South Carolina

Storm Strength	Applicant	Composite Data
Category 5	0	0
Category 4	2	4
Category 3	12	11
Category 2	10	15
Category 1	20	20
Total	**44**	**50**

The staff considers the revised composite data counts to be a reasonable representation of overall hurricane frequency for the two states over the 107-year POR from 1899 through 2005, with slight differences in counts because of potential double-counting of storm entries made for the two adjacent states. Data reported by the applicant indicates that hurricanes have made landfall in these two states as early as July and as late as November; the greatest frequencies of these events occur during September and August. The most intense of these hurricanes (rated on the Saffir-Simpson scale as Category 4 storms (209 to 261 km/hr (130 to 156 mi/hr) sustained wind speed)) occurred during September and October.

WLS COL FSAR Section 2.3.1.2.1 correctly states that hurricanes lose strength as they move inland from the coast. The applicant's October 10, 2008, response to RAI 446, Question 02.03.01-2 also indicated that only two of these storms tracked within 65 statute miles of the proposed WLS site during the POR considered (i.e., Hurricane Able in 1952 and Hurricane Hugo in 1989). Current NOAA-CSC records checked by the staff show that the path of a third (unnamed) hurricane traversed within this radial distance during 1893. At their maximum intensity while in this area, Hurricane Able had diminished to a tropical storm, the unnamed storm noted above decreased to Category 1 level, and Hugo, although weakened, was still a Category 2 hurricane as it passed through and continued into western North Carolina.

Consistent with the intent of the recently issued RG 1.221 and in support of Section 3.3 of this report, the staff estimated the maximum hurricane wind speed in the site area based on observed data to help assess whether the proposed WLS site might reasonably be expected to

[8] NOAA Technical Memorandum NWS SR-206, "Atlantic Tropical Storms and Hurricanes Affecting the United States: 1899-1999," http://www.srh.noaa.gov/ssd/techmemo/sr206.htm Accessed November 3, 2014.

experience hurricane-related wind speeds in excess of design-basis tornado wind speeds. WLS COL FSAR Table 2.3-203 includes Saffir-Simpson wind speed ranges associated with each of the five storm categories. Based on the discussion above, the highest storm category experienced in the site area over at least a 113-year POR (i.e., going back to the unnamed 1893 hurricane identified by the staff – and likely closer to a 150-year POR given that NOAA-CSC records extend back to the 1850's) was designated as a Category 2 hurricane. Sustained winds from Category 2 storms, under the Saffir-Simpson scale, range from 155 to 177 km/hr (96 to 110 mi/hr).

Sustained wind speeds nominally represent the peak speed averaged over a one-minute period. Conversely, design-basis tornado wind speeds from RG 1.76 and hurricane-related wind speeds based on RG 1.221 are both expressed as 3-second gust speeds for evaluating structural wind pressure loadings and for the generation of wind-borne missiles. As such, these wind speeds are not directly comparable.

The American Society of Civil Engineers – Structural Engineering Institute (ASCE-SEI) Standard 7-10, "Minimum Design Loads for Buildings and Other Structures," Table C26.5-2 provides an approximate relationship between sustained and 3-second gust wind speeds. For Category 2 hurricanes, the corresponding 3-second gust wind speeds over land at 10 m (33 ft) above open ground range from 170 to 195 km/hr (106 to 121 mi/hr). In comparison, the information in RG 1.221 indicates a 258 km/hr (160 mi/hr) hurricane wind speed for the WLS site area, noting that this latter value corresponds to an exceedance probability of 1×10^{-7}. Section 2.3.1.4.2.2 of this report provides a comparison of these values with the maximum design-basis tornado wind speed.

2.3.1.4.2.2 *Tornadoes and Waterspouts*

The applicant used a 56-year period of tornado reports (1950 to 2005) from the NCDC[9] to determine the number of tornadoes recorded near the proposed WLS site. In WLS COL FSAR Section 2.3.1.2.2, the applicant stated that 118 tornadoes had been reported in the eight counties that contain and surround the site, including Cherokee, Spartanburg, Union, Chester, and York in South Carolina, and Cleveland, Gaston, and Mecklenburg in North Carolina. Using the same tornado database, the staff independently confirmed the tornado statistics, as presented in WLS COL FSAR Table 2.3-204.

Following the methods of H.C.S. Thom, as presented in WASH-1300[10], "Technical Basis for Interim Regional Tornado Criteria," and using data from the tornado reports referred to above, the applicant estimated the probability of a tornado striking any particular location in the eight-county area around the proposed WLS site, during any one year, as:

[9] National Climatic Data Center, "Storm Events Database," http://www.ncdc.noaa.gov/stormevents/ Accessed October 13, 2014.
[10] WASH-1300, "Technical Basis for Interim Regional Tornado Criteria," U.S. Atomic Energy Commission, Washington, DC, May 1974.

$$P = \frac{z_{avg} \bullet t_{avg}}{A}$$

where, for the area being considered:

P = mean probability of a tornado striking a point in any year

t_{avg} = mean number of tornadoes per year

z_{avg} = mean path area of a tornado (sq-mi)

A = total area being considered

or 2.00×10^{-4} per year. The equivalent recurrence interval based on this probability, calculated as (1 / P), was estimated to be once every 5,000 years.

Similarly, the applicant used the same methods to estimate the probability of a tornado striking any particular location within Cherokee County, SC (where the proposed WLS site is located), as 1.64×10^{-4} per year, or an equivalent recurrence interval of 6,108 years, slightly lower than for the larger area that includes the surrounding counties.

The staff attempted to verify the applicant's probabilistic estimate for the eight-county area using the same on-line tornado database from the NCDC (i.e., "Storm Events" for North Carolina and South Carolina, Tornado Event Summaries). However, the staff could not confirm the area considered, finding a smaller total land area using the current (i.e., 2010) version of the same U.S. Census Bureau database – that is, 11,190 sq-km (4,320 sq-mi) versus the 13,290 sq-km (5,131 sq-mi) area used by the applicant. The staff also noted that observations of tornado path length and path width (used to estimate the area of individual tornado events) were missing for nearly one-third of the 118 recorded events. Both of these inputs appear to have competing effects on the probability (recurrence interval) calculation.

In Revision 9 of the WLS COL FSAR, the applicant reconciled the area used in the probabilistic tornado strike estimate (i.e., 13,290 sq-km) (5,131 sq-mi)) with the counties from which tornado occurrences were selected. In resolving this issue, the applicant added tornado event observations reported for Polk and Rutherford Counties, in North Carolina, over the 56-year POR, one and six events, respectively. The additional land area for these two counties, based on 2010 U.S. Census Bureau data, accounts for the discrepancy noted by the staff in its initial evaluation of the area considered. In re-confirming the applicant's probabilistic estimate for the ten-county area around the proposed WLS site, the staff finds the applicant's estimated strike probability and recurrence interval conservative and reasonable.

The staff also considered the tornado strike probabilities and other characteristics calculated for the 1°, 2°, and 4° box areas of latitude and longitude (Latitude/Longitude (Lat/Lon)) in the continental U.S. as reported in NUREG/CR-4461, "Tornado Climatology of the Contiguous United States," Revision 2. Based on the Lat/Lon coordinates of the two units at the proposed WLS site (i.e., approximately 35.0° north / 81.5° west – see WLS COL FSAR Section 2.1.1) and the results for the 1° and 2° box areas in NUREG/CR-4461, Revision 2, that include the site, the staff concluded that the applicant included the correct tornado statistics in WLS COL FSAR

Section 2.3.1.2.2. Based on the statistical summaries for the 1° and 2° Lat/Lon box areas around the proposed WLS site, the staff notes that the site is located in an area of transition with decreasing tornado point strike probabilities and increasing recurrence intervals to the south and east. Using the 2° Lat/Lon box area, the staff considers the point strike probability of about 2.0×10^{-4} reasonable. The applicant followed the methods presented in NUREG/CR-4461 to determine the tornado strike probabilities. Therefore, the staff finds information presented by the applicant acceptable.

The applicant chose tornado site characteristics based on RG 1.76, Revision 1, which specifies design-basis tornado characteristics for three tornado intensity regions defined within the continental U.S. Each region's parameters represent a 10^{-7} per year probability of occurrence. The applicant stated, and the staff agrees, that the proposed WLS site is located in Tornado Intensity Region I. The applicant proposed the following tornado site characteristics, which are listed in WLS COL FSAR Table 2.0-201:

- Maximum Wind Speed 230 mi/hr (370 km/hr)
- Pressure Drop 1.2 pounds per square inch (psi) (8.2 kN/mm^2)

The maximum tornado wind speed above is comparable (conservatively so) to the 10^{-7} probability wind speeds presented in NUREG/CR-4461, Revision 2 for the adjacent 1° and 2° Lat/Lon box areas that include and that are located to the south of the proposed WLS site.

The AP1000 site parameter tornado wind speed value of 482 km/hr (300 mi/hr) conservatively bounds the applicant's site characteristic tornado wind speed of 370 km/hr (230 mi/hr). In addition, this maximum design-basis tornado wind speed is well above the derived 3-second gust for hurricane wind speed. The wind speeds could range from about 170 to 195 km/hr (106 to 121 mi/hr), based on both the highest intensity hurricane recorded in the WLS site area (i.e., a Category 2 storm) and the 258 km/hr (160 mi/hr) 3-second gust hurricane wind speed for the site area based on RG 1.221 (Section 2.3.1.4.2.1 of this report has a more detailed discussion). The staff finds that the applicant's tornado wind speed site characteristic value follows the guidance provided in RG 1.76, Revision 1, and is therefore acceptable.

Finally, the applicant stated that waterspouts can occur over seas, bays, and lakes, but that none are expected near the proposed WLS site because the only nearby body of water is the Broad River. The staff agrees that waterspouts are typically a coastal phenomenon and notes that such events, when they do occur, are typically less intense than tornadoes. Given that and the design-basis tornado characteristics for the site as indicated above, the staff does not consider this weather element (i.e., waterspouts) an issue at the proposed WLS site.

2.3.1.4.2.3 *Thunderstorms, Lightning, and Hail*

The applicant's discussions on thunderstorms, lightning frequency, and hail occurrences in WLS COL FSAR Subsections 2.3.1.2.3-2.3.1.2.5 are intended to provide a general understanding of the occurrence of these severe weather events in the site area but do not result in the generation of site characteristics for use as design or operating bases.

The applicant's initial characterization of thunderstorm activity that could potentially occur in the site area used information from the NCDC's online "Storm Events" database. However, the staff noted an inconsistency in thunderstorm reporting when using the "Storm Events" database, rather than Local Climatological Data (LCD)[11] station summaries, also available from the NCDC. Therefore in RAI 446, Question 02.03.01-4, the staff requested that the applicant clarify the data referenced.

In an October 10, 2008, response to RAI 446, Question 02.03.01-4, the applicant provided a revised thunderstorm frequency analysis based on the LCD summaries for the first-order NWS stations at Greenville/Spartanburg (Greer), SC (GSP) Airport and Charlotte, NC (CLT) covering the periods from 1963 through 2007 and 1948 through 2007, respectively. WLS COL FSAR Table 2.3-205 was updated by the applicant to report the monthly and annual average frequencies of thunderstorms for the two locations and the composite average for both stations. From this evaluation, the applicant stated that thunderstorms occur on average 41.6 days per year.

Consistent with NUREG-0800, Section 2.3.1, the applicant compiled information about thunderstorm frequencies in the site area based on readily available LCD summaries from the NCDC. The staff notes that these data represent average frequencies of thunderstorm days (i.e., days on which thunder is either heard or lightning is observed when environmental noise is high at a manned observation station, or, in more recent years, when lightning is detected by a sensor at an automated weather observing station). The staff also notes that the reported average frequencies appear to represent arithmetic rather than weighted averages despite the different periods of record between the GSP and CLT stations (i.e., 45 and 60 years, respectively). Nevertheless, the staff finds the applicant's updated information reasonable and acceptable.

The applicant indicated that about 14 - 26 lightning strikes to earth per year per square mile could be expected in the WLS site area and that this range likely represents upper bounds for the lighting strike density. The staff independently evaluated this estimated range and assessment based on:

- the average annual frequencies of thunderstorm days reported in the LCD summaries for Greenville/Spartanburg (Greer), SC airport and Charlotte, NC, and a method attributed to the Electric Power Research Institute (EPRI) (yielding an expected frequency of about 13 strokes to earth per square mile per year)

- a more recent 14-year flash density map from the National Lightning Detection Network (NLDN) operated by Vaisala, Inc. (indicating a range of about 9 to 12 strokes to earth per square mile per year)[12]

- a 1999 paper by G. Huffines and R.E. Orville, "Lightning Ground Flash Density and Thunderstorm Duration in the Continental United States": 1989-96 (indicating about

[11] National Climatic Data Center, "Local Climatological Data." http://www.ncdc.noaa.gov/IPS/lcd/lcd.html Accessed October 13, 2014.
[12] Vaisala's National Lightning Detection Network, "Cloud-to-Ground Lightning Incidence in the Continental US (1997-2007)," http://www.lightningsafety.noaa.gov/stats/08_Vaisala_NLDN_Poster.pdf Accessed November 3, 2014.

3 to 5 strokes to earth per square kilometer (8 to 13 strokes to earth per square mile) per year)[13]

The staff considers the applicant's estimate of the frequency of lightning strokes to earth in the site area acceptable and conservative.

In WLS COL FSAR Section 2.3.1.2.5, Revision 0, the applicant's initial analysis of the frequency of hail events utilized information from the NCDC's online "Storm Events" database. The staff noted, as it did above with respect to thunderstorm activity, that use of the "Storm Events" database reflected some inconsistencies; in this case, that the frequency of recorded hail events appeared to increase markedly and become more consistent after 1984. Therefore, in RAI 446, Question 02.03.01-5, the staff requested that the applicant clarify the using data reported over the POR from 1950 through 2005.

In an October 10, 2008, response to RAI 446, Question 02.03.01-5, the applicant indicated that because there is no NCDC Local Climatic Data to be used in lieu of the hail event data from the "Storm Events" database, a revised analysis would be limited to the years from 1995 through 2006. WLS COL FSAR Table 2.3-206, Revision 0 was updated based on the revised POR. The applicant indicated that 432 hailstorms occurred in an eight-county region around the site over the period from January 1, 1995, through May 31, 2006, with the largest hail size reported at 7 cm (2.75 in.) in diameter. Cherokee County received approximately 10 percent of the reported hailstorms on an annual basis, averaging about 3.5 events per year. The staff performed an independent verification using information from the NCDC's "Climate Atlas of the United States" and obtained similar results.

With regard to the annual percent frequencies of occurrence by county, as listed in WLS COL FSAR Table 2.3-206, the staff notes that hail events are point observations, where aerial distribution may be biased by the population distribution depending on the area being considered, as recognized by the NCDC. This appears to be the case in the WLS site region, which reflects higher percentages reported for Spartanburg County in South Carolina and Mecklenburg County in North Carolina. Among the eight considered, these two counties have the highest populations and population densities based on U.S. Census Bureau data for 2010.

Although the database was limited to a POR of about 11.5 years (based on the applicant's response to RAI 446, Question 02.03.01-5), the frequency of hailstorm events could still be examined using other means of available data (some being recorded as early as the mid to late 1950s). The staff queried the NWS' Storm Prediction Center's severe weather database (Online Severe Plot 3.0)[14] for the eight-county area and determined that several events had occurred in the site region with hailstones larger than the maximum reported by the applicant (i.e., 7 cm (2.75 in.)). The largest hailstone reported in the area was 11.5 cm (4.5 in.) in diameter (a large softball is 9.7 cm (3.8 in.) in diameter) in Gaston County, NC, on April 3, 1974.

[13] G. R. Huffines and R. E. Orville, "Lightning Ground Flash Density and Thunderstorm Duration in the Continental United States: 1989-96," Cooperative Institute for Applied Meteorological Studies, Department of Meteorology, Texas A&M University, College Station, TX, August 25, 1998.

[14] Storm Prediction Center, "SPC National Severe Weather Database Browser," http://www.spc.noaa.gov/climo/online/sp3/plot.php Accessed November 7, 2014.

Given that these characteristics are not used for design or operating bases of safety-related SSCs, the staff finds the applicant's analysis acceptable.

2.3.1.4.2.4 *Regional Air Quality*

The applicant's discussion on regional air quality in WLS COL FSAR Section 2.3.1.2.6 intends to provide a general understanding of the air quality conditions in the site region and area but does not result in the generation of site characteristics for use as a design basis.

Consistent with 40 CFR 81.106, "Greenville-Spartanburg Intrastate Air Quality Control Region," the applicant correctly indicated that Cherokee County, SC, where the proposed WLS site is located, is included in the Greenville-Spartanburg Intrastate Air Quality Control Region (AQCR). National Ambient Air Quality Standards (NAAQS) have been issued by the EPA under 40 CFR Part 50 for the six principal air pollutants: carbon monoxide, nitrogen dioxide, ozone, particulate matter less than 10 microns and less than 2.5 microns in aerodynamic diameter (or PM-10 and PM-2.5), sulfur dioxide, and lead. Under 40 CFR 81.341, Cherokee County is designated as being in attainment (i.e., currently meets) or as unclassifiable/attainment (i.e., meeting the standard or expected to be meeting the standard despite a lack of monitoring data) for all criteria pollutants. The same is true for all other counties in South Carolina, except as indicated below for ozone.

The discussion in WLS COL FSAR Section 2.3.1.2.6 refers to a "newly promulgated" 8-hour ozone standard of 0.08 parts per million (ppm) at "62 *Federal Register* 36," dated July 18, 1997 (the staff notes that the correct regulatory citation is 62 FR 38894). Shortly after the initial submittal of the application in December 2007, the EPA revised that standard on March 27, 2008 (i.e., at 73 FR 16511) under new section 40 CFR 50.15, to be 0.075 ppm expressed to three decimal places.

Further, the listing of non-attainment counties/areas in South Carolina for the 8-hour ozone standard has remained unchanged in WLS COL FSAR Section 2.3.1.2.6 since the original application submittal. However, the staff notes that only a portion of York County, SC, is currently designated as being in marginal non-attainment for the 8-hour ozone standard (2008) per EPA's "Green Book"[15] (as of July 2, 2014). The staff agrees with the stated attainment and non-attainment statuses described for the surrounding counties in North Carolina.

As one means of characterizing the potential for restrictive atmospheric dispersion conditions, the applicant estimated average ventilation rates for the WLS site region based on a 7-year dataset of upper air observations (1984 to 1987 and 1989 to 1991) taken at the NWS station at Greensboro-High Point, NC. The data were obtained from the EPA Support Center for Regulatory Atmospheric Modeling (SCRAM) archive.[16] That NWS observing station is located in north-central North Carolina about 185 km (115 mi) to the northeast of the proposed WLS

[15] US Environmental Protection Agency, "The Green Book Nonattainment Areas for Criteria Pollutants" http://www.epa.gov/oar/oaqps/greenbk/ As of July 2, 2014.
[16] http://www.epa.gov/scram001/* Technology Transfer Network Support Center for Regulatory Atmospheric Modeling" U.S. EPA.

site. The elevation of the observing station is about 274 m (900 ft) above mean sea level (MSL) compared to an elevation of about 180 m (590 ft) above MSL for the proposed WLS site.

The ventilation rate was determined by the applicant as the product of the mixing height and wind speed within the mixing layer. Higher ventilation rates typically correspond to greater atmospheric dispersion potential, and vice versa. The applicant presented mean morning, afternoon, and daily ventilation rates in WLS COL FSAR Table 2.3-207 and discussed time periods during which relatively more or less atmospheric dispersion is expected to occur. However, the discussion of ventilation rates, as it relates to higher and lower air pollution potential, is somewhat redundant and disjointed between WLS COL FSAR Sections 2.3.1.2.6 and 2.3.2.4.1. The composite information is evaluated below.

In WLS COL FSAR Section 2.3.1.2.6, the applicant indicated that the “highest daily air pollution potentials” (or more restrictive dispersion conditions) exist in the morning from June through October, owing to lower ventilation rates. Conversely, the "lowest daily air pollution potentials” (or less restrictive dispersion conditions) occur from December through March because of relatively higher morning ventilation rates. In addition, WLS COL FSAR Section 2.3.2.4.1 states that highest daily air pollution potentials exist during the afternoon from August through January when ventilation rates are lower and that lowest air pollution potentials occur in the spring because of relatively high mean ventilation rates.

The staff performed an independent analysis using data derived from the online Ventilation Climate Information System (VCIS), one of the Wildland Fire Air Quality Tools developed through the U.S. Forest Service as part of the Wildland Fire Decision Support System. Similar results were found in terms of when lower and higher mean morning ventilation rates might be expected to occur—that is, June through October, and January through April, respectively. Likewise, lower mean afternoon ventilation rates (i.e., more restrictive dispersion) based on VCIS are estimated to occur from about September to February, with higher mean afternoon ventilation rates (i.e., less restrictive dispersion) from about April to July.

The method used by the applicant to estimate ventilation rates is similar to that employed by Holzworth (Reference 24 in NUREG-0800, Section 2.3.1). In both cases, the wind speed component of the calculation accounts for winds throughout the entire mixing layer, which can extend a few thousand meters in the vertical. Comparatively, the VCIS methods uses surface-level wind speeds that are usually lower because wind speed generally increases with height. The staff considers surface-level winds to be more representative of the dispersion conditions that potential radiological releases from an AP1000 unit would experience (i.e., conditions at or near ground level). Consequently, the staff’s results suggest that the ventilation rates reported by the applicant could be consistently overstated throughout the year: in the morning by about a factor of two (for some months even more); in the afternoon; and similarly on a mean daily basis, about 1.5 to 2.5 times higher.

In addition, the VCIS method adjusts (models) its calculations for the location of interest and accounts for topographic conditions. More importantly, the VCIS database covers a longer, more climatologically representative, period of record—about 30 years—compared to the 7-year POR used by the applicant. All of these factors likely contribute to the slight differences observed between the two approaches in the months associated with relatively more or less restrictive atmospheric dispersion conditions. At the time of the initial WLS COL FSAR submittal, the VCIS database was available through the Bureau of Land Management. After contacting the Bureau of Land Management, the staff determined that this database appears to no longer be supported or available for use.

As another indicator of the potential for restrictive dispersion conditions at the WLS site, in WLS COL FSAR Section 2.3.1.2.6, the applicant described the frequency of air stagnation conditions in the site region. Based on a 51-year POR (1948 to 1998) as reported by Wang and Angell, the applicant stated that South Carolina averages four air stagnation cases per year, with a mean duration of 5 days for each case, and 20 air stagnation days per year. The staff finds this description reasonable and acceptable and slightly conservative with respect to frequency of occurrence.

The applicant also indicated that air stagnation days occur more frequently from July through October and less often from November through March. This monthly variation is generally in agreement with the applicant's analysis and the staff's evaluation of morning ventilation rates (above); less so with respect to the monthly variation of afternoon and mean daily ventilation rates. Given that no site characteristics relate to this information, nor is this information used as input for any of the applicant's atmospheric dispersion modeling, the staff considers the applicant's discussion of these conditions in the WLS site region reasonable and acceptable.

2.3.1.4.2.5 *Severe Winter Storm Events*

In WLS COL FSAR Section 2.3.1.2.7, the applicant addressed severe winter storms in the site region from two perspectives. First, the introduction describes the frequency and other characteristics of such storms to provide a general understanding of the phenomena in the site region. As a group, these events may include snow, sleet, or freezing rain (the latter term usually referred to as ice storms in climatological records). However, this information does not result in the generation of site characteristics for use as design-or-operating-basis conditions or for comparison to corresponding site parameter values.

Second, specific characteristics of certain events that are considered in evaluating the acceptability of structural (roof) loads expected for the proposed WLS site are addressed in WLS COL FSAR Sections 2.3.1.2.7.1 through 2.3.1.2.7.3. This information generates a site characteristic value that is compared against a corresponding Tier 1 and Tier 2 site parameter.

2.3.1.4.2.5.1 General Conditions

In WLS COL FSAR Section 2.3.1.2.7, the applicant summarized 283 ice storms (i.e., snow, sleet, and/or freezing rain), by date, which were reported in the NCDC's online "Storm Events" database between 1993 and 2005. These data were obtained for the eight counties that include and surround the proposed WLS site (i.e., Cherokee, Spartanburg, Union, Chester, and York in South Carolina, and Cleveland, Gaston, and Mecklenburg in North Carolina). Based on this information, which appears in WLS COL FSAR Table 2.3-208, the applicant estimated that 22 such events occur annually in this region. The staff notes that this value simply represents the average number of individual dates with an event reported at any location across all eight counties during this 13-year POR. The applicant also stated that Cherokee County averages 3.6 such events per year.

Using the same database from this eight-county region, the staff, through analysis of the events by county in total each year, determined that a winter storm event could last for one or more days (as some of the "Storm Data" entries indicate) and impact multiple counties included in the region of analysis during the same time period. The staff's independent evaluation shows that winter storms occur, on average, about 2 to 3 days per year, with Cherokee and Spartanburg Counties, in South Carolina, and Cleveland County, in North Carolina, at the upper end of that range. As many as eight events, and as few as none, have occurred during any given year over the selected 13-year POR when looking at a particular county included in the eight county region. On average, the data show that the region around the WLS site might experience winter storms about four times per year. The staff considers the frequency of winter storm events based on the average of regional individual or multi-day occurrences to be more representative of conditions that might possibly occur at the proposed WLS site.

In WLS COL FSAR Section 2.3.1.2.7, the applicant also stated that the equivalent ice thickness because of freezing rain with concurrent 3-second gust speeds for a 100-year mean recurrence interval is 1.9 cm (0.75 in.) for north-central South Carolina. The 3-second gust speed value(s) was not indicated. The applicant based this ice thickness value on information in the American Lifelines Alliance's report titled "Extreme Ice Thicknesses from Freezing Rain" (September 2004). In its review of the cited reference, the staff determined that a 1.9 cm (0.75 in.) ice thickness is also associated with a 45 km/hr (30 mi/hr) concurrent 3-second gust wind speed and a 50-year mean recurrence interval. The reference also provides a conversion factor of 1.25 for scaling 50-year return period values for radial ice thicknesses in the range of 1.3 cm (0.5 in.) and greater to a 100-year return period. Expecting the 100-year mean recurrence period value to generally be higher than the 50-year value, in RAI 446, Question 02.03.01-6, the staff requested that the applicant verify the accuracy of the 100-year value.

In an October 10, 2008, response to RAI 446, Question 02.03.01-6, the applicant verified that the equivalent radial ice thicknesses for both the 50-year and 100-year mean recurrence intervals were the same for the WLS site location (i.e., 1.9 cm (0.75 in.)) based on the respective data plots in the cited reference. The RAI response also confirmed that the concurrent 3-second gust wind speed associated with these values was 48 km/hr (30 mi/hr) but was silent with respect to the scaling factor. The applicant concluded that no conversion factor was necessary to estimate a 100-year mean recurrence interval equivalent ice thickness value from the 50-year value.

Based on review of reports in the NCDC on-line "Storm Events" database covering the period back to 1996, the staff accepts the conclusion provided by the applicant. These event summaries suggest that radial ice thickness values because of freezing rain in the range of 1.9 cm (0.75 in.), up to about 2.5 cm (1 in.), could possibly occur in the WLS site region.

As stated previously, this general information on severe winter storm events does not generate site characteristics used as design or operating-basis conditions nor was it made to be compared against a corresponding AP1000 site parameter value. Nevertheless, with the clarifications noted above from the staff's review, these descriptions are reasonable indicators of severe winter storm events in the WLS site region and their potential effects.

2.3.1.4.2.5.2 Design-Basis Conditions

As discussed in the preceding paragraphs, snow and ice events do occur in the WLS region of South Carolina. The staff issued DC/COL-ISG-007, which clarifies the staff's position for identifying winter precipitation events as site characteristics and site parameters for determining normal and extreme winter precipitation loads on the roofs of Seismic Category I structures. The ISG revises the previously issued staff guidance as discussed in NUREG-0800, Section 2.3.1.

The ISG states that normal and extreme winter precipitation events should be identified in NUREG-0800, Section 2.3.1, as COL site characteristics for use in NUREG-0800, Section 3.8.4, in determining the normal and extreme winter precipitation loads on the roofs of Seismic Category I structures. The normal winter precipitation roof load is a function of the normal winter precipitation event; whereas, the extreme winter precipitation roof loads are based on the weight of the antecedent snowpack resulting from the normal winter precipitation event plus the larger resultant weight from either: (1) the extreme frozen winter precipitation event; or (2) the extreme liquid winter precipitation event. The extreme frozen winter precipitation event is assumed to accumulate on the roof on top of the antecedent normal winter precipitation event; whereas, the extreme liquid winter precipitation event may or may not accumulate on the roof, depending on the geometry of the roof and the type of drainage provided. The ISG further states:

- The normal winter precipitation event should be the highest ground-level weight (in pounds per square foot (lb/ft^2)) among: (1) the 100-year return period snowpack; (2) the historical maximum snowpack; (3) the 100-year return period two-day snowfall event; or (4) the historical maximum two-day snowfall event in the site region.

- The extreme frozen winter precipitation event should be the higher ground-level weight (in lb/ft^2) between: (1) the 100-year return period two-day snowfall event; and (2) the historical maximum two-day snowfall event in the site region.

- The extreme liquid winter precipitation event is defined as the theoretically greatest depth of precipitation for a 48-hour period that is physically possible over a 25.9-sq-km (10-sq- mi)) area at a particular geographical location during those months with the historically highest snowpack.

During the review, the staff identified slightly higher extreme snowfall statistics at another nearby cooperative observing station, Gaffney 6E, also located in Cherokee County and within 16 km (10 mi) of the proposed WLS site. These values included a 100-year return period snowfall amount of 41 cm (16.3 in.), a maximum observed snowfall of 43 cm (17.0 in.), and a maximum observed snow depth of the same amount. Therefore, in RAI 446, Question 02.03.01-8, the staff requested that the applicant justify why the WLS COL FSAR did not consider the more conservative snowfall data source.

In an October 10, 2008, response to RAI 446, Question 02.03.01-8, the applicant reiterated a portion of DC/COL-ISG-7 by restating the four parameters to be considered in establishing what that guidance refers to as the "Normal Winter Precipitation Event." These parameters include the highest among: the 100-year return period snowpack, the historical maximum snowpack, the

100-year return period snowfall event, or the historical maximum snowfall event in the site region. The applicant also stated that the 100-year return period snowpack from Ninety-Nine Islands is the maximum of these values and that the more conservative Gaffney 6E data would "be used in the PMWP evaluation." For clarification, the staff notes that the information discussed above pertains to determining the normal winter precipitation event, not the PMWP (i.e., probable maximum winter precipitation). The 48-hour PMWP component, discussed under the heading that follows, in combination with the controlling normal winter precipitation event, is used to estimate the extreme winter precipitation event.

Beginning with WLS COL FSAR, Revision 1, Section 2.3.1.2.7.1, the applicant incorporated the normal winter precipitation event value based on the Gaffney 6E data (i.e., 43 cm (17.0 in.)) and estimated the corresponding ground-level weight as 17.7 lb/ft^2 for the proposed WLS site, using the conversion factors indicated above. Therefore, with the clarifications and corrections noted above from the staff's review, the staff finds the applicant's estimate acceptable and considers RAI 446, Question 02.03.01-8 resolved.

<u>48-hour Probable Maximum Winter Precipitation</u>

As indicated previously and in DC/COL-ISG-7, the 48-hour PMWP is the second of two components used to estimate the extreme winter precipitation event applicable to a given site, which, in turn, provides input for the determination of the extreme winter precipitation live roof load calculation.

The introductory discussion of severe winter storm events discussed in WLS COL FSAR Section 2.3.1.2.7, Revision 0, reported a 48-hour PMWP value of 8.99 cm (3.54 in.) for the proposed WLS site. This value was based on data from the Greenville/Spartanburg (Greer) NWS station over a 9-year POR from 1997 to 2005. The applicant did not discuss the methods for its determination. Generally, the staff does not consider a 9-year period of record to be climatologically representative of long-term conditions. Therefore, in RAI 446, Question 02.03.01-7, the staff requested that the applicant justify the reported 48-hour PMWP value of 8.99 cm (3.54 in.) in light of 24-hour winter precipitation totals recorded at the nearby Ninety-Nine Islands and Gaffney 6E cooperative observing stations (i.e., 8.56 and 9.55 cm (3.37 and 3.76 in.)) and at the Greenville/Spartanburg (Greer) NWS station (i.e., 9.65 cm (3.80 in.)). In addition, the staff requested that the applicant either clarify the terminology used in WLS COL FSAR Section 2.3.1.2.7 to describe this 8.99 cm (3.54 in.) estimate or to explain an 8.99 cm (3.54 in.) 48-hour PMWP value, also discussed in WLS COL FSAR Section 2.3.1.2.7.2 but attributed to the Ninety-Nine Islands cooperative observing station. The latter section also identified another 48-hour PMWP value of 77.5 cm (30.5 in.) for the site area based on NOAA HMR 53[17].

In an October 10, 2008, response to RAI 446, Question 02.03.01-7, the applicant stated that the data from the Ninety-Nine Islands station is most representative of site conditions because of its location on the Broad River and close proximity to the site. However, the applicant also stated that the data from this location (referring to the Ninety-Nine Islands observing station) does not

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[17] NOAA Hydrometeorological Report No. 53 – Seasonal Variation of 10-Square Mile Probable Maximum Precipitation Estimates, United States East of the 105th Meridian, NUREG/CR-1486.

provide a bounding value for design conditions, while also acknowledging that the PMWP value is established by NOAA's HMR 53 report as 77.5 cm (30.5 in.) as stated in WLS COL FSAR Section 2.3.1.2.7.2. Subsequently, in WLS COL FSAR Revision 1, the applicant reconciled the conflicting text in WLS COL FSAR Sections 2.3.1.2.7 and 2.3.1.2.7.2 by:

- in WLS COL FSAR Section 2.3.1.2.7, deleting the maximum one-day rainfall total of 18.2 cm (7.16 in.) for the Ninety-Nine Islands cooperative observing station

- correcting the descriptor for the 48-hour rainfall total from the Greenville/Spartanburg NWS station in WLS COL FSAR Section 2.3.1.2.7, now referring to that measurement as a maximum recorded winter precipitation value rather than a 48-hour PMWP value

- in WLS COL FSAR Section 2.3.1.2.7.2, deleting the 8.99 cm (3.54 in.) 48-hour PMWP value that had been attributed to the Ninety-Nine Islands cooperative observing station

The staff agrees with the applicant's use of NOAA HMR No. 53, consistent with the guidance in DC/COL-ISG-7 and NUREG-0800, Section 2.3.1, for characterizing the 48-hour PMWP for the proposed WLS site. Furthermore, the staff's independent analysis indicates that the applicant's 48-hour PMWP estimate of 77.5 cm (30.5 in.) is conservatively higher than the staff's evaluation, which the staff finds acceptable. With the clarifications noted above, the staff considers RAI 446, Question 02.03.01-7 resolved.

Regarding the applicant's characterization of design-basis conditions associated with the 48-hour probable maximum winter precipitation, in RAI 446, Question 02.03.01-9, the staff requested that the applicant describe the additional weight contributed by the 48-hour PMWP if the central hole on the roof of the passive containment cooling system (PCS) tank is blocked by a pre-existing 100-year snowpack. The staff also requested that the applicant estimate the additional weight if at least part of the 48-hour PMWP falls as frozen precipitation.

In an October 10, 2008, response to RAI 446, Question 02.03.01-9, the applicant first reiterated a portion of the discussion from WLS COL FSAR Section 2.3.1.2.7.2 concluding that water and snowmelt build-up on the roofs of the Nuclear Island are negligible because the roofs have no lips (i.e., parapets) around their edges. Furthermore, the shield building roof is sloped with no lips around its edges that would allow water build-up. The PCS tank (set atop the Shield Building) is also flat with no lip; therefore, water would not build up. The PCS tank also includes a central hole that would allow water to drain down in between the shield wall and the steel containment vessel, further reducing accumulation of water on the PCS tank roof area.

The applicant also summarized a response to an AP1000 design certification-related RAI, which discussed design features, as well as physical conditions, during normal operation and accident scenarios to protect against ice blockage of the chimney area. Provisions include the use of non-safety-related electrical heaters for air inlet structures to keep these structures free of ice and snow build-up; heat transferred from the reactor containment would help keep the chimney gratings free of ice build-up because of the flow of heated air upward through the gratings. The staff accepts, as reasonable, the preceding explanations as justification that the central PCS tank hole would not be blocked by a pre-existing 100-year snowpack.

The applicant's response to RAI 446, Question 02.03.01-9 addresses an additional roof loading because of rain-on-snow based on ASCE Standard 7-05, "Minimum Design Loads for Buildings and Other Structures." ASCE/SEI 7-05 (and its successor, ASCE/SEI 7-10) indicates that if the ground snow load is less than 20 lb/ft^2, an additional rain-on-snow surcharge of 5 lb/ft^2 is to be applied to low-sloped roofs. The previously stated ground snow load associated with the normal winter precipitation event for the WLS site region (i.e., 17.7 lb/ft^2) meets this criterion. Therefore, the applicant indicated that the 5 lb/ft^2 rain-on-snow surcharge applied. Given the shedding characteristics for rain and snowmelt from the roofs of Seismic Category I structures in the AP1000 design, and the staff's guidance in DC/COL-ISG-7, this rain-on-snow surcharge value corresponds to the extreme liquid winter precipitation event.

The ISG recommends characterization of the extreme frozen winter precipitation event as the higher ground-level weight of either the 100-year return period snowfall event or the historical maximum snowfall event in the site region. As identified earlier, the ground-level weight of the historical maximum snowfall event is 17.7 lb/ft^2 for the WLS site and as such, represents the extreme frozen winter precipitation event (as well as the controlling normal winter precipitation event). This value is the greater of the extreme frozen and extreme liquid winter precipitation events (i.e., 17.7 lb/ft^2 versus 7.2 lb/ft^2) and, therefore, represents the controlling ground-level weight of the extreme winter precipitation event.

Based on the additional information provided by the applicant regarding roof design and the application of the ASCE guidance, the staff finds the applicant's evaluation of the estimated weight of the 48-hr PMWP acceptable.

<u>Weight of Snow and Ice on Safety-Related Structures</u>

As discussed above, the applicant's revised site characteristic value for the weight of winter precipitation on safety-related structures equals 17.7 lb/ft^2. WLS COL FSAR Table 2.0-201 presents a comparison between the AP1000 Tier 1 site parameter and corresponding WLS site characteristic values for snow load. The staff notes that, as discussed above, the applicant did not include an additional rain-on-snow surcharge of 5 lb/ft^2 called for in ASCE 7/05. The staff recognizes that because of the large difference between the site characteristic snowload and the AP1000 DCD site parameter, this surcharge would not change the conclusions reached in this report. The AP1000 DCD site parameter value conservatively bounds the applicant's site characteristic for snow load, and therefore, the staff finds this acceptable.

2.3.1.4.2.6 *100-Year Return Period Fastest Mile of Wind*

In WLS COL FSAR Section 2.3.1.2.8, the applicant provided a 100-year return period fastest-mile wind estimate for Cherokee County, SC, of 140 km/hr (88 mi/hr) based on NWS-recorded storm events and a Gumbel-Lieblein extreme value analysis. The dataset included a maximum reported value of 130 km/hr (80.6 mi/hr or 70 knots). Using the same data source, the staff independently confirmed the applicant's wind estimate to be numerically reasonable. Given the stated POR (i.e., 1950 to 2006), the dataset used by the applicant appears to be from the NCDC's online "Storm Events" database, where thunderstorm winds and high winds typically represent a wind gust (i.e., a 5-second average wind speed). The staff notes that the AP1000 DCD site parameter for the operating basis wind speed, while associated with a 100-year return period, represents a 3-second gust wind speed rather than a fastest-mile wind speed.

The applicant estimated the 100-year return period site characteristic 3-second gust wind speed for the proposed WLS site based on ASCE 7-95, “Minimum Design Loads for Buildings and Other Structures.” The applicant stated that the 50-year return period 3-second gust speed is 150 km/hr (90 mi/hr) and used a conversion factor of 1.07 (based on ASCE-95. Table C6-5) to determine a 100-year return period 3-second gust speed of 155 km/hr (96 mi/hr). Although the applicant refers to this value as the design basis wind velocity, this site parameter is more appropriately referred to as the operating basis wind speed consistent with NUREG-0800, Section 2.3.1, WLS COL FSAR Table 2.0-201, and AP1000 DCD, Tier 2, Table 2-1.

The staff compared the applicant’s 3-second gust wind speed against a more recent version of the industry standard, “Minimum Design Loads for Buildings and Other Structures” (i.e., ASCE/SEI 7-05, which is cited in NUREG-0800, Section 2.3.1). The staff noted the values were the same. Further, this site characteristic value was also compared by the staff to various reports of high wind events in the eight counties that include and surround the proposed WLS site (i.e., Cherokee, Spartanburg, Union, Chester, and York in South Carolina, and Cleveland, Gaston, and Mecklenburg in North Carolina) as recorded in the NCDC “Storm Events” database. Thunderstorm wind gusts of 150 km/hr (80 knots (about 92 mi/hr)) were reported in Spartanburg County, SC in 1991 and 1992, and Mecklenburg County, NC in 1962. Based on the staff’s independent review, the 100-year return period 3-second gust site characteristic value is considered to be reasonably representative of conditions that might be expected to occur in the WLS site area.

2.3.1.4.2.7 *Probable Maximum Annual Frequency and Duration of Dust Storms*

The following discussion on dust storms intends to provide a general understanding of the occurrence of this phenomenon in the site region but does not result in the generation of site characteristics for use as design or operating bases.

The applicant stated that there have been no reported dust storms in the WLS site area. Consistent with NUREG-0800, Section 2.3.1, the applicant compiled this information from the NCDC. The staff independently confirmed the information provided by the applicant and finds it reasonable because this area is not prone to dust storms.

2.3.1.4.2.8 *Climate Change*

RAI 446, Question 02.03.01-10 was asked by the staff to the applicant in light of 10 CFR 52.79(a)(1)(iii), which requires the applicant to identify the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and time in which the historical data have been accumulated. In addition, NUREG-0800, Section 2.3.1, states that the applicant should substantiate the applicability of data on severe weather phenomena to represent site conditions during the expected period of reactor operation.

In an October 20, 2008, response to RAI 446, Question 02.03.01-10, the applicant identified that certified climatological data was used for the severe weather phenomena evaluations and that the data selected supports accurate severe weather phenomena projections for the area near the proposed WLS site. The site characteristic temperatures were developed considering both 100-year return period and zero percent exceedance temperatures. The applicant stated that the margin between the WLS site characteristics and the DCD site parameters accounts for any limitations to the accuracy, quantity, and period of time in which the historical data have been accumulated, and the applicant predicts the site parameters presented in the DCD to be bounding for the expected period of reactor operation.

In addition, the applicant stated that the WLS site characteristics were already based on an extensive record of readily available and well-documented climatological observations. The applicant also cited that general predictions of potential global or U.S. climate change, particularly on severe weather phenomena during the period of reactor operation, are uncertain and only possible on a macroclimatic scale. Maximum data spans were used in the severe weather analysis for the proposed WLS site, and projections were made based on best-available historic data.

The U.S. Global Change Research Program (USGCRP) released a report to the President and Members of Congress in June 2009,[18] and again in May 2014,[19] titled, "Global Climate Change Impacts in the United States," and, "Climate Change Impacts in the United States." These reports, produced by an advisory committee chartered under the Federal Advisory Committee Act, summarize the science of climate change and the impacts of climate change on the United States.

In the southeast region, which encompasses both North and South Carolina, both the 2009 and 2014 report noted that average annual air temperature did not increase significantly over the last century as a whole. However, the 2009 report indicated that average annual temperature has risen about 1.1 °C (2 °F) since 1970, with the most significant seasonal increase occurring during the winter months. The 2009 report also stated that average temperatures are projected to rise even more by the 2080s, by a potential range of 2.5 °C to 5.0 °C (4.5 °F to 9.0 °F) (with a possible 5.83 °C (10.5 °F increase in the summer), depending on the released emissions scenario. The 2014 report provides a slightly lower increased temperature range for the region of 2.2 °C to 4.4 °C (4 °F to 8 °F), but this range accounts for both higher and lower emission scenarios throughout this century.

The 2009 report mentioned that, despite the fact that autumn precipitation has increased by about 30 percent in the southeast region since the early 1900s, summer and winter precipitation have both decreased by about 10 percent. The 2009 report states that the percentage of the region in moderate to severe drought has risen by 12 percent and 14 percent, respectively, since the mid-1970s. The 2014 report also states, "Because the Southeast is located in the transition zone between projected wetter conditions to the north and drier conditions to the southwest, many of the model projections show only small changes relative to natural variations. However, many models do project drier conditions in the far southwest of the region

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[18] U.S. Global Change Research Program, "Global Climate Change Impacts in the United States," A State of Knowledge Report, Federal Advisory Committees, June 2009.
[19] U.S. Global Change Research Program, "Climate Change Impacts in the United States," US National Climate Assessment, Federal Advisory Committees, May 2014.

and wetter conditions in the far northeast of the region, consistent with the larger continental-scale pattern of wetness and dryness." As far as future changes in total precipitation, climate models are not as consistent with predicting precipitation totals for the region as compared to predicting temperature changes, so there is no consensus on how seasonal amounts will change, except for possible indications that rainfall amounts from individual hurricanes will increase.

Both USGCRP reports also mention that the intensity and occurrences of hurricanes have increased in recent decades and are likely to increase even more into this century, correlating with an increase in sea surface temperatures. The 2014 report even goes on to state that the number of Category 4 and 5 hurricanes in the Atlantic basin has increased substantially since the early 1980s when compared to the historical record dating back to the mid-1880s; this can be attributed to both natural variability and climate change. However, there has been little change regarding the total number of landfalling hurricanes because a variety of features influence whether a hurricane can potentially make landfall. As mentioned previously, hurricane-related rainfall amounts, in addition to wind, are expected to increase.

Both reports mention that severe weather event reports, including tornado reports, have increased since the 1950s, but improvements in monitoring technologies, growth in population, and increased public awareness most likely account for this surge in reports. When these factors are taken into account, there is no clear trend in the frequency or strength of tornadoes since the 1950s in the United States.

The staff acknowledges that long-term climatic change resulting from human or natural causes may introduce changes into the most severe natural phenomena reported for the site. However, no conclusive evidence or consensus is available on the rapidity or nature of such changes. There is a level of uncertainty in projecting future conditions because the assumptions regarding the future level of emissions of heat trapping gases depend on projections of population, economic activity, and choice of energy technologies. If it becomes evident that long-term climatic change is influencing the most severe natural phenomena reported at the site, the COL holders have a continuing obligation to ensure that their plants stay within the licensing basis.

2.3.1.5 *Post Combined License Activities*

There are no post COL activities associated with this WLS COL FSAR section.

2.3.1.6 *Conclusion*

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to regional climatology, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the application are documented in NUREG-1793 and its supplements.

COL Information Item 2.3-1 states that a COL applicant shall address the site-specific regional climatology information. As set forth above, the applicant has presented and substantiated information to establish the regional meteorological characteristics. The staff reviewed the information provided and for the reasons given above, concludes that the applicant has established the meteorological characteristics at the site and in the surrounding area acceptable to meet the requirements of 10 CFR 100.20(c)(2) and 10 CFR 100.21(d) with respect to determining the acceptability of the site. The staff finds that the applicant has provided a sufficient description to adequately address COL Information Item 2.3-1.

The staff finds that the applicant has considered the most severe natural phenomena historically reported for the site and surrounding area in establishing the site characteristics. Specifically, the staff has accepted the methodologies used to analyze these natural phenomena and determine the severity of the weather phenomena reflected in these site characteristics. Since the applicant has correctly implemented these methodologies, as described above, the staff finds that the applicant has considered these historical phenomena with margin sufficient for the limited accuracy, quantity, and period of time in which the data have been accumulated in accordance with 10 CFR 52.79(a)(1)(iii).

2.3.2 Local Meteorology

2.3.2.1 *Introduction*

WLS COL FSAR Section 2.3.2, "Local Meteorology," addresses the local (site) meteorological parameters, assesses the potential influence of construction and operation of the proposed plant and its facilities (e.g., terrain modifications, buildings and other plant infrastructure, and sources of heat and moisture) on local meteorological conditions and the impact of those modifications on plant design and operation. WLS COL FSAR Section 2.3.2 also provides a topographical description of the site and its environs and the design basis dry and wet-bulb temperatures, in addition to the evaluation of the UHS. To follow the format of the WLS COL FSAR, the staff incorporated the design basis dry and wet-bulb temperatures and the evaluation of the UHS into Section 2.3.2 of this report.

This section of the report also addresses the supplemental information in WLS COL FSAR Section 2.3.6 related to local meteorology.

2.3.2.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 2.3, incorporates by reference AP1000 DCD, Revision 19, Section 2.3. In WLS COL FSAR Section 2.3.2, the applicant addressed the following:

AP1000 COL Information Item

- WLS COL 2.3-2

The applicant provided additional information in WLS COL 2.3-2 to address COL Information Item 2.3-2. WLS COL 2.3-2 addresses site-specific information related to local meteorology. In addition, the applicant chose to provide the design basis temperature and moisture-related site characteristics in WLS COL FSAR Section 2.3.2, rather than WLS COL FSAR Section 2.3.1, as suggested in RG 1.206, Subsection C.I.2.3.1.2, Part III, This portion of WLS COL 2.3-2 correlates with COL Information Item 2.3-1 from Tier 2, Table 1.8-2, of the AP1000

DCD. This additional design basis information addresses Interface Item 2.4 related to extreme meteorological conditions for the design of systems and components exposed to the environment and Interface Item 2.8 related to ambient air temperatures.

2.3.2.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. The acceptance criteria and information needed to evaluate local meteorological and climatological characteristics are based on meeting the relevant requirements of NRC regulations under 10 CFR Part 52 and 10 CFR Part 100. The staff considered the following regulatory requirements in reviewing the applicant’s descriptions of the local meteorological and climatological characteristics around the proposed WLS site:

- 10 CFR 52.79(a)(1)(iii), as it relates to identifying the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and time in which the historical data have been accumulated.

- 10 CFR 100.20(c)(2) and 10 CFR 100.21(d), with respect to the consideration given to the local meteorological and air quality characteristics of the site.

NUREG-0800, Section 2.3.2, recommends the following acceptance criteria to be provided by a COL applicant:

- Summaries of local meteorological data based on onsite measurements made in accordance with RG 1.23, “Meteorological Monitoring Programs for Nuclear Power Plants,” Revision 1, and NWS station summaries or other standard installation summaries from appropriate nearby locations (e.g., within 80 km (50 mi)) should be presented as specified in RG 1.206, “Combined License Applications for Nuclear Power Plants,” Part I, Subsection C.I.2.3.2.1.

- A complete topographical description of the site and environs out to a distance of 80 km (50 mi) from the plant, as described in RG 1.206, Part I, Subsection C.I.2.3.2.2, should be provided.

- A discussion and evaluation of the influence of the plant and its facilities on the local meteorological and air quality conditions should be provided. Applicants should also identify potential changes in the normal and extreme values, as presented in the safety analysis report, resulting from plant construction and operation.

- The description of local site airflow should include wind roses and annual joint frequency distributions (JFDs) of wind speed and wind direction by atmospheric stability for all measurement levels using the criteria provided in RG 1.23.

The following regulatory guides were also taken into consideration in the staff’s review of WLS COL FSAR Section 2.3.2 and Appendices 2CC and 2DD (as applicable):

- RG 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants," Revision 1, which provides criteria for establishing and operating an onsite meteorological measurements program for the collection of basic meteorological data needed to support plant licensing and operation.

- RG 1.206, "Combined License Applications for Nuclear Power Plants," which identifies the types of local meteorological and other climatological data summaries, topographic information, and any local meteorological and air quality conditions used for design and operating bases that an applicant should provide in FSAR Section 2.3.2 when describing these conditions at the site, and in the site vicinity and area.

2.3.2.4 *Technical Evaluation*

The staff reviewed WLS COL FSAR Section 2.3.2 and checked the applicable site parameters, including the design basis temperature and moisture related site parameters, in AP1000 DCD Tier 1 (Table 5.0-1) and AP1000 DCD Tier 2 (Table 2-1) to ensure that the combination of the AP1000 DCD and the COL application represents the complete scope of information relating to the review topic. The staff's review confirmed that the information contained in the application and incorporated by reference (as related to site parameters, where applicable) addresses the required information relating to local meteorology. The results of the staff's evaluation of the information incorporated by reference in the application are documented in NUREG-1793 and its supplements. The staff reviewed the information in the WLS COL FSAR corresponding to the following:

AP1000 COL Information Item

- WLS COL 2.3-2

WLS COL 2.3-2 addresses COL Information Item 2.3-2 in AP1000 DCD Tier 2, Table 1.8-2 regarding local meteorological conditions.

The staff reviewed WLS COL 2.3-2, related to local meteorological conditions included under WLS COL FSAR Section 2.3.2. COL Information Item 2.3-2 in AP1000 DCD, Section 2.3.6.2 states:

> Combined License applicants referencing the AP1000 certified design will address site-specific local meteorology information.

In this section, the staff reviewed information presented by the applicant concerning the local (site) meteorological parameters, including their normal and extreme values. Summaries of parameters should be based on long-term data from nearby reasonably representative locations (e.g., within 80 km (50 mi), and shorter-term onsite data, based on NRC guidance).

Representativeness of submitted data should be demonstrated by comparing the longer-term datasets to the shorter-term onsite data. The applicant should provide at least two consecutive annual cycles (and preferably three or more entire years) of onsite meteorological data for the evaluation. The applicant initially submitted the minimum amount (i.e., 1 year) of onsite meteorological data at the time of application and provided a complete 2-year dataset once it had been collected. Therefore, in RAI 447, Question 02.03.02-1, the staff requested that the applicant provide revisions related to discussions presented WLS COL FSAR Section 2.3.2 based on the completed 2-year dataset or justification for an alternate approach.

The applicant chose an alternative approach, which involved demonstrating that the first year of meteorological data was representative when compared to the 2-year dataset. The applicant stated that revision of the suggested sections would create an inconsistency in the WLS COL FSAR unless all sections that use the onsite data were revised. The applicant provided a new appendix (i.e., Appendix 2CC) to the WLS COL FSAR to demonstrate that analyses based on the first year of onsite monitoring data were consistent with the full, 2-year dataset.

Based on the information provided in the new appendix, the applicant concluded that the 2-year meteorological dataset is consistent with the first year dataset and the historic dataset from the nearby Greenville/Spartanburg (Greer) NWS station. No anomalous behavior was observed between the 2 years' comparison of the normal conditions observed at the nearby NWS station. The applicant concluded that WLS COL FSAR Section 2.3 needed no changes based on the second year of meteorological data collected.

The staff analysis results agree that the 2-year onsite meteorological dataset is consistent with the 1-year dataset. A comparison, between the 2 years, did not exhibit abnormal trends or behaviors. Therefore, this is acceptable to the staff. Sections 2.3.4 and 2.3.5 of this report evaluate further analysis of these data with respect to atmospheric dispersion modeling.

The staff relied upon the review procedure presented in NUREG-0800, Section 2.3.2, to independently assess the technical sufficiency of the information presented by the applicant.

2.3.2.4.1 Winds

2.3.2.4.1.1 *Offsite Wind Direction and Wind Speed Distributions*

In WLS COL FSAR Section 2.3.2.1.1, the applicant provided monthly and annual wind data summaries from data obtained at the first-order NWS station at Greenville/Spartanburg (Greer), SC for a POR from 1997 through 2005. This data is presented in WLS COL FSAR Tables 2.3-209 to 2.3-221. The applicant presented these data summaries in the form of joint frequency distributions of wind speed and wind direction and graphical wind roses. NUREG-0800, Section 2.3.2, recommends that the offsite wind summaries be based on data from nearby representative stations (e.g., within 80 km (50 mi). The Greenville/Spartanburg (Greer) NWS station is approximately 66 km (41 mi) west-southwest of the site; the staff does not consider wind data from this NWS station to be representative for characterizing site-specific wind speed and direction conditions. Therefore, in RAI 447, Question 02.03.02-2, the staff requested that the applicant support the statement that the net regional air movement can be deduced from the monthly wind joint frequency distributions for the Greenville/Spartanburg International Airport. The staff notes that the applicant revised WLS COL FSAR Section 2.3.2 by removing the statement. The staff finds that the WLS COL FSAR provides an acceptable description of the wind distribution for the proposed WLS site and considers RAI 447, Question 02.03.02-2 resolved.

2.3.2.4.1.2 *Lee Nuclear Wind Distribution*

For the WLS site, the applicant stated that the annual wind direction frequency is fairly uniform. The northwest wind direction represents the prevailing wind direction. Wind from the west occurs least frequently at 3 percent of the time. The applicant also stated that the mean annual wind speed was about 8 km/hr (5 mi/hr).

The staff acknowledges that the proposed WLS site is located in a river valley area having distinct wind characteristics unlike the Greenville/Spartanburg (Greer) and Charlotte wind distributions described in the previous section. The staff independently reviewed the wind speed and direction distributions for the proposed WLS site and finds the results consistent with the information provided by the applicant in the WLS COL FSAR.

2.3.2.4.1.3 *Wind Direction Persistence*

WLS COL FSAR Tables 2.3-242 through 2.3-244 summarize maximum persistence periods (in consecutive hours) by direction sector for each year of a 9-year POR from 1997 to 2005. These tables are based on hourly observations taken at the Greenville/Spartanburg (Greer) NWS station. Each sector also identifies the maximum and average persistence durations for the composite 9-year POR. The applicant indicated that the longest periods of wind direction persistence for a single sector and for three and five adjoining sectors were all associated with winds from the northeast (i.e., 23, 82, and 150 consecutive hours). The northeast sector also showed the highest average of all annual persistence maxima during the nine-year POR (i.e., 14.0, 57.8, and 91.0 hours).

WLS COL FSAR Table 2.3-245, “Comparison of Maximum Wind Persistence at Lee Nuclear Station Site and Greenville/Spartanburg, South Carolina,” lists wind direction persistence durations for a single sector and for three and five adjacent sectors based on wind measurements at the 10 m (33 ft) level from the proposed WLS site during the period of December 1, 2005, through November 30, 2006. WLS COL FSAR Figure 2CC-208, “Wind Direction Frequency (10 m (33 ft) level),” and Figure 2CC-209, “Wind Direction Frequency (60 m (197 ft) level),” compare the wind direction frequencies for the 1-year and 2-year periods of the onsite meteorological record. These two figures show that the datasets are very similar in regards to wind direction frequency. For comparison, WLS COL FSAR Table 2.3-245 also summarizes persistence based on hourly data from the Greenville/Spartanburg (Greer) NWS station, which corresponds to the same period of onsite data as well as the maximum persistence durations, again by sector, identified in WLS COL FSAR Tables 2.3-242 through 2.3-244.

For the onsite meteorological data, the applicant indicated that the longest periods of persistence for a single sector and for three adjoining sectors were associated with winds from the northwest and that the longest duration of persistence for five adjoining sectors was for the north-northeast direction. No comparative statement regarding persistence was provided between the concurrent onsite and offsite datasets or between the 1-year and 9-year offsite datasets. However, the staff notes that during the concurrent offsite data period, the longest durations of persistence were:

- 15 hours for a single-sector under north winds

- 50 hours for three adjacent sectors under north-northeast winds
- 88 hours for five adjacent sectors under southwest and west-southwest winds

The concurrent onsite/offsite datasets did not compare very well in terms of directional persistence. The same can be said when comparing the persistence statistics based on the one-year onsite dataset and the nine-year dataset from the Greenville/Spartanburg (Greer), SC, NWS station and an inter-comparison between the one and nine-year offsite datasets. The wind direction persistence data does not directly lead to any site characteristic values and therefore is accepted as general climatological information. Offsite wind information is discussed in more detail in Section 2.3.2.4.1.1 of this report.

2.3.2.4.2 Air Temperature

In WLS COL FSAR Section 2.3.2.2, the applicant stated that the average maximum temperatures at the site ranged from between 10 and 13 °C (50 and 55 °F) in January to between 29 and 32 °C (85 and 90 °F) in July. Comparatively, average minimum temperatures ranged from between -3.9 and -1.1 °C (25 and 30 °F) in January to between 18 and 21 °C (65 and 70 °F) in July. The applicant stated that the average maximum monthly temperature at the Ninety-Nine Islands cooperative observing station occurred in July with a temperature of 32 °C (89 °F). The average minimum monthly temperature was -2.94 °C (26.7 °F), which occurred in January at the Ninety-Nine Islands station. Using local climate data provided by NCDC and data provided by the South Carolina State Climatology Office, the staff independently reviewed the average temperatures and average temperature ranges and finds this information acceptable.

Unlike wind speed and direction, ambient air temperatures across the region are similar. FSAR Figure 2CC-201, "Dry Bulb Temperature Comparison," shows a comparison of ambient dry-bulb temperature statistics between measurements provided by the applicant and measurements provided by a first-order NWS station and a cooperative observing station. The staff considers the Greenville/Spartanburg airport off-site station representative of expected site conditions and accepts site-specific dry-bulb and wet-bulb temperature summaries based on data from this station.

NUREG-0800, Section 2.3.1, identifies the need for meteorological data in the design of the ultimate heat sink. The applicant provided an analysis using only 9 years (1997-2005) of meteorological data. Therefore, in RAI 447, Question 02.03.02-3, the staff requested that the applicant provide justification for not providing 30 years of meteorological data, a standard recommend by NUREG-0800 in Section 2.3.1.

In a November 25, 2008, response to RAI 447, Question 02.03.02-3, the applicant stated that, for a passive containment plant (e.g., AP1000), a site-specific UHS evaluation was not necessary. The applicant revised WLS COL FSAR Section 2.3.2.2 by removing the worst 1-day, 5-day and 30-day discussion and by deleting WLS COL FSAR Tables 2.3-247 through 2.3-252. The staff finds this acceptable because many plants use a cooling tower as a UHS to dissipate residual heat after an accident. Instead of using a cooling tower to release heat to the atmosphere, the AP1000 design uses a passive containment cooling system (PCS) to provide the safety-related UHS. The PCS is designed to withstand the maximum safety dry-bulb and coincident wet-bulb air temperature site parameters specified in AP1000 DCD Tier 1, Table 5.0-1 and Tier 2, Table 2-1. Therefore, the staff notes that the applicant need not identify meteorological characteristics for evaluating the design of a UHS cooling tower.

The applicant reported the maximum safety dry and wet-bulb temperature and the minimum safety dry-bulb temperature site characteristics in WLS COL FSAR Table 2.0-201 using the 0.4 percent exceedance values summarized by the American Society of Heating, Refrigerating and Air Conditioning Engineers (ASHRAE), as listed in WLS COL FSAR Table 2.3-293, "Lee Nuclear Station Design Temperatures." This was not consistent with the methodology of choosing the higher of the historic (zero percent exceedance) value(s) or the 100-year return period value(s) based on 10 CFR 52.79(a)(1)(iii). In RAI 447, Question 02.03.02-4, the staff requested that the applicant address this concern.

In a November 25, 2008, response to RAI 447, Question 02.03.02-4, the applicant provided a revision that provided the zero percent exceedance values rather than the higher 100-year return period values for the maximum and minimum safety dry- or wet-bulb temperature site characteristics. As summarized in WLS COL FSAR Table 2.0-201, the applicant used 100-year return period temperatures for comparison against the AP1000 maximum and minimum safety air temperature site parameters. The applicant also used the 0.4 percent annual exceedance temperatures for comparison against the AP1000 maximum and minimum normal air temperature site parameters. The staff reviewed the site characteristic temperature values presented in WLS COL FSAR Table 2.0-201 and finds them acceptable and considers RAI 447, Question 02.03.02-4 resolved.

2.3.2.4.3 Atmospheric Moisture

In WLS COL FSAR Section 2.3.2.3, the applicant addressed atmospheric moisture conditions in terms of relative humidity levels observed at the site and the nearby first-order NWS station located at the Greenville/Spartanburg airport. The applicant also discussed several other topics related to atmospheric moisture in WLS COL FSAR Section 2.3.2.3. These topics include precipitation, snowfall, and fog in WLS COL FSAR Sections 2.3.2.3.1 through 2.3.2.3.3. The staff's evaluation of this information follows a similar sequence.

In the analysis of atmospheric moisture, the applicant stated that the local site meteorological conditions reflect the synoptic-scale atmospheric processes consistent with the regional meteorology, except for the local effects of the Broad River. The applicant further stated that there was slightly higher humidity directly adjacent to the river, and that the site humidity data was more appropriate for site estimates than the Greenville/Spartanburg data. In RAI 447, Question 02.03.02-5, the staff requested that the applicant clarify why the wet-bulb temperature site characteristic, as presented in WLS COL FSAR Table 2.0-201 and based on data from the Greenville/Spartanburg NWS station, was more appropriate for site estimates.

In a November 25, 2008, response to RAI 447, Question 02.03.02-5, the applicant stated that the site data had been recorded for only a relatively short period and that no long-term wet-bulb or humidity measurements were recorded at a location near the site and in close proximity to a body of water. The applicant stated that the variability in humidity measurements because of a body of water diminishes dramatically within a short distance of the shoreline. Therefore, it was expected that any increase in humidity at the site location would be minimal. The applicant also stated that significant variability exists within a single year dataset when compared to a 30-year dataset (i.e., data from the Greenville/Spartanburg (Greer) NWS station). As a result, the applicant chose to use the longer-term NWS dataset to minimize variability and more accurately represent long-term averages. The staff notes that the revised WLS COL FSAR Section 2.3.2 clarifies the applicability of the Greenville/Spartanburg (Greer) data, which the staff finds acceptable and, therefore, considers RAI 447, Question 02.03.02-5 resolved.

2.3.2.4.3.1 *Precipitation*

The applicant indicated that precipitation averages 122.9 cm (48.37 in.) annually at the Ninety-Nine Islands cooperative observing station and is generally well distributed throughout the year, as shown in WLS COL FSAR Table 2.3-246, "Ninety-Nine Islands Monthly Climate Summary, NCDC 1971-2000 Monthly Normals," (as reported by the Southeast Regional Climate Center). A total rainfall of 100.9 cm (39.72 in.) was reported for the initial 1-year onsite dataset (December 2005, through November 2006) at the proposed WLS site. In WLS COL FSAR Appendix 2CC.2 of the "Precipitation Comparison" section, the applicant reported a total rainfall of 83.1 cm (32.7 in.) for the second year (December 2006 through November 2007), and cited drought conditions, in addition to geographical influences, as a possible factor for the decrease in precipitation between the two years. The staff independently confirmed the information provided by the applicant and finds it acceptable.

The staff was unable to verify the accuracy of the Greenville/Spartanburg (Greer) NWS data presented in WLS COL FSAR Section 2.3.2.3.1. Therefore, in RAI 447, Question 02.03.02-6, the staff requested that the applicant provide further justification of the data. In a November 25, 2008, response to RAI 447, Question 02.03.02-6, the applicant revised the applicable information regarding precipitation under WLS COL FSAR Sections 2.3.2.1, 2.3.7, and Table 2.3-256, "Precipitation Data (inches of rain) Greenville/Spartanburg, South Carolina," with data reported in the 2007 NCDC LCD summary for Greenville/Spartanburg (Greer), SC. The staff independently verified the new information and finds the response acceptable.

The staff also confirmed the average daily and monthly precipitation and extreme precipitation statistics provided by the applicant in WLS COL FSAR Figures 2.3-242 and 2.3-243 and independently confirmed the accuracy of WLS COL FSAR Tables 2.3-259, 2.3-260, and 2.3-261. The three tables summarized the monthly percent of total observations of precipitation and wind direction, the monthly rainfall frequency distribution, and various precipitation data, including the maximum 24-hour rainfall, at the proposed WLS site. These data showed that the highest rainfall frequency occurs during the months of October through January and the highest frequency directions are north through northeast. The staff finds the applicant's data acceptable.

2.3.2.4.3.2 *Snow*

The applicant stated that the average annual snowfall at the proposed WLS site is approximately 7.6 cm (3.0 in.). This estimate was based on snowfall data from the nearby Ninety-Nine Islands cooperative observing station. Based on the NCDC U.S. Snow Climatology Database for three nearby observation stations (Gaffney 6E, Gaston Shoals, and Ninety-Nine Islands) in Cherokee County, the staff independently confirmed the applicant's annual snowfall estimate to be reasonable.

The applicant also stated that the maximum monthly snowfall at Ninety-Nine Islands was 35 cm (14 in.) in February 1978-79. The staff concluded that the maximum 24-hour snowfall for Cherokee County was approximately 33 cm (13 in.). Therefore, the staff finds the applicant's discussion on particular maximum snowfall amounts acceptable.

2.3.2.4.3.3 *Fog*

The staff notes that onsite fog observations are not taken. The closest observation location that routinely records the occurrence of fog conditions is the NWS station at Greenville/Spartanburg (Greer), SC (approximately 66 km (41 mi) from the WLS site). The applicant stated that this station has averaged approximately 38 hours per year of fog, with November, December, and January having the greatest frequency of occurrence. The staff was unable to confirm these values. Therefore, in RAI 447, Question 02.03.02-7 for clarification on the information reported in the WLS COL FSAR.

In a November 25, 2008, response to RAI 447, Question 02.03.02-7, the applicant stated that its statistics on fog occurrence were based on 9 years (i.e., 1997-2005) of unedited data from the NCDC for Greenville/Spartanburg (Greer), SC, for any hourly observation described as "fog." The applicant also provided a second estimate of 29 hours per year of heavy fog conditions (visibility less than or equal to 0.4 km (0.25 mi) based on 44 years (1964-2007) of edited data from the NCDC LCD summary for the Greenville/Spartanburg (Greer) NWS station. The applicant's original dataset of approximately 38 hours per year of fog was more conservative when compared with the NCDC data, so the applicant retained this dataset.

The staff notes that based on the 2006 NCDC LCD summary for Greenville/Spartanburg airport for a period of 43 years, there are 29.3 days per year, on average, with heavy fog conditions (visibility less than or equal to 0.4 km (0.25 mi). November, December, and January have the highest average number of days with observed heavy fog conditions.

Since these data do not establish design-related site characteristics, the staff accepts the applicant's data as reasonable estimates. However, the staff also recognizes that the higher frequency of occurrence based on the 9-year dataset includes any observation-recorded fog as opposed to those conditions that qualify, more stringently, as "heavy fog." Nevertheless, a higher frequency of occurrence is a reasonably representative indicator for this weather element because of the site's location in the valley of the Broad River.

2.3.2.4.4 Atmospheric Stability

The information presented by the applicant regarding atmospheric stability in WLS COL FSAR Section 2.3.2.4 is not to be understood as having the same meaning as atmospheric stability determined from meteorological tower measurements in accordance with RG 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants," Revision 1 (see WLS COL FSAR Table 2.3-281). In addition to this, the information is not used as input for the dispersion modeling analyses addressed in WLS COL FSAR Sections 2.3.4 and 2.3.5 (see WLS COL FSAR Section 2.3.4.2). Rather, WLS COL FSAR Section 2.3.2.4 discusses atmospheric stability in terms of the frequency and relative strength of temperature inversions based on regional-scale observations taken by balloon soundings.

The applicant evaluated the frequency and strength of inversion layers using 7 years of weather balloon data collected at the Greensboro radiosonde station. An inversion refers to a layer of air in which the dry-bulb temperature increases with height. In WLS COL FSAR Table 2.3-276, “Inversion Heights and Strengths, Greensboro, North Carolina,” the applicant presented annual averages of the number of mornings and afternoons containing temperature inversions, the average elevation of these inversion layers, and the average strength of the inversion, in terms of the change in degrees Celsius per meter in height. The staff compared delta-temperature measurements at the proposed WLS site from December 2005 - November 2006 to the inversion data presented by the applicant. The staff concluded that the average morning and afternoon inversion strength at the proposed WLS site is consistent with those values presented by the applicant. The staff finds the applicant’s discussion on atmospheric stability reasonable and acceptable.

Information on temperature inversions, like the discussions of ventilation rates and air stagnation in WLS COL FSAR Section 2.3.1.2.6, and on mixing heights and ventilation rates in WLS COL FSAR Section 2.3.2.4.1, provides a general indication of atmospheric dispersion potential in the larger region that includes the site. However, information on any of these conditions does not result in the generation of site characteristics used as design or operating bases. The staff finds the applicant’s discussion on atmospheric stability in WLS COL FSAR Section 2.3.2.4 reasonable.

2.3.2.4.4.1 *Mixing Heights*

As indicated previously in this report, the applicant obtained mixing height data based on measurements taken at the Greensboro-High Point, upper air observing station and archived in the EPA Support Center for Regulatory Atmospheric Modeling (SCRAM) database. Initially, the staff was unable to reproduce the mean afternoon mixing height values summarized by the applicant in WLS COL FSAR Table 2.3-277, “Mixing Heights at Greensboro, North Carolina.” Therefore, in RAI 447, Question 02.03.02-8, the staff requested that the applicant verify these values. In a November 25, 2008, response, the applicant stated that the monthly mixing height values were averaged directly from the referenced EPA SCRAM morning and afternoon mixing height data for a 6-year POR (1984-1987, 1990, and 1991). The staff reviewed the averages obtained from this data and finds them correct.

Using the online VCIS database (see Section 2.3.1.4.2.4 of this report), the staff determined average morning mixing heights for the WLS site area about 20 percent lower than mixing heights reported in WLS COL FSAR Table 2.3-277 but with similar month-to-month variation (i.e., lowest values occurring during autumn and highest values seen from about mid-winter to early spring). The VCIS database generated mean afternoon mixing heights within about 10 percent of the values reported in WLS COL FSAR Table 2.3-277. Month-to-month variation is essentially the same with the lowest mixing height values occurring from about mid-autumn to mid-winter and the highest values seen from about mid-spring to mid-summer. At the time of the initial WLS COL FSAR submittal, the VCIS database was available through the Bureau of Land Management. However, this database appears to be unsupported or unavailable for use. The staff notes that the mixing height data has not changed in the latest revision of the WLS COL FSAR and, therefore, the staff still finds this analysis correct and adequate.

The staff recognizes that there are no corresponding site characteristics related to mixing height, nor are these data input for any of the applicant's atmospheric dispersion modeling. Therefore, the staff considers the applicant's discussion of mixing height conditions in the WLS site region reasonable and acceptable.

2.3.2.4.5 Potential Influence of the Plant and Its Facilities on Local Meteorology

WLS COL FSAR Section 2.3.2.5 states that the potential for the operation of WLS Units 1 and 2 to influence (impact) local climatology (meteorological conditions) will be negligible. The applicant later addresses several aspects of plant operations that could have climate-related effects on the environment – particulate emissions (presumably linked to the increased occurrence of fog and haze) and the discharge of heat into the Broad River.

The applicant expects the likely sources of particulate emissions (i.e., infrequent diesel generator operations and a modest increase in vehicular traffic) to only result in a negligible net increase in particulates. Most of the plant heat will be ejected into the atmosphere by the cooling towers, resulting in a relatively small amount of heat input for the Broad River. The applicant then reiterates its earlier conclusion. Although not discussed in much detail, the staff agrees with the applicant's assessment that operational effects on meteorological conditions should be minimal.

In the context of potential construction-related impacts, in WLS COL FSAR Section 2.3.2.5, the applicant briefly describes some of the "climate-shaping" topographic features at and in the area of the proposed WLS site (e.g., the proximity of the proposed WLS site to the Ninety-Nine Islands Reservoir and the Broad River). These features potentially lead to increased humidity levels at the site and demonstrate a slight tendency for the river valley to channel lower-level winds. The applicant also identified several new types of construction activities that will take place at the site (e.g., some ground leveling, some tree removal, the building of plant structures and facilities). The applicant also indicated that no significant changes are predicted or proposed in terms of local hydrologic features and anticipate no changes to local roadways in support of the proposed new facility.

As discussed above, the applicant indicated that there is a slight tendency for the Broad River valley to channel lower-level winds along its length. The staff agrees with this observation but further notes that these conditions appear to be associated with night-time drainage flow as these conditions predominate during moderately and extremely stable atmospheric conditions (i.e., Stability Classes F and G) as indicated by the JFDs presented in WLS COL FSAR Tables 2.3-235 through 2.3-241. This characteristic is important to recognize because it is reflected in the locations of the maximum offsite, accident-related dispersion estimates addressed in WLS COL FSAR Section 2.3.4 and the maximum offsite, routine release-related dispersion estimates discussed in WLS COL FSAR Section 2.3.5.

2.3.2.4.5.1 *Cooling Tower Plumes*

In WLS COL FSAR Revision 9, the applicant's design included three circular mechanical draft cooling towers (CMDCTs). In WLS COL FSAR Revision 9, the design changed to reflect two CMDCTs per unit (4 total), still applicable to the current revision. Each unit will result in the emission of small water droplets entrained in the tower's exhaust airflow (i.e., drift). The droplets contain the dissolved solids found in the circulating water (e.g., salts) that may eventually deposit on the ground as well as on structures, equipment, and vegetation. The use of drift eliminators, which rely on inertial separation caused by exhaust flow directional changes, controls the drift droplet emissions. Drift eliminators installed in the CMDCTs are capable of reducing the emissions to approximately 0.0005 percent of the circulating water flow. Under relatively high wind speeds and humid conditions, the aerodynamic wake turbulence caused by air flowing around the tower housing may result in the visible plume touching down causing ground level fogging and, under freezing conditions, icing.

The applicant stated that an analysis of the potential environmental impacts caused by the operation of CMDCTs was conducted using the EPRI-sponsored Seasonal/Annual Cooling Tower Impact (SACTI) Program. The applicant further chose NWS Charlotte (CLT) wind distributions to represent site conditions. In RAI 447, Question 02.03.02-9, the staff requested that the applicant provide SACTI input files for conducting a confirmatory analysis and a justification for using CLT wind data.

In a November 25, 2008, response to RAI 447, Question 02.03.02-9, the applicant provided SACTI input files and a justification for using NWS data from CLT. The applicant chose CLT data because a 5-year dataset (2001-2005) was available from CLT, but less than 1 year of data was available from the proposed WLS site at the time of the analysis. CLT was judged to be spatially more representative than NWS station at Greenville/Spartanburg in terms of distance from the site and ground elevation. CLT's annual average wind speed and annual average humidity also appear to be closer to onsite data than to the Greenville/Spartanburg (Greer) data. The applicant further performed sensitivity tests to address the differences in the three datasets.

In WLS COL FSAR Table 2.3-278, "Visible Plume Frequency of Occurrence by Season (All Wind Directions)," the applicant provided the visible plume frequency of occurrence by season. The applicant stated that the majority (i.e., greater than 50 percent) of visible plumes did not reach 1000 m (0.6 mi) downwind and 300 m (0.2 mi) in height. The longest and largest visible plumes tend to occur in the winter, with smaller plumes occurring in the spring and especially in the summer. The cold air in winter tends to cause condensation of moist plumes more extensively when compared to warmer seasons because the dewpoint temperature and air temperature are normally closer together nearer the surface, which generally leads to complete saturation. Since both the dewpoint and air temperature are lower, the colder air has a much smaller capacity for holding water vapor. Further, the applicant stated that the largest visible plumes reach a downwind distance of 9900 m (6.2 mi) and a height of approximately 1600 m (1.0 mi) approximately one percent of the time.

In WLS COL FSAR Table 2.3-279, "Frequency of Plume Shadowing by Season (Average for All Wind Directions," the applicant provided the frequency of plume shadowing by season. The applicant stated that consistent with the visible plume frequency results, most plume shadowing occurs in the winter season, with lesser amounts in the spring and fall and the least amounts in the summer. The applicant stated further that plume shadowing effects reach 1200 m (0.75 mi) downwind 1 percent of the time, with the farthest impact reaching approximately 4000 m (2.5 mi) downwind 0.5 percent of the time on an annual basis.

Additionally, the applicant stated that the SACTI output showed that there were virtually no occurrences of ground level fogging, and more importantly, no occurrences of ground-level icing were predicted by SACTI.

In WLS COL FSAR Table 2.3-280, "Maximum Salt Drift Deposition Rate (KG/KM^2/MO), the applicant provided the maximum salt drift rate in kg/km^2 per month. The applicant stated that salt deposition was negligible, with a maximum rate of approximately 1.03 kg/km^2 (2.11E-07 lb/ft^2) per month occurring 200 m (656 ft) to the north of the towers in the summer, and that all other salt deposition amounts are below 1 kg/km^2 (2.05E-07 lb/ft^2) per month.

The staff focused on the salt deposition that would impact the switchyard southwest of Unit 1 and southeast of Unit 2 cooling towers at an approximate distance of 600 m (based on WLS COL FSAR Figure 2.1-201). The staff concluded from the results in WLS COL FSAR Table 2.3-280 that salt deposition to this switchyard would be well below 1 kg/km^2 (2.05E-07 lb/ft^2) per month (worst case 1.03 kg/km^2 (2.11E-07 lb/ft^2) per month).

The applicant stated that the maximum water deposition rate that occurs during the fall season is 740 kg/km^2 (1.52 lb/ft^2) per month at a downwind distance of 900 m (2,950 ft) southeast of the cooling towers. The applicant further stated that this rate was a trivial amount, equivalent to the rainfall equivalent of 0.8 microns (0.00003 in.) per month based on the density of water (i.e., 1000 kg/m^3 (62.43 lb/ft^3)). The staff confirmed the applicant's information based on FSAR Figure 2.3-278.

The applicant provided an electronic copy of the input and output files from the SACTI computer model. The staff reviewed the model input files to assure that the applicant made conservative assumptions. The SACTI results indicate that several months of salt accumulation would result in amounts far below the lower end of the "Light Contamination Level" range defined by the Institute of Electrical and Electronic Engineers (IEEE) standard. The staff independently verified the source cited by the applicant. The staff finds that the description of the total accumulation reaching amounts that require mitigation as highly unlikely because of local precipitation removing any salt deposits before it reaches a level of concern is acceptable and considers RAI 447, Question 02.03.02-9 resolved.

2.3.2.4.6 Topographical Description of the Surrounding Area

In WLS COL FSAR Section 2.3.2.6, the applicant provided a general understanding of the topographic features and characteristics within an 80-km (50-mi) radius of the proposed WLS site. The proposed WLS site is a greenfield site in that there are no operating nuclear units. The applicant indicated that the proposed site is located approximately 910 m (1000 yards) west of the Broad River. The staff notes, however, that at its closest approach the Broad River is only a few hundred yards from the northern boundary of the site based on WLS COL FSAR Figure 1.1-202. The applicant also stated that mountain ridges ranging in height from 300 to 760 m (1000 to 2500 ft) above MSL are located to the northwest, north, and northeast. Although not indicated in the description, the terrain elevation profiles in WLS COL FSAR Figure 2.3-246 show that the highest terrain is located at least 64 km (40 mi) from the proposed WLS site; elevations exceeding 0.3 km (1000 ft) above MSL are located as close as 16 km (10 mi) to the northeast.

Gently rolling hills characterize the terrain in the immediate vicinity of the proposed site. The applicant identified several elevated terrain features within about 8 km (5 mi) of the facility, the most notable being McKowns Mountain, located to the south--southwest with an elevation of approximately 240 m (800 ft) above MSL (or about 60 m (200 ft) above plant grade). Based on WLS COL FSAR Section 2.3.4.1, McKowns Mountain is within 1.6 km (1 mi) from the proposed site.

WLS COL FSAR Figure 2.3-245 displays topographic maps of the areas within 80 and 8 km (50 and 5 mi) of the proposed WLS site. While each sheet does state that the contour interval is 150 m (500 ft), there does not appear to be a reference contour for particular terrain elevations. The staff used WLS COL FSAR Figures 2.5.1-221 and 2.1-204 to provide additional information for the review of the near-field topographical description. Given the information provided in WLS COL FSAR Section 2.3.2.6 and other applicable locations in the application, the staff considers the general topographic description acceptable.

2.3.2.4.7 Current and Projected Site Air Quality Conditions

The applicant's discussion of current air quality conditions near the WLS site in WLS COL FSAR Section 2.3.2.7 presents some similarities with regard to the information pertaining to air quality initially presented in WLS COL FSAR Section 2.3.1.2.6, including similar discussions with respect to the following:

- the 8-hour NAAQS for ozone
- the regulatory citation documenting promulgation of that standard
- identification of the now current attainment status for that criteria air pollutant

Refer to Section 2.3.1.4.2.4 of this report for an explanation of the staff's clarifications and corrections pertaining to this information.

The applicant stated that these air quality characteristics are not expected to be a significant factor in the design and operating bases for proposed Units 1 and 2. The applicant also identified several sources of criteria pollutant emissions during facility operations (e.g., auxiliary boilers, emergency diesel generators, and station blackout generators) but indicated that this equipment should not develop into significant sources of such emissions because it would operate on an intermittent test or emergency basis. The staff agrees with this information and the applicant's assessment of the expected emissions, assuming that the equipment is operated and well maintained in accordance with good practice and any applicable permit requirements.

2.3.2.5 *Post Combined License Activities*

There are no post COL activities associated with this WLS COL FSAR section.

2.3.2.6 *Conclusion*

The staff reviewed the WLS COL application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to local meteorology, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the application are documented in NUREG-1793 and its supplements.

COL Information Item 2.3-2 states that a COL applicant will provide site-specific local meteorology information. As set forth above, the applicant has presented and substantiated information describing the local meteorological conditions and the air quality and topographic characteristics important to evaluating the adequacy of the design and siting of this plant. The staff reviewed the information provided in COL Information Item 2.3-2 and, for the reasons given above, concludes that the identification and consideration of the meteorological, air quality, and topographical characteristics of the site and the surrounding area are acceptable and meet the requirements of 10 CFR 100.20(c)(2) and 10 CFR 100.21(d). In addition, COL Information Item 2.3-2 includes COL Information Item 2.3-1 when reviewing the design basis dry and wet-bulb site characteristics pertaining to extreme meteorological conditions for the design of structures, systems, and components exposed to the environment and ambient air temperatures. The staff finds that the applicant has provided a sufficient description to adequately address COL Information Item 2.3-2 and, where applicable, COL Information Item 2.3-1.

The staff finds that the applicant has considered the appropriate site phenomena in establishing the site characteristics. Specifically, the staff has accepted the methodologies used to determine the meteorological, air quality, and topographic characteristics. Since the applicant has correctly implemented these methodologies, as described above, the staff finds that the site characteristics, including margins, are sufficient for the limited accuracy, quantity, and period of time in which the data have been accumulated in accordance with 10 CFR 52.79(a)(1)(iii).

2.3.3 Onsite Meteorological Measurements Program

2.3.3.1 *Introduction*

The WLS onsite meteorological measurements program addresses the need for onsite meteorological monitoring and the resulting data. The staff's review covers the following specific areas: (1) meteorological instrumentation, including siting of sensors, sensor type, and performance specifications, methods and equipment for recording sensor output; the quality assurance program for sensors and recorders, data acquisition and reduction procedures, and special considerations for complex terrain sites; and (2) the resulting onsite meteorological database, including consideration of the period of record and amenability of the data for use in characterizing atmospheric dispersion conditions.

The WLS COL FSAR describes a program, which, if successfully implemented, provides an appropriate onsite meteorological measurements program and that data from this program provides an acceptable basis for estimating atmospheric dispersion for DBAs and routine releases from an AP1000 design.

Supplemental information in WLS COL FSAR Section 2.3.6 related to the onsite meteorological measurements program is addressed in this report.

2.3.3.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 2.3, incorporates by reference AP1000 DCD, Revision 19, Section 2.3.

AP1000 COL Information Item

- WLS COL 2.3-3

The applicant provided additional information in WLS COL 2.3-3 to address COL Information Item 2.3-3. WLS COL 2.3-3 addresses the onsite meteorological measurements program. In addition, this WLS COL FSAR section addresses Interface Item 2.9 related to the onsite meteorological measurements program.

2.3.3.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. The acceptance criteria and information needed to evaluate an onsite meteorological measurements program are based on meeting the relevant requirements of NRC regulations under 10 CFR Part 20, 10 CFR Part 50, 10 CFR Part 52, and 10 CFR Part 100. The staff considered the following regulatory requirements in reviewing the applicant's descriptions of the pre-construction (pre-operational) and operational phases of the onsite meteorological measurements program:

- 10 CFR Part 20, Subpart D, as it relates to the meteorological data used to demonstrate compliance with dose limits for individual members of the public. 10 CFR Part 50, Paragraphs 50.47(b)(4), 10 CFR 50.47(b)(8), and 10 CFR 50.47(b)(9), as well as Appendix E, "Emergency planning and preparedness for production and utilization facilities," Section IV.E.2 as they relate to the onsite meteorological information available for determining the magnitude and continuously assessing the impact of releases of radioactive materials to the environment during a radiological emergency.

- 10 CFR Part 50, Appendix A, GDC 19, "Control room," as it relates to the meteorological considerations used to evaluate the personnel exposures inside the control room during radiological and airborne hazardous material accident conditions.

- 10 CFR Part 50, Appendix I, as it relates to meteorological data used in determining compliance with the numerical guides for design objectives and limiting conditions for operation to meet the requirement that radioactive material in effluents released to unrestricted areas be kept as low as is reasonably achievable (ALARA).

- 10 CFR 100.20(c)(2), as it relates to the meteorological characteristics of the site that are necessary for safety analysis or that may have an impact upon plant design in determining the acceptability of a site for a nuclear power plant.

- 10 CFR 100.21(c), as it relates to the meteorological data used to evaluate site atmospheric dispersion characteristics and establish dispersion parameters such that: (1) radiological effluent release limits associated with normal operation can be met for any individual located off site, and (2) radiological dose consequences of postulated accidents meet prescribed dose limits at the EAB and the outer boundary of the LPZ.

The following regulatory guide is applicable for this section:

- RG 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants," Revision 1

NUREG-0800, Section 2.3.3, requires the following acceptance criteria to be provided by a COL applicant:

- The pre-operational and operational phase monitoring programs should be described, including: (1) a site map (drawn to scale) that shows meteorological tower location and True North with respect to manmade structures, topographic features, and other features that may influence site meteorological measurements; (2) distances to nearby obstructions to air flow in each downwind direction sector; (3) measurements made; (4) elevations of measurements; (5) exposure of instruments; (6) instrument descriptions; (7) instrument performance specifications; (8) calibration and maintenance procedures and frequencies; (9) data output and recording systems; and (10) data processing, analysis, and archiving procedures.

- Meteorological data should be presented in the form of JFDs of wind speed and wind direction by atmospheric stability class in the format described in RG 1.23, Revision 1. An hour-by-hour listing of the hourly-averaged parameters should be provided in the format described in RG 1.23, Revision 1. If possible, evidence of how well these data represent long-term conditions at the site should also be presented, possibly through comparison with offsite data.

- At least two consecutive annual cycles (and preferably 3 or more whole years), including the most recent 1-year period, should be provided with the application. If 2 years of onsite meteorological data are not available at the time the application is filed, the staff expects that the applicant will provide at least one annual cycle of meteorological data collected on site with the application. These data should be used by the applicant to calculate: (1) the short-term atmospheric dispersion estimates for accident releases discussed in NUREG-0800, Section 2.3.4; and (2) the long-term atmospheric dispersion estimates for routine releases discussed in NUREG-0800, Section 2.3.5. The applicant should continue to monitor the data and submit the complete 2-year dataset when it has collected all the data. This supplemental submittal should also include a reanalysis of the FSAR Section 2.3.4 and 2.3.5 atmospheric dispersion estimates based on the complete 2-year dataset.

- The applicant should identify and justify any deviations from the guidance provided in RG 1.23, Revision 1.

The following regulatory guides and other related guidance documents were also taken into consideration in the staff's review of WLS COL FSAR Section 2.3.3, as applicable.

- RG 1.206, "Combined License Applications for Nuclear Power Plants," which essentially reiterates the types of information, identified in NUREG-0800, Section 2.3.3, that an applicant should provide in FSAR Section 2.3.3 when describing the pre-operational and operational phase onsite meteorological monitoring programs.

- NUREG-0917 (July 1982), "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data," which includes a series of computer programs used to examine the quality and validity of an applicant's hourly meteorological data. The staff subsequently adapted these routines to run on a desktop computer spreadsheet program.

2.3.3.4 *Technical Evaluation*

The staff relied upon the review guidance in NUREG-0800, Section 2.3.3 and the regulatory guidance referred to in the preceding subsection to independently assess the technical sufficiency of the information presented by the applicant.

The staff based their evaluation on information obtained during a pre-application readiness assessment held on May 9-10, 2007.[20] The purpose of the readiness assessment was to: (1) become familiar with the then-prospective applicant's site and site selection process, plans, schedules, and initiatives; (2) observe and review the pre-construction (pre--operational) phase onsite meteorological monitoring program; and (3) review the applicant's plans for the operational phase of its onsite monitoring program.

The staff reviewed WLS COL FSAR Section 2.3.3 (including WLS COL 2.3-3) to ensure that the discussion represents the necessary scope of information related to the onsite meteorological measurements programs. The topics related to the siting and instrumentation associated with the onsite meteorological monitoring programs, handling of the collected data (i.e., acquisition, processing, and validation), and the inspection and maintenance of its systems are, for the most part, organized in the same sequence as they were presented in WLS COL FSAR Section 2.3.3. However, section numbering is consistent with the organization of this report. Likewise, section titles are generally similar to those in WLS COL FSAR Section 2.3.3.

2.3.3.4.1 Pre-Operational Meteorological Measurements Program

In WLS COL FSAR Section 2.3.3.1, Revision 0, the applicant stated that there are two meteorological towers at the site. An older meteorological tower (referred to as Tower 1) was part of the original Cherokee Nuclear site. The applicant indicated that Tower 1 did not meet the structural requirements of RG 1.23, Revision 1, and consequently the applicant decided not to use any of that monitoring data to support preparation of the application for WLS Units 1 and 2. The applicant subsequently indicated, beginning in Revision 4 of the application, that Tower 1 was decommissioned in May 2011.

The second meteorological tower (referred to as the primary tower, or Tower 2) was installed for the pre-construction (i.e., pre-operational) measurement phase and began data collection in December 2005 to support the COL application. The second meteorological tower is a 60 m (200 ft) tower with an elevation at the base of about 186 m (611 ft) above MSL, or about 5.5 m

[20] William S. Lee Pre-Application Site Visit: Review of the Pre-Operational Onsite Meteorological Monitoring Program.

(18 ft) higher than the expected final plant grade of 181 m (593 ft) above MSL. The staff agrees with the applicant that Tower 2 is sufficiently removed from any existing structures or significant topographic features, based on the following criteria:

- the staff's pre-application site visit in May 2007
- information depicted in FSAR Figure 2.3-247 (an aerial photograph adapted to illustrate the general locations of current pre-application structures and Towers 1 and 2)
- the near-field topographic map in FSAR Figure 2.3-245 (Sheet 2 of 2)

Consistent with the guidance in RG 1.23, Revision 1, the staff agrees with the applicant that Tower 2 is sufficiently removed from any existing structures or significant topographic features.

The original application submittal in December 2007 included 1 year of onsite meteorological data from Tower 2 covering the period from December 1, 2005, through November 30, 2006. These data were used in WLS COL FSAR Section 2.3.2 to describe local meteorological conditions (i.e., winds, temperature, humidity, and rainfall) and in WLS COL FSAR Sections 2.3.4 and 2.3.5 for estimating short-term and long-term atmospheric dispersion characteristics. However, the guidance in NUREG-0800, Section 2.3.3, and RG 1.23 recommends a minimum of two consecutive annual cycles (i.e., a 24-month period) of onsite meteorological data to be provided at the time of application for a combined license. Therefore, in RAI 448, Question 02.03.03-3, the staff requested that the applicant address this issue.

In a December 17, 2008, response to RAI 448, Question 02.03.03-3, the applicant stated that the evaluation and submittal of the 2-year onsite meteorological dataset was provided in its November 25, 2008, response to RAI 447, Question 02.03.02-1, which included a then-new Appendix 2CC. This appendix presented comparisons of onsite temperature, moisture, atmospheric stability class, precipitation, and winds between the 1- and 2-year datasets and both onsite monitoring periods against longer-term temperature, moisture, and rainfall summaries for the Greenville / Spartanburg, SC, NWS station. Except for the data recovery, Section 2.3.2.4 of this report addresses the staff's evaluation of the response to RAI 447, Question 02.03.02-1.

The applicant's response to RAI 448, Question 02.03.03-3 further supplemented this new Appendix 2CC by comparing or revising estimated short- and long-term atmospheric dispersion factors based on both the 1- and 2-year datasets. The staff's evaluations of these comparisons are provided in Sections 2.3.4.4 and 2.3.5.4 of this report, respectively.

WLS COL FSAR Section 2.3.3.1 concludes by reporting meteorological data recovery statistics for the first year of onsite monitoring at Tower 2 (i.e., for December 2005 through November 2006). Based on the data evaluation, the applicant determined a rate of 96.5 percent for the joint recovery of wind direction, wind speed, and delta-temperature (change in temperature), using the data screening criteria in NUREG-0917, and a rate of 99.2 percent for the same period (presumably for the joint recovery of these parameters) prior to applying this additional screening.

As a follow-up to RAI 448, Question 02.03.03-3 and based on NUREG-0800, Section 2.3.3 and RG 1.23, Revision 1, Regulatory Position C.5, in RAI 6357, Question 02.03.03-5, the staff requested that the applicant provide information demonstrating that 90 percent data recovery was achieved during each annual cycle of the 2-year POR and, because a normalized 2-year dataset had been input into the dispersion modeling, for the composite 2-year period.

- for the joint recovery of all meteorological variables used to model atmospheric dispersion (i.e., the joint frequency of wind speed, wind direction, and atmospheric stability class)
- individually for all measured parameters

In a May 2, 2012, response to RAI 6357, Question 02.03.03-5, the applicant provided a table listing the data recoveries for individual meteorological parameters for the requested times without having applied the data screening criteria in NUREG-0917. While not summarized in the table, the applicant reiterated the 96.5 percent joint recovery rate for the dispersion model input data during the first year of onsite monitoring (December 2005 through November 2006). The applicant also stated the joint recovery rates for the second year of monitoring (December 2006 through November 2007) and the composite 2-year POR as 95.7 and 96.1 percent, respectively, noting also that the NUREG-0917 screening criteria had been applied to these parameters, including the recovery rate for the first yearly record. Data recoveries for the pre-operational onsite meteorological monitoring program meet or exceed the 90-percent objective in RG 1.23, Regulatory Position C. 5, Revision 1. Therefore, the staff considers these data acceptable for use in the summaries and analyses under WLS COL FSAR Sections 2.3.2, 2.3.4, and 2.3.5. Accordingly, the staff considers RAI 448, Question 02.03.03-3 and RAI 6357, Question 02.03.03-5 resolved.

After completion of the pre-construction (pre-operational) monitoring phase, the applicant plans to terminate the meteorological measurement program at Tower 2. Information regarding the monitoring program for the operational phase is discussed in Section 2.3.3.4.2 of this report.

2.3.3.4.1.1 *Tower Configuration, Instrument Descriptions, and Siting*

Tower 2 has two instrumented levels, 10 and 60 m (32 and 200 ft) above ground level, at which ambient dry-bulb temperature, wind speed, and wind direction are measured. The vertical temperature gradient (or delta-temperature), an indicator of atmospheric stability, is calculated based on the difference between the 60-m and 10-m (32 and 200 ft) level temperature measurements. Dewpoint temperature is also measured at the 10-m (32 ft) level. Located near Tower 2, station pressure and ambient temperature measurements are taken at 2 m (7 ft) above ground level; precipitation, incoming solar and outgoing longwave radiation are measured at a height of 1 m (3 ft). Delta-temperature was also determined between the 10 m and 2 m (32 ft and 7 ft) temperature measurement levels but not used in any calculations that rely on atmospheric stability.

The accuracy and (presumed) precision of each instrument was initially provided in WLS COL FSAR Table 2.3-281. In RAI 538, Question 07.05-1, the staff requested, in part, that the applicant define the monitored meteorological parameters in compliance with RG 1.97, Revision 3 as had been committed to in the application. In an October 1, 2009, response, the applicant stated that meteorological monitoring at the Lee Nuclear Station conformed to RG 1.97, Revision 4 per WLS COL FSAR Appendix 1AA and that the meteorological variables are defined consistent with RG 1.97, Revision 3. As part of its October 1, 2009, response to RAI 538, Question 07.05-1, the applicant updated WLS COL FSAR Table 2.3-281 by:

- identifying the measurement range of each instrument
- clarifying the accuracy specifications and measurement resolution (previously referred to as “precision”) of each instrument
- identifying the regulatory or technical basis for all specifications
- adding two other monitoring system variables (i.e., time and data sampling rate)

The applicant stated that all instrumentation and measurements associated with Tower 2 meet the guidance provided in RG 1.23, Revision 1. However, the staff noted that WLS COL FSAR Table 2.3-281 identifies station pressure, incoming solar radiation, and outgoing longwave radiation as optional measurement variables for which the accuracy guidance in ANS/ANSI 3.11-2005 has been adopted as the “state-of-the-art specification” in the absence of such criteria in RG 1.23. The staff finds the applicant’s instrument complement and corresponding specifications for the pre-operational phase acceptable and, therefore, considers RAI 538, Question 07.05-1 resolved.

RG 1.23 states that wind sensors should be located over level, open terrain at a distance of at least 10 times the height of any nearby obstruction, if the height of the obstruction exceeds one-half the height of the wind measurement. The applicant stated that trees and vegetation were cleared around Tower 2 to ensure an open exposure area to meet this criterion. Further, wind and temperature instruments were mounted on 2.4 m (8 ft) booms extending out from the northwest side of the tower, generally oriented perpendicular to the prevailing (predominant) winds. Temperature sensors were housed in aspirated radiation shields. The applicant stated that this configuration should avoid measurement interference that could be caused by the tower itself.

Consequently, based on the applicant’s description of the meteorological equipment installed on Tower 2 and under WLS COL FSAR Section 2.3.3.1, and considering the staff’s pre-application site visit, the staff finds the siting of meteorological Tower 2 and the deployment of its instrumentation acceptable.

2.3.3.4.1.2 *Data Acquisition*

WLS COL FSAR Section 2.3.3.2.1 summarizes the data flow as follows. Electronic signals from the instrument sensors located on or near the meteorological tower are sent to signal conditioning equipment in a co-located instrument shelter/building and from there to a data logger. The data logger then stores the measured data and performs some additional processing. The system also includes provisions for remote access. The data are downloaded from the data logger by a dedicated computer (“central PC”) at the Duke Energy Environmental Center for validation, reporting, and archiving. The onsite meteorological data are then recorded in digital form.

The data acquisition system was one of the topics discussed during the staff's pre-application site visit. The equipment and techniques employed by the applicant meet the recommendations provided in RG 1.23, Regulatory Position C.6, Revision 1.

2.3.3.4.1.3 *Data Processing*

WLS COL FSAR Section 2.3.3.2.2 summarizes the automated processing of the meteorological measurement data from Tower 2 as follows. The data logger, at a minimum of once per second, samples data channels; output from each sensor is scaled to the appropriate units of measure; then 1-minute and 1-hour averages or totals are calculated, recorded, and assigned a time stamp for each calculation. Among the measured parameters:

- wind speed and wind direction are determined as scalar values
- precipitation amounts represent total accumulation
- incoming solar radiation is the total incident shortwave radiation
- outgoing longwave terrestrial radiation represents total upwelling infrared radiation from the ground

The data logger also checks the quality of the data and assigns a quality flag as needed.

RG 1.23, Regulatory Position C.6, Revision 1 states that digital data should be: (1) compiled as 15-minute average values for real-time display in the appropriate emergency response facilities (e.g., the Control Room, Technical Support Center, and Emergency Operations Facility); and (2) compiled and archived as hourly values for use in historical climatic and dispersion analyses. While the first of these two items pertains specifically to the operational phase of the monitoring program, in RAI 448, Question 02.03.03-1, the staff requested that the applicant address these topics in WLS COL FSAR Section 2.3.3.2.2.

The applicant provided the requested information in Revision 1 of the COL application, but only with respect to the first item. In a May 2, 2013, response to RAI 448, Question 02.03.03-1, the applicant correctly pointed out that WLS COL FSAR Sections 2.3.3.2.2 and 2.3.3.2.3 already addressed the compilation and archival of hourly data values. The staff agrees with that portion of the applicant's response as well as the revisions made. Data processing was one of the topics discussed during the staff's pre-application site visit. The staff finds that the equipment and techniques employed meet the recommendations provided in RG 1.23, Regulatory Position C.6, Revision 1. Accordingly, the staff considers RAI 448, Question 02.03.03-1 resolved.

2.3.3.4.1.4 *Data Validation*

WLS COL FSAR Section 2.3.3.2.3 briefly summarizes the data review and validation process as follows:

- daily data are reviewed to detect system problems and to perform preliminary data verifications

- onsite system checks are conducted by field staff at least monthly to verify proper operation of the systems

- site technicians review all of the meteorological data collected for the previous month after completion of the system checks

In-house personnel, including a staff meteorologist, also review the data. Following these reviews, the database is edited, as necessary, on the central computer. Both raw (unedited) and quality assured (edited) data are archived, along with backup copies, on the central computer.

The data review process was among the topics discussed during the staff's pre-application site visit. The staff finds the applicant's general approach to data review and validation consistent with the intent of achieving a valid data recovery rate of at least 90 percent on an annual basis as stated in RG 1.23, Regulatory Position C.5, Revision 1.

2.3.3.4.1.5 *Inspection and Maintenance of Meteorological Instrumentation*

WLS COL FSAR Section 2.3.3.3 states that meteorological equipment (i.e., sensors, recorders, electronics, data logger, etc.) are inspected and serviced at a frequency designed to assure at least 90 percent data recovery, as specified in RG 1.23, Revision 1, and to minimize extended periods of instrument outage. This process includes the performance of field checks, field calibrations, or replacement by laboratory-calibrated components. Equipment is either calibrated or replaced after at least every 6 months of service. A reserve of spare parts and equipment is maintained to minimize periods of instrument outage. Administrative controls such as maintenance procedures are used to calibrate and maintain meteorological and station equipment. Records documenting the results of calibrations, major causes of instrument outages or drift from calibration, and corrective action taken are maintained. The meteorological instrumentation inspection, maintenance, and calibration processes summarized by the applicant meet the intent of RG 1.23, Regulatory Position C.5, Revision 1, for achieving acceptable data recovery and are, therefore, acceptable to the staff.

2.3.3.4.2 Operational Meteorological Measurement Program

The applicant's initial description of the onsite meteorological measurement program for the operational phase is somewhat limited. WLS COL FSAR Section 2.3.3 and Section 2.3.3.1 briefly indicated that the applicant plans to end the pre-operational monitoring program at Tower 2 and install a new permanent meteorological tower at a different location on the plant site to support operation of WLS Units 1 and 2. WLS COL FSAR Section 2.3.3.3 states that monitoring is continuous during the operational phase beginning with initial fuel loading and continuing through the life of the plant. The applicant further states that the meteorological program "has been developed to be consistent with the guidance given in RG 1.23, Revision 1." Its basic objective is "to maintain data collection performance to assure at least 90 percent joint recoverability and availability of data needed for assessing the relative concentrations and doses resulting from accidental or routine releases."

In RAI 448, Question 02.03.03-2, the staff requested that the applicant address the following aspects of the operational phase of the onsite meteorological measurement program:

- the siting of Tower 3, including tower elevation, and the representativeness of the location and resultant data
- a comparison of the meteorological data between Tower 2 and Tower 3 for data consistency
- proposed or existing nearby obstructions, including distance from the tower (such as the containment building, cooling towers, trees, nearby terrain, etc.), and the potential impact on the accuracy and representativeness of the measurements
- the instrument maintenance and servicing schedules and the planned data reduction and compilation procedures for the operational meteorological program (including data acquisition, processing, and validation)
- how data from the operational program will be used to support emergency preparedness procedures, compiled and archived for later use

In a December 17, 2008, response to RAI 448, Question 02.03.03-2, the applicant stated that the data collected from Tower 2 would continue to serve as the licensing basis for the application and that Tower 3 was not erected with the intent of providing data in support of the application. The applicant also stated in the response that monitoring at Tower 3 was not yet operational and that a full year of data would not be collected until the end of 2009. The applicant also indicated its expectation to transition from Tower 2 to Tower 3 as the recognized source of onsite meteorological data following receipt of a combined license and prior to receipt of fuel.

The applicant stated that Tower 3 will comply with the guidance in RG 1.23, Revision 1, and that it will use the same procedures and maintenance / servicing schedules as Tower 2 that are already described in WLS COL FSAR Section 2.3.3.3. The applicant also stated that the compilation and archiving of meteorological data are addressed in WLS COL FSAR Section 2.3.3.2 and that the use of meteorological data in emergency planning is described in Lee Nuclear Station Emergency Plan, Section II.H.8 and Appendix 2. This addresses to the fourth and fifth items of RAI 448, Question 02.03.03-2.

NUREG-0800, Section 2.3.3 provides guidance for the staff to make two findings through its evaluation of the onsite meteorological monitoring program. The first is how the program relates to determining the acceptability of the site (based on the data measured at the tower site and as input for required atmospheric dispersion modeling analyses), and the second is the monitoring program's adequacy to support facility operations. Regarding the latter, NUREG-0800, Section 2.3.3, Subsection IV states that the staff should verify that the applicant has provided sufficient information to conclude that the equipment that measures meteorological parameters during the course of accidents is sufficient to provide reasonable prediction of atmospheric dispersion of airborne materials in accordance with 10 CFR Part 50, Appendix E. The staff considers RAI 448, Question 02.03.03-2 resolved.

In follow-up RAI 6357, Question 02.03.03-4, the staff requested that the applicant provide information that would allow the staff to complete its evaluation of the adequacy and acceptability of the onsite meteorological measurement program for the operational phase pursuant to:

- 10 CFR 50.47(b)(8) and (b)(9) via 10 CFR 52.79(a)(21), as it relates to emergency planning
- RG 1.206, Section C.IV.4.2, as it relates to the treatment of operational programs in the applications
- 10 CFR 52.79(a)(41), as it relates to the staff's evaluation against relevant NUREG-0800 acceptance criteria

The staff requested that the applicant update WLS COL FSAR Section 2.3.3 by either completing the descriptions of the operational monitoring program or justifying why they did not need this information. Consistent with guidance in NUREG-0800, Section 2.3.3, Subsection II and RG 1.23, Revision 1, the staff requested that the applicant provide or specify the following:

- a site map (to scale) showing the location of Tower 3 relative to any existing and planned structures for the proposed facility and topographic and other features that may influence site meteorological measurements
- a corresponding list of distances from Tower 3 to nearby obstructions to air flow by direction sector
- an update to WLS COL FSAR Table 2.3-281 or a separate listing of the performance specifications for the meteorological instrumentation to be installed on Tower 3
- the elevation at the base of Tower 3 relative to finished grade elevations at existing (if applicable) and planned structures (e.g., cooling towers, buildings from which accident or routine radioactive releases to the atmosphere may occur, trees or other vegetation, topographic features)
- the orientation and length of the various instrument booms at all tower levels, the type of tower construction, and longest horizontal dimension (side) of the tower
- the characteristics of the surface underlying the meteorological tower

In a May 2, 2012, response to RAI 6357, Question 02.03.03-4, the applicant included changes to WLS COL FSAR Section 2.3.3.1 that designate Tower 3 as the permanent meteorological (MET) tower, identify the tower's relative location on the plant site, and state that it is sufficiently removed from the influence of permanent plant structures and topographical features (supported by an updated version of WLS COL FSAR Figure 1.1-202). The applicant also indicated the elevation of the tower at its base (i.e., 181.5 m (595.5 ft) above MSL), which is about 1.5 m (5 ft) above expected plant grade.

Further, these changes reiterate that the instrumentation associated with the Permanent MET Tower meets the guidance provided in RG 1.23, Revision 1, consolidate information about the sensors installed on all towers in an update to WLS COL FSAR Table 2.3-281, and indicate that nearby trees and vegetation were also cleared around Tower 3 to ensure an open exposure. The staff notes that updates to WLS COL FSAR Section 2.3.3.2.2 clarify the processing and storage of 1-minute and 1-hour data averages by the data logger.

Part of the applicant's May 2, 2012, response to RAI 6357, Question 02.03.03-4 included a separate listing of distances between potential obstructions to airflow and the Permanent MET Tower, along with two separate figures to illustrate, in more detail, their relative orientations to and distances from the tower. The staff finds that this information, although not incorporated into the WLS COL FSAR but retained as part of the response to RAI 6357, Question 02.03.03-4, demonstrates that the separation criterion from such obstructions in RG 1.23 is satisfied. WLS COL FSAR Figure 1.1-202, "Site Layout," provides a detailed representation of the location of Tower 3 in relation to other plant structures.

In addition, the applicant's response to RAI 6357, Question 02.03.03-4 indicated that the Permanent MET Tower is an open-lattice structure installed above a natural, grassy surface. The "tower" (instrument) booms are 2.4 m (8 ft) in length and oriented to 300 degrees relative to True North and that the sensor cross-arm assemblies are more than two tower widths away from the tower. This information demonstrates that potential influences of the tower and underlying ground surface should have negligible effects on wind and temperature measurements made at the 10-m and 60-m (32 and 197 ft) levels of the tower.

Given the applicant's statements that the onsite meteorological measurements program during the operational phase will be the same as the pre-construction (pre-operational) monitoring program (in terms of instrument complement and specifications; data acquisition, processing, and validation; and implementation of inspection and maintenance procedures) and given the applicant's responses to RAI 448, Question 02.03.03-2 and RAI 6357, Question 02.03.03-4, the staff expects the operational meteorological monitoring program to be established and operated in conformance to RG 1.23, Regulatory Positions C.2 through C.6, and C.8, Revision 1. Therefore, the staff considers this acceptable. Accordingly, the staff considers RAI 448, Question 02.03.03-2 and RAI 6357, Question 02.03.03-4 resolved.

2.3.3.5 *Post Combined License Activities*

Part 10 of the WLS COL application describes COL conditions, including inspections, tests, analyses, and acceptance criteria (ITAAC). Table 3.8-1 in Part 10 of the WLS COL application contains the emergency planning (EP) ITAAC. The following two EP ITAAC involve demonstrating that the operational onsite meteorological monitoring program appropriately supports the emergency plan for WLS Units 1 and 2:

- EP-ITAAC 6.3: The means exist to continuously assess the impact of the release of radioactive materials to the environment, accounting for the relationship between effluent monitor readings, and onsite and offsite exposures and contamination for various meteorological conditions. An analysis of emergency plan implementing procedures will be performed to demonstrate that the corresponding acceptance criterion has been—that is, that a methodology has been provided to establish the relationship between effluent monitor readings and onsite and offsite exposures and contamination for various radiological conditions.

- EP-ITAAC 6.4: The means exist to acquire and evaluate meteorological information. An inspection of the control room, Technical Support Center (TSC), and Emergency Operations Facility will be performed to demonstrate that the corresponding acceptance criterion has been met, that is, that the specified meteorological data (i.e., wind speed, wind direction, and air temperature at the 10-m and 60-m (32 ft and 197 ft) levels) were available at these facilities.

EP, including EP ITAAC is addressed in Section 13.3, "Emergency Planning" of this report.

2.3.3.6 *Conclusion*

The staff reviewed the application, including the information provided as WLS COL 2.3-3, evaluated the applicant's responses to several RAI questions issued as a result of that review, and checked the referenced AP1000 DCD. The staff's review confirmed that the applicant has addressed the required information relating to the onsite meteorological measurements programs for the pre-operational and operational phases. The staff notes there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section.

COL Information Item 2.3-3 states that a COL applicant shall address the site-specific onsite meteorological measurements program. As set forth above, the applicant has presented and substantiated information pertaining to the onsite meteorological measurements program and the resulting database. The staff reviewed the information provided in WLS COL 2.3-3. The staff concludes that the applicant has established consideration of the onsite meteorological measurements program, and the resulting databases are acceptable and meet the requirements of 10 CFR 100.20 with respect to determining the acceptability of the site. The staff also finds that the onsite data also provide an acceptable basis for making estimates of atmospheric dispersion for DBA and routine releases from the plant to meet the requirements of 10 CFR 100.21, GDC 19, 10 CFR Part 20, and 10 CFR Part 50, Appendix I. Finally, the equipment provided for measurement of meteorological parameters during the course of accidents is sufficient to provide reasonable prediction of atmospheric dispersion of airborne radioactive materials in accordance with 10 CFR Part 50, Appendix E. Part 5, "Emergency plan" of the WLS COL application identifies alternative offsite sources of meteorological data during an emergency. The staff finds that the applicant has provided a sufficient description to adequately address COL Information Item 2.3-3.

2.3.4 Short-Term Diffusion Estimates

2.3.4.1 *Introduction*

The short-term diffusion estimates are used to determine the amount of airborne radioactive materials expected to reach a specific location during an accident situation. The diffusion estimates address the requirement for conservative atmospheric dispersion (relative concentration) factor (χ/Q value) estimates at the EAB, the outer boundary of the LPZ, and at the control room for postulated design-basis accidental radioactive airborne releases. The review covers the following specific areas: (1) atmospheric dispersion models to calculate atmospheric dispersion factors for postulated accidental radioactive releases; (2) meteorological data and other assumptions used as input to atmospheric dispersion models; (3) derivation of diffusion parameters (e.g., σ_y and σ_z); (4) cumulative frequency distributions of χ/Q values; (5) determination of conservative χ/Q values used to assess the consequences of postulated design-basis atmospheric radioactive releases to the EAB, LPZ, and control room; and (6) any additional information requirements prescribed in the "Contents of application" sections of the applicable subparts to 10 CFR Part 52, "Licenses, certifications, and approvals for nuclear power plants." This section of the report also addresses the supplemental information in WLS COL FSAR Section 2.3.6 related to the short-term diffusion estimates.

2.3.4.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 2.3, incorporates by reference AP1000 DCD, Section 2.3.

In addition, in WLS COL FSAR Section 2.3, the applicant addressed the following:

Departures

- WLS DEP 18.8-1

The applicant provided additional information in WLS COL 2.3-4 to address COL Information Item 2.3-4. WLS COL 2.3-4 addresses the provision of site-specific short-term diffusion estimates for staff review to ensure that the bounding values (AP1000 DCD, Appendix 15A, Table 2-1) of relative concentrations are not exceeded. In addition, this WLS COL FSAR section addresses Interface Item 2.4 related to the limiting meteorological parameters (χ/Q) for DBAs.

AP1000 COL Information Item

- WLS COL 2.3-4

The applicant provided additional information in WLS COL 2.3-4 to address COL Information Item 2.3-4 (COL Action Item 2.3.4-1). WLS COL 2.3-4 addresses the provision of site-specific short-term diffusion estimates for NRC review to ensure that the bounding values (Table 2-1 and Appendix 15A from the AP1000 DCD) of relative concentrations are not exceeded.

In addition, this WLS COL FSAR section addresses Interface Item 2.4 related to the limiting meteorological parameters (χ/Q) for DBAs.

2.3.4.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. The applicable regulatory requirements for the applicant's description of atmospheric diffusion estimates for accidental releases are as follows:

- 10 CFR Part 50, Appendix A, GDC 19, as it relates to the meteorological considerations used to evaluate the personnel exposures inside the control room during radiological and airborne hazardous material accident conditions.

- 10 CFR 52.79(a)(1)(vi), as it relates to a safety assessment of the site, including consideration of major SSCs of the facility and site meteorology, to evaluate the offsite radiological consequences at the EAB and LPZ.

- 10 CFR 100.21(c)(2), as it relates to the atmospheric dispersion characteristics used in the evaluation of the EAB and LPZ radiological dose consequences for postulated accidents.

NUREG-0800, Section 2.3.4 recommends the following acceptance criteria to be provided by a COL applicant:

- A description of the atmospheric dispersion models used to calculate χ/Q values for accidental releases of radioactive and hazardous materials to the atmosphere. The models should be documented in detail and substantiated within the limits of the model so that the staff can evaluate their appropriateness of use with regard to release characteristics, plant configuration, plume density, meteorological conditions, and site topography.

- Meteorological data used for the evaluation (as input to the dispersion models), which represent annual cycles of hourly values of wind direction, wind speed, and atmospheric stability for each mode of accidental release. Any dispersion estimates should be calculated from the most representative meteorological data available for the site.

- A discussion of atmospheric diffusion parameters, such as lateral and vertical plume spread (σ_y and σ_z) as a function of distance, topography, and atmospheric conditions should be related to measured meteorological data. The methodology for establishing these relationships should be appropriate for estimating the consequences of accidents within the range of distances which are of interest with respect to site characteristics and established regulatory criteria.

- Hourly cumulative frequency distributions of χ/Q values from the effluent release point(s) to the EAB and outer boundary of the LPZ should be constructed to describe the probabilities of these χ/Q values being exceeded. All cumulative frequency distributions of χ/Q values should be presented for appropriate distances (as indicated above) and time periods as specified in RG 1.206, Part I, Subsection C.I.2.3.4.2. The methods for generating these distributions should be adequately described.

- Atmospheric dispersion factors used for the assessment of consequences related to atmospheric radioactive release to the control room for design-basis, other accidents, and onsite and off-site releases of hazardous airborne materials should be provided.

- For control room habitability analysis, a site plan drawn to scale should be included showing true North and potential atmospheric accident release pathways, control room air intake(s), and unfiltered in-leakage pathways.

The following regulatory guides and other related guidance documents were also taken into consideration in the staff's review of WLS COL FSAR Section 2.3.4, as applicable:

- RG 1.23, “Meteorological Monitoring Programs for Nuclear Power Plants” Revision 1, which includes guidance on the measurement and processing of onsite meteorological data for use as input to atmospheric dispersion models in support of plant licensing.

- RG-1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants”, Revision 1, which provides guidance on appropriate dispersion models for estimating offsite relative air concentrations (χ/Q values) as a function of downwind direction and distance (i.e., at the EAB and outer boundary of the LPZ) for various short-term time periods (up to 30 days) after an accident, provisions and guidance to account for ground-level and elevated releases, meteorological conditions, and modified plume dispersion due to building wake effects, plume meander under low wind speed conditions, non-straight plume trajectories, and fumigation conditions at coastal and inland site locations.

- RG-1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants”, which discusses acceptable approaches for estimating short-term (i.e., 2 hours to 30 days post-accident) average χ/Q values in the vicinity of buildings at control room ventilation air intakes and at other locations of significant air in-leakage to the control room envelope due to postulated design-basis accidental radiological airborne releases, provisions and guidance for determining release point characteristics, receptors, source-receptor distances and directions, and meteorological input data.

- RG 1.206, “Combined License Applications for Nuclear Power Plants”, which summarizes the types of information, identified in NUREG-0800, Section 2.3.4, that an applicant should provide in FSAR Section 2.3.4 for estimating dispersion factors (χ/Q values) used to assess the consequences of design-basis and other accidental atmospheric radiological releases on control room habitability.

- NUREG/CR-2858 (November 1982), “PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations”, prepared by Pacific Northwest Laboratory (PNL-4413), the user’s manual for the NRC-sponsored PAVAN dispersion model.

- NUREG/CR-6331 (Revision 1, May 1997), “Atmospheric Relative Concentrations in Building Wakes”, prepared by Pacific Northwest National Laboratory (PNNL-10521), the user’s manual for the NRC-sponsored ARCON96 dispersion model.

2.3.4.4 *Technical Evaluation*

The staff reviewed WLS COL FSAR Section 2.3.4 and checked the applicable site parameters in AP1000 DCD Tier 2, Table 2-1, to ensure that the combination of the DCD and the information in the COL application represent the complete scope of information relating to this review topic. The staff’s review confirmed that the information contained in the application and incorporated by reference addresses the required information relating to the short-term dispersion estimates. AP1000 DCD, Section 2.3.4 was reviewed by the staff under Docket

Number 52-006. The results of the staff's evaluation of the information incorporated by reference in the application are documented in NUREG-1793 and its supplements.

The staff reviewed the information contained in the WLS COL FSAR corresponding to the following:

AP1000 COL Information Item

- WLS COL 2.3-4

The staff reviewed information related to the short-term diffusion estimates of accidental releases to the atmosphere included by the applicant under WLS COL FSAR Section 2.3.4 in response to COL Information Item 2.3-4. The specific text of this COL Information Item in AP1000 DCD, Section 2.3.6.4 states:

> Combined License applicants referencing the AP1000 certified design will address the site-specific χ/Q values specified in subsection 2.3.4. For a site selected that exceeds the bounding χ/Q values, the Combined License applicant will address how the radiological consequences associated with the controlling design basis accident continue to meet the dose reference values given in 10 CFR Part 50.34 and control room operator dose limits given in General Design Criteria 19 using site-specific χ/Q values. The Combined License applicant should consider topographical characteristics in the vicinity of the site for restrictions of horizontal and/or vertical plume spread, channeling or other changes in airflow trajectories, and other unusual conditions affecting atmospheric transport and diffusion between the source and receptors.
> No further action is required for sites within the bounds of the site parameters for atmospheric dispersion.
>
> With regard to assessment of the postulated impact of an accident on the environment, the COL applicant will provide χ/Q values for each cumulative frequency distribution that exceeds the median value (50 percent of the time).

Regarding the latter portion of WLS COL 2.3-4, the staff notes that the postulated impact of an accident on the environment is addressed in the Environmental Report. The staff relied upon the review procedures presented in NUREG-0800, Section 2.3.4, to independently assess the technical sufficiency of the information presented by the applicant.

2.3.4.4.1 Atmospheric Dispersion Models

2.3.4.4.1.1 *Offsite Dispersion Estimates*

The applicant used the PAVAN computer code (NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design-Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations") to estimate ground-level χ/Q values at the EAB and at the outer boundary of the LPZ for potential accidental releases of radioactive material to the atmosphere. The PAVAN model implements the methods outlined in RG 1.145.

The PAVAN code estimates χ/Q values for various time-average periods ranging from 2 hours to 30 days. The meteorological input to PAVAN consists of a JFD of hourly values of wind speed and wind direction by atmospheric stability class. The χ/Q values calculated through PAVAN are based on the theoretical assumption that material released to the atmosphere will be normally distributed (Gaussian) about the plume centerline. A straight-line trajectory is assumed between the point of release and all downwind receptor distances for which χ/Q values are calculated.

For each of the 16 downwind direction sectors (true North, North-Northeast, Northeast, East-Northeast, etc.), PAVAN calculates χ/Q values for each combination of wind speed and atmospheric stability at the appropriate downwind distance (i.e., the EAB and the outer boundary of the LPZ). The χ/Q values calculated for each sector are then ordered from greatest to smallest and an associated cumulative frequency distribution is derived based on the frequency distribution of wind speed and stabilities for each sector. The smallest χ/Q value in a distribution will have a corresponding cumulative frequency equal to the wind direction frequency for that particular sector. PAVAN determines for each sector an upper envelope curve based on the derived data (plotted as χ/Q versus probability of being exceeded), such that no plotted point is above the curve. From this upper envelope, the χ/Q value, which is equaled or exceeded 0.5 percent of the total time, is obtained. The maximum 0.5 percent χ/Q value from the 16 sectors becomes the 0-2 hour “maximum sector χ/Q value.”

Using the same approach, PAVAN also combines all χ/Q values independent of wind direction into a cumulative frequency distribution for the entire site. An upper envelope curve is determined, and the program selects the χ/Q value which is equaled or exceeded 5.0 percent of the total time. This is known as the 0-2 hour “5-percent overall site χ/Q value.”

The larger of the two χ/Q values, either the 0.5-percent maximum sector value or the 5-percent overall site value, is selected to represent the χ/Q value for the 0-2 hour time interval (note that this resulting χ/Q value is based on 1-hour averaged data but is conservatively assumed to apply for 2 hours). An alternative method to determine the χ/Q value for the 0-2 hour time interval is to retain the maximum possible χ/Q value based on the distance, calm wind speeds, and G-stability.

To determine χ/Q values for longer time periods (i.e., 0-8 hour, 8-24 hour, 1-4 days, and 4-30 days), PAVAN performs a logarithmic interpolation between the 0-2 hour χ/Q value and the corresponding annual average (8760-hours) χ/Q value for each of the 16 sectors and overall site. For each time period, the highest among the 16 sectors and overall site χ/Q values are identified and are designated as the short-term site characteristic χ/Q value for that time period.

2.3.4.4.1.2 *Control Room Dispersion Estimates*

The applicant used the computer code ARCON96 (NUREG/CR-6331, “Atmospheric Relative Concentrations in Building Wakes”) to estimate χ/Q values at the control room for potential accidental releases of radioactive material. The ARCON96 model implements the methods outlined in RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants.”

The ARCON96 code estimates χ/Q values for various time-averaged periods ranging from 2 hours to 30 days. The meteorological input to ARCON96 consists of hourly values of wind speed, wind direction, and atmospheric stability class. The χ/Q values calculated through ARCON96 are based on the theoretical assumption that material released to the atmosphere will be normally distributed (Gaussian) about the plume centerline. A straight-line trajectory is assumed between the release points and receptors. The diffusion coefficients account for enhanced dispersion under low wind speed conditions and in building wakes.

The hourly meteorological data are used to calculate hourly relative concentrations. The hourly relative concentrations are then combined to estimate concentrations ranging in duration from 2 hours to 30 days. Cumulative frequency distributions, prepared from the average relative concentrations and the relative concentrations that are exceeded no more than five percent of the time for each averaging period, are determined.

2.3.4.4.2 Meteorological Input Data

2.3.4.4.2.1 *Offsite Dispersion Estimates*

The meteorological data input to the PAVAN dispersion model is in the form of a JFD of wind speed and wind direction by atmospheric stability class. The structure of the JFD meets the intent of the guidance in RG 1.23, Regulatory Position C.6, Revision 1, and the resolution of lower wind speed classes as recommended in the review procedures under NUREG--0800, Section 2.3.3.

As indicated previously, the JFD was prepared based on hourly-averaged onsite measurements for a 2-year POR from December 1, 2005, through November 30, 2007. Wind data were obtained from the 10-m (32 ft) level of the onsite meteorological tower; stability data were derived from the vertical temperature difference between the 60 m and 10 m (197 and 32 m) measurement levels.

With regard to the 1-year of meteorological data, RG 1.23, Revision 1, states that the minimum amount of onsite meteorological data to be provided at the time of application for a combined license is a consecutive 24-month period of data that is defendable, representative, and complete but not older than 10 years from the date of the application. However, the guidance further states that three or more years of data are preferable and, if available, should be submitted with the application. To rectify this deviation from guidance, in RAI 448, Question 02.03.03-3, the staff requested that the applicant address the scope of the meteorological dataset.

In a December 17, 2008, response to RAI 448, Question 02.03.03-3, the applicant included a 2-year meteorological dataset for the period December 1, 2005, through November 30, 2007. Additionally, In WLS COL FSAR, Appendix 2CC, the applicant provided an evaluation of short-term offsite dispersion estimates based on the 2-year dataset with a comparison against the 1-year site characteristic χ/Q values and the corresponding site parameter χ/Q values in the AP1000 DCD. The applicant stated that the site characteristic χ/Q values were consistent for the two datasets.

The staff observed the consistency identified by the applicant but also observed that the 2-year short-term offsite dispersion estimates were generally higher, and therefore the use of the 1-year dataset appeared to be non-conservative when compared to the 2-year dataset. Therefore, in follow-up RAI 3726, Question 02.03.04-4, the staff requested that the applicant resolve the issue. In an April 6, 2010, response to RAI 3726, Question 02.03.04-4, the applicant included an analysis of the 2-year short-term offsite dispersion estimates and updated both text and tables relating to these estimates. The staff reviewed and confirmed the updated text and tables from the RAI and confirmed that these changes were incorporated into the WLS COL FSAR. Accordingly, the staff finds the response to RAI 3726, Question 02.03.04-4 acceptable and considers the RAI resolved. Subsequent revisions to the WLS COL FSAR removed the comparison of 1-year and 2-year of short-term offsite dispersion estimates from WLS COL FSAR Appendix 2CC.

The staff completed a detailed review related to the acceptability and representativeness of the hourly meteorological data as discussed in Sections 2.3.2 and 2.3.3 of this report. Based on this review, the staff considers the onsite meteorological database suitable for input to the PAVAN model.

2.3.4.4.2.2 *Control Room Dispersion Estimates*

The meteorological data input to the ARCON96 dispersion model consisted of hourly-averaged values of wind speed, wind direction, and atmospheric stability class covering a 2-year POR of onsite measurements from December 1, 2005, through November 30, 2007. The wind data were obtained from the 10 m and 60 m (32 and 197 ft) levels of the onsite meteorological tower (the 60 m (197 ft)) wind data were only used by the model when data from the 10 m (32 ft) level were unavailable). The stability data were derived from the vertical temperature difference between the 60 m and 10 m (32 and 197 ft) measurement levels.

The staff completed a detailed review related to the acceptability and representativeness of the hourly meteorological data as discussed in Sections 2.3.2 and 2.3.3 of this report. Based on this review, the staff considers the onsite meteorological database suitable for input to the ARCON96 model.

2.3.4.4.3 Diffusion Parameters

2.3.4.4.3.1 *Offsite Dispersion Estimates*

The applicant chose to implement the diffusion parameter assumptions outlined in RG 1.145, Revision 1, as a function of atmospheric stability and downwind distance from the source, for its PAVAN model runs. Significant local topography can influence short-term dispersion estimates. In RAI 449, Question 02.03.04-1, the staff requested that the applicant discuss the potential effects of the local topography on the atmospheric dispersion estimates. In an October 10, 2008, response to RAI 449, Question 02.03.04-1, the applicant included an update to WLS COL FSAR Section 2.3.4.1. In the response the applicant described the terrain in the immediate vicinity of the site as gently rolling hills and that McKowns Mountain was the only notable terrain feature. Further, the applicant stated that the mountain peak was approximately 61 m (200 ft) above plant grade at a distance of approximately 1.6 km (1 mi). Given the distance and minimal elevation rise, the applicant concluded that McKowns Mountain would have no significant effect on short term diffusion estimates. The staff agrees with the discussion provided in the applicant's response to RAI 449, Question 02.03.04-1 and confirmed that the proposed changes were incorporated into the WLS COL FSAR. Accordingly, the staff considers RAI 449, Question 02.03.04-1 resolved.

The staff evaluated the applicability of the PAVAN diffusion parameters and concluded that no unique topographic features (such as rough terrain, restricted flow conditions, coastal or desert areas) preclude the use of the PAVAN model for the proposed WLS site. Therefore, the staff finds that the applicant's use of diffusion parameter assumptions, as outlined in RG 1.145, Revision 1, acceptable.

2.3.4.4.3.2 *Control Room Dispersion Estimates*

The diffusion coefficients used in ARCON96 have three components. The first component is the diffusion coefficient used in other NRC models such as PAVAN. The other two components are corrections to account for enhanced dispersion under low wind speed conditions and in building wakes. These components are based on analysis of diffusion data collected in various building wake diffusion experiments under a range of meteorological conditions. Since the diffusion occurs at short distances within the plant's building complex, the ARCON96 diffusion parameters are not affected by nearby topographic features such as bodies of water or elevated terrain. Therefore, the staff finds the applicant's use of the ARCON96 diffusion parameter assumptions acceptable.

2.3.4.4.4 Relative Concentrations for Accident Consequences Analyses

2.3.4.4.4.1 *Conservative Short-Term Atmospheric Dispersion Estimates for the EAB and LPZ*

The applicant stated that it modeled one ground-level release point for each unit by assuming a 136 m (448 ft) radius circle, centered on each unit's containment, which encompasses all release points. Further, the applicant stated it conservatively estimated the area of the reactor building to be used for determining wake effects by calculating the above ground, cross-sectional area of the shield building. Shorter EAB and LPZ distances (assuming the 136 m (448 ft) radius circle) resulted in higher (more conservative) χ/Q values. The LPZ distances are defined by a 3.2 km (2 mi) radius circle centered on the midpoint between Units 1 and 2 containment buildings. A ground-level release with the stated assumptions is consistent with RG 1.145, Revision 1 and, therefore, is acceptable to the staff.

In accordance with AP1000 DCD, Section 2.3.6.4, WLS COL FSAR Section 2.3.4.2 compared the site characteristic EAB and LPZ χ/Q values to the corresponding site parameters provided in the DCD. This comparison, provided in WLS COL FSAR Table 2.0-201, showed that the

AP1000 DCD EAB and LPZ χ/Q values conservatively bound the site characteristic χ/Q values.[21]

Using the information provided by the applicant, including the 10 m (32 ft) level joint frequency distributions of wind speed, wind direction, and atmospheric stability discussed in Section 2.3.2 of this report, the staff confirmed the applicant's χ/Q values by running the PAVAN computer code and obtaining consistent results. Therefore, the staff accepts the short-term χ/Q values presented by the applicant.

2.3.4.4.4.2 *Short-Term Atmospheric Dispersion Estimates for the Control Room*

The applicant provided the following as the necessary input to the ARCON96 dispersion modeling analysis:

- Onsite Hourly Meteorological Data: December 1, 2005 - November 30, 2007
- AP1000 DCD Table 15A-7: Control Room Source / Receptor Data
- AP1000 DCD Figure 15A-1: Site Plan with Release and Intake Locations
- WLS COL FSAR Table 2.3-284: Lee Control Room χ/Q Input Data
- WLS COL FSAR Figure 2.1-202: Plant Layout on the Lee Site

In accordance with the AP1000 DCD, two receptor (i.e., air intake) points, the Control Room HVAC Intake and Annex Building Access (control room door), were modeled for the following eight release points:

- Plant Vent
- PCS Air Diffuser
- Fuel Building Blowout Panel
- Radwaste Building Truck Staging Area Door
- Steam Line Break Releases
- Power Operated Relief Valves (PORV) / Safety Valves
- Condenser Air Removal Stack

[21] Smaller χ/Q values are associated with greater dilution capability, resulting in lower radiological doses. When comparing a DCD site parameter χ/Q value and a site characteristic χ/Q value, the site is acceptable for the design if the site characteristic χ/Q value is smaller than the site parameter χ/Q value. Such a comparison shows that the site has better dispersion characteristics than that required by the reactor design.

- Containment Shell PCS Air Diffuser

WLS COL FSAR Table 2.3-285 lists the site-specific control room χ/Q values calculated by the applicant. WLS COL FSAR Table 2.0-202 compared the site-specific control room χ/Q values to the corresponding site parameters provided in the AP1000 DCD. This comparison showed that the AP1000 control room χ/Q values conservatively bounded the site-specific values.

The staff confirmed the applicant's atmospheric dispersion estimates by running the ARCON96 computer model and obtaining similar results. Both the staff and applicant used a ground-level release assumption for each of the release/receptor combinations as well as other conservative assumptions. Based on its confirmatory analysis, the staff finds the applicant's control room χ/Q values acceptable.

2.3.4.4.4.3 *Short-Term Atmospheric Dispersion Estimates for the Technical Support Center*

The applicant provided the following as the necessary input to the ARCON96 dispersion modeling analysis:

- Onsite Hourly Meteorological Data: December 1, 2005 - November 30, 2007
- WLS COL FSAR Table 2.3-294: Lee TSC HVAC Distances and Directions
- WLS COL FSAR Section 2.3.4.4: Control Room Source / Receptor Data

The Technical Support Center (TSC) HVAC Intake was modeled for the following two release points:

- Unit 1 Containment Shell
- Unit 2 Containment Shell

WLS COL FSAR Table 2.3-295 lists the site-specific TSC χ/Q values calculated by the applicant.

The staff confirmed the applicant's atmospheric dispersion estimates by running the ARCON96 computer model and obtaining similar results. Both the staff and applicant used a ground-level release assumption for each of the release/receptor combinations as well as other conservative assumptions. Based on its confirmatory analysis, the staff finds the applicant's TSC χ/Q values acceptable.

The applicant took a departure in WLS COL FSAR Chapter 18, "Human Factors Engineering," in that the WLS TSC is not located in the control building as identified in the AP1000 DCD (WLS DEP 18.8-1). Additional information regarding this departure can be found in the WLS COL application, Part 7 "Departures and Exemptions Requests."

2.3.4.4.5 Onsite and Offsite Hazardous Materials

A review of the applicant's identification of onsite and offsite hazardous materials that could threaten control room habitability is performed in Sections 2.2.1, 2.2.2, and 2.2.3 of this report. The accident scenarios, including release characteristics and atmospheric dispersion model descriptions, model inputs, and assumptions are also discussed in these sections.

2.3.4.5 *Post Combined License Activities*

There are no post COL activities related to this section.

2.3.4.6 *Conclusion*

The staff reviewed the WLS COL application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to short-term diffusion estimates, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

COL Information Item 2.3-4 states that a COL applicant shall address the site-specific χ/Q values as specified in AP1000 DCD, Section 2.3.4. The staff concludes that the applicant's atmospheric dispersion estimates are acceptable and meet the relevant requirements of 10 CFR 100.21(c)(2). This conclusion is based on the conservative assessments of post-accident atmospheric dispersion conditions that have been made by the applicant and the staff from the applicant's meteorological data and appropriate diffusion models. These atmospheric dispersion estimates are appropriate for the assessment of consequences from radioactive releases for DBAs in accordance with 10 CFR 52.79(a)(1)(vi), 10 CFR 100.21(c)(2), and GDC 19. The staff finds that the applicant has provided sufficient information to adequately address COL Information Item 2.3-4.

2.3.5 Long-Term Diffusion Estimates

2.3.5.1 *Introduction*

The long-term diffusion estimates are used to determine the amount of airborne radioactive materials expected to reach a specific location during normal operations. The diffusion estimates address the requirement concerning atmospheric dispersion and dry deposition estimates for routine releases of radiological effluents to the atmosphere. The review covers the following specific areas: (1) atmospheric dispersion and deposition models used to calculate concentrations in air and amount of material deposited as a result of routine releases of radioactive material to the atmosphere; (2) meteorological data and other assumptions used as input to the atmospheric dispersion models; (3) derivation of diffusion parameters (e.g., σ_z); (4) atmospheric dispersion (relative concentration) factors (χ/Q values) and deposition factors (D/Q values) used for assessment of consequences of routine airborne radioactive releases; (5) points of routine release of radioactive material to the atmosphere, the characteristics of each release mode, and the location of potential receptors for dose computations; and (6) any additional information requirements prescribed in the “Contents of Application” sections of the applicable subparts to 10 CFR Part 52. This section of the report also addresses the supplemental information in WLS COL FSAR Section 2.3.6 related to the long-term diffusion estimates.

2.3.5.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 2.3, incorporates by reference AP1000 DCD, Section 2.3. In WLS COL FSAR Section 2.3.5, the applicant addressed the following:

AP1000 COL Information Item

- WLS COL 2.3-5

The additional information provided in WLS COL 2.3-5 is responsive to AP1000 DCD Tier 2, Table 1.8-2, COL Information Item 2.3-5 by addressing long-term (atmospheric) diffusion estimates at and near the site boundary and out to a distance of 80 km (50 mi). In addition, this WLS COL FSAR section addresses Interface Item 2.4 related to the limiting meteorological parameters (χ/Q values) for routine airborne radiological releases.

2.3.5.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. The acceptance criteria for evaluating the analysis of long-term atmospheric dispersion and deposition conditions for routine releases of radiological effluents to the atmosphere during normal plant operation are based on meeting the relevant requirements of 10 CFR Part 20, 10 CFR Part 50, and 10 CFR Part 100. The staff considered the following regulatory requirements in reviewing the applicant’s estimates of atmospheric dispersion and deposition:

- 10 CFR Part 20, Subpart D, as it relates to establishing atmospheric dispersion-related site characteristics for demonstrating compliance with dose limits for individual members of the public;

- 10 CFR 50.34a, “Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors,” and 10 CFR Part 50, Appendix I Sections II.B, II.C, and II.D, as they relate to the numerical guides for design objectives and limiting conditions for operation to meet the requirements that radioactive material in effluents released to unrestricted area be kept ALARA.

- 10 CFR 100.21(c)(1), as it relates to establishing atmospheric dispersion-related site characteristics such that radiological effluent release limits associated with normal operation can be met for any individual located offsite.

The related acceptance criteria from NUREG-0800, Section 2.3.5 are as follows:

- A detailed description of the atmospheric dispersion and deposition models used to calculate annual average concentrations in air and amount of material deposited as a result of routine releases of radioactive materials to the atmosphere.

- A discussion of atmospheric diffusion parameters, such as vertical plume spread (σ_z) as a function of distance, topography, and atmospheric conditions.

- Meteorological data summaries (onsite and regional) used as input to the dispersion and deposition models.

- Points of routine release of radioactive material to the atmosphere, including the characteristics (e.g., location, release mode) of each release point.

- The specific location of potential receptors of interest (e.g., nearest vegetable garden, nearest resident, nearest milk animal, and nearest meat cow in each 22.5 degree direction sector within an 8 km (5 mi) radius of the site).

- The χ/Q and D/Q values to be used for assessment of the consequences of routine airborne radiological releases as described in RG 1.206, Section C.I.2.3.5.2: (1) maximum annual average χ/Q values and D/Q values at or beyond the site boundary and at specific locations of potential receptors of interest using appropriate meteorological data for each routine venting location; and (2) estimates of annual average χ/Q values and D/Q values for 16 radial sectors to a distance of 80 km (50 mi) from the plant using appropriate meteorological data.

The following regulatory guides and other related guidance documents were also taken into consideration in the staff's review of WLS COL FSAR Section 2.3.5, as applicable:

- RG 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants," Revision 1, which includes guidance on the measurement and processing of onsite meteorological data for use as input to atmospheric dispersion models in support of plant licensing and operation

- RG 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, which includes guidance on identifying the location of potential receptors of interest

- RG 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, which discusses different types of atmospheric transport and diffusion models and criteria for characterizing long-term (annual) average atmospheric dispersion and deposition conditions

- RG 1.112, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors," Revision 1, which includes guidance on identifying release point characteristics

- RG 1.206, “Combined License Applications for Nuclear Power Plants,” which summarizes the types of information, identified in NUREG-0800, Section 2.3.5, that an applicant should provide in FSAR Section 2.3.5 regarding the estimation of annual average χ/Q and D/Q values used for annual average release limit calculations and person-rem estimates

- NUREG/CR-2919, “XOQDOQ: Computer Program for the Meteorological Evaluation of Routine Effluent Releases at Nuclear Power Stations,” prepared by Pacific Northwest Laboratory (PNL-4380), the user’s manual for the NRC-sponsored XOQDOQ dispersion model

2.3.5.4 ***Technical Evaluation***

The staff reviewed WLS COLFSAR Section 2.3.5 and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic. The staff’s review confirmed that the information in the application and incorporated by reference addresses the required information relating to the long-term diffusion estimates. The results of the staff’s evaluation of the information incorporated by reference in the application are documented in NUREG-1793 and its supplements. The staff reviewed the information contained in the WLS COL FSAR corresponding to the following:

AP1000 COL Information Item

- WLS COL 2.3-5

The staff reviewed information related to the long-term diffusion estimates of routine releases to the atmosphere included by the applicant in WLS COL FSAR Section 2.3.5 in response to WLS COL 2.3-5. The specific text of this COL information item in AP1000 DCD, Section 2.3.6.5 states:

> Combined License applicants referencing the AP1000 certified design will address long-term diffusion estimates and χ/Q values specified in subsection 2.3.5. The Combined License applicant should consider topographical characteristics in the vicinity of the site for restrictions of horizontal and/or vertical plume spread, channeling or other changes in airflow trajectories, and other unusual conditions affecting atmospheric transport and diffusion between the source and receptors. No further action is required for sites within the bounds of the site parameter for atmospheric dispersion.
>
> With regard to environmental assessment, the COL applicant will also provide estimates of annual average χ/Q values for 16 radial sectors to a distance of 50 mi from the plant.

The staff relied upon the review procedures presented in NUREG-0800, Section 2.3.5, to independently assess the technical sufficiency of the information presented by the applicant.

2.3.5.4.1 Atmospheric Dispersion Model

The applicant used the NRC-sponsored computer code XOQDOQ (described in NUREG/CR-2919, “XOQDOQ Computer Program for the Meteorological Evaluation of Routine Releases at Nuclear Power Stations”) to estimate ground-level χ/Q and D/Q values resulting from routine airborne releases. The XOQDOQ model implements the constant mean wind direction model methods outlined in RG 1.111, “Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors,” Revision 1.

The XOQDOQ model is a straight-line Gaussian plume model based on the theoretical assumption that material released to the atmosphere will be normally distributed (Gaussian) about the plume centerline. In predictions of χ/Q and D/Q values over long time periods (i.e., annual averages), the plume’s horizontal distribution is assumed to be distributed within the downwind direction sector (i.e., “sector averaging”). A straight-line trajectory is assumed between the release point and all downwind receptors.

2.3.5.4.2 Release Characteristics and Receptors

Routine operational releases of radioactive materials to the atmosphere typically occur through vents near the tops of buildings or from tall stacks. In its XOQDOQ dispersion modeling analysis, the applicant assumed that such releases occur at ground level although the exhaust point for the plant vent is 55.7 m (183 ft) above grade based on AP1000 DCD Tier 2, Table 15A-7. The applicant assumed that these releases occurred at ground level from a single point located mid-way between the Units 1 and 2 reactor buildings.

The applicant assumed a minimum building cross-sectional area of 2,843 m^2 (0.70 acres) and a building height of 69.8 m (229 ft), but did not provide a reference for this information. Therefore, in RAI 451, Question 02.03.05-1, the staff requested that the applicant provide a reference for these inputs to the XOQDOQ model. In a December 17, 2008, response to RAI 451, Question 02.03.05-1, the applicant provided the requested references from AP1000 DCD Tier 1 material and Westinghouse drawing APP-1000-P2-902, Revision 0, used to calculate the height and the building cross-sectional area of the Shield Building. The applicant provided a discussion in the response to RAI 451, Question 02.03.05-1 that a sensitivity study was performed to determine the effects of changes to the dimensions of the Shield Building. The applicant stated that the effects were negligible. The staff confirmed the information provided in the response to RAI 451, Question 02.03.05-1 through the documentation provided in the AP1000 DCD. Accordingly, the staff considers RAI 451, Question 02.03.05-1, resolved.

The applicant assumed a ground-level release in modeling routine releases. A ground-level release is a conservative assumption resulting in higher χ/Q and D/Q values when compared to a mixed-mode (i.e., part-time ground-level, part-time elevated) release or a 100 percent elevated release, as discussed in RG 1.111, Revision 1. A ground-level release assumption is therefore acceptable to the staff.

The distance to the receptors of interest (i.e., EAB, milk animal (cow/goat), garden, meat animal, and resident) were presented in WLS COL FSAR Section 2.3.5.2. The distances to each of these receptors has been calculated from a location defined as the mid-point of the two proposed units. This method provides representative results and is acceptable to the staff.

2.3.5.4.3 Meteorological Input Data

The meteorological data input to the XOQDOQ dispersion model consisted of a JFD of wind speed, wind direction, and atmospheric stability class based on hourly onsite data collected during a 2-year period from December 1, 2005, through November 30, 2007. The wind data were obtained from the 10 m (32 ft) level of the onsite meteorological tower, and the stability data were derived from the vertical temperature difference (delta-temperature) measurements taken between the 60 m and 10 m (197 and 32 ft) levels on the onsite meteorological tower.

Based on the applicant’s responses to all RAIs related to the acceptability of the hourly meteorological data as discussed in Section 2.3.3 of this report, the staff considers the December 1, 2005, through November 30, 2007, onsite meteorological database suitable for input to the XOQDOQ model.

2.3.5.4.4 Diffusion Parameters

The applicant chose to implement the diffusion parameter assumptions outlined in RG 1.111, Revision 1, as a function of atmospheric stability and downwind distance from the source, for its XOQDOQ model runs. The staff evaluated the applicability of the XOQDOQ diffusion parameters and concluded that no unique topographic features preclude the use of the XOQDOQ model for the proposed WLS site. This section of the WLS COL FSAR, however, lacked a discussion of nearby topographical features that would preclude the use of the XOQDOQ model on long-term dispersion estimates. Therefore, in RAI 451, Question 02.03.05-2, the staff requested that the applicant provide this information.

In a December 17, 2008, response to RAI 451, Question 02.03.05-2, the applicant provided a revision to WLS COL FSAR Section 2.3.5.1. The revision provided a description of the terrain in the vicinity of the proposed WLS site and further discussed that the terrain in the vicinity of the proposed site would not have a significant effect on atmospheric dispersion estimates. The staff finds the applicant’s use of diffusion parameter assumptions, as outlined in RG 1.111, Revision 1, acceptable. Accordingly, the staff considers RAI 451, Question 02.03.05-2 resolved.

2.3.5.4.5 Resulting Relative Concentration and Relative Deposition Factors

For the purpose of conducting confirmatory analyses of the applicant’s long-term, routine release dispersion estimates using the XOQDOQ computer code, in RAI 451, Question 02.03.05-3, the staff requested that the applicant provide the input files or output files for the 1-year meteorological data set and the 2-year meteorological data set (requested in RAI 3727, Question 02.03.05-4). Along with a December 17, 2008, letter, the applicant submitted the XOQDOQ input and output files for the 2-year meteorological dataset. The staff notes that input and output files were updated for the most recent version of the WLS COL FSAR in a letter dated September 30, 2013. The staff confirmed the receipt of these files and, therefore, considers RAI 3727, Question 02.03.05-4 resolved.

WLS COL FSAR Table 2.3-289, "χ/Q and D/Q Values for Normal Releases," lists the long-term atmospheric dispersion and deposition estimates for the EAB and special receptors of interest that the applicant derived from its XOQDOQ modeling results. WLS COL FSAR Table 2.3-287, "Annual Average χ/Q (sec/m^3) for Normal Releases No Decay, Undepleted," and Table 2.3-288, "Annual Average χ/Q (sec/m^3) for Normal Releases No Decay, Depleted," also contain the applicant's long-term atmospheric dispersion and deposition estimates for the 16 radial sectors from the site boundary to a distance of 80 km (50 mi) from the proposed WLS site.

The χ/Q values presented in WLS COL FSAR Tables 2.3-289 and 2.3-288 reflect several plume radioactive decay and depletion by deposition scenarios. RG 1.111, Regulatory Position C.3 states that radioactive decay and dry deposition should be considered in radiological impact evaluations of potential annual radiation doses to the public, resulting from routine releases of radioactive materials in gaseous effluents. RG 1.111, Revision 1, Regulatory Position C.3 states that an overall half-life of 2.26 days is acceptable for evaluating the radioactive decay of short-lived noble gases and an overall half-life of 8 days is acceptable for evaluating the radioactive decay for all Iodines released to the atmosphere. Definitions for the χ/Q categories listed in the headings of Table 2.3.5-1 of this report are as follows:

- Undepleted/No Decay χ/Q values are used to evaluate ground-level concentrations of long-lived noble gases, Tritium, and Carbon-14. The plume is assumed to travel downwind, without undergoing dry deposition or radioactive decay.

- Undepleted/2.26-Day Decay χ/Q values are used to evaluate ground-level concentrations of short-lived noble gases. The plume is assumed to travel downwind without undergoing dry deposition, but is decayed assuming a half-life of 2.26 days based on the half-life of Xenon-133m.

- Depleted/8.00-Day Decay χ/Q values are used to evaluate ground-level concentrations of radioiodine and particulates. The plume is assumed to travel downwind, with dry deposition, and is decayed, assuming a half-life of 8.00 days, based on the half-life of Iodine-131.

Using the information provided by the applicant, including the 10-m level JFDs of wind speed, wind direction, and atmospheric stability presented in WLS COL FSAR Tables 2.3-235 through 2.3-241, the staff confirmed the applicant's χ/Q and D/Q values by running the XOQDOQ computer code and obtaining consistent results. The JFDs used by the applicant for the long-term diffusion estimates consisted of 13 wind speed categories, compared to the 11 wind speed categories in RG 1.23, Revision 1. To provide a consistent confirmatory analysis of the applicant's methods, the staff also used 13 wind speed categories.

AP1000 DCD Tier 2, Section 2.3.6.5 also states that with regard to environmental assessment, estimates of annual average χ/Q values for 16 radial sectors to a distance of 80 km (50 mi) from the plant should be provided. The applicant provided these values in WLS COL FSAR Tables 2.3-287 through 2.3-292. Using staff-generated JFDs and the XOQDOQ computer code, these χ/Q values were confirmed by the staff and were found to be adequate and acceptable.

2.3.5.5 *Post Combined License Activities*

There are no post COL activities associated with this WLS COL FSAR section.

2.3.5.6 *Conclusion*

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to long-term diffusion estimates and there is no outstanding information expected to be addressed in the WLS COL FSAR relating to this section. The results of the staff's technical evaluation of the information incorporated by reference in the application are documented in NUREG-1793 and its supplements.

COL Information Item 2.3-5 states that a COL applicant shall address the site-specific diffusion estimates and χ/Q values as specified in AP1000 DCD, Section 2.3.5. Based on the meteorological data provided by the applicant and an atmospheric dispersion model that is appropriate for the characteristics of the site and release points, the staff concludes that representative atmospheric dispersion and deposition factors have been calculated for 16 radial sectors from the site boundary to a distance of 80 km (50 mi) as well as for specific locations of potential receptors of interest. The characterization of atmospheric dispersion and deposition conditions are acceptable to meet the criteria described in RG 1.111, Revision 1 and are appropriate for the evaluation to demonstrate compliance with the numerical guides for doses in 10 CFR Part 20 Subpart D and 10 CFR Part 50, Appendix I. The staff finds that the applicant has provided sufficient information to adequately address COL Information Item 2.3-5.

2.4 Hydrologic Engineering

To ensure that a nuclear power plant or plants can be designed, constructed, and safely operated on the combined license (COL) applicant's site and in accordance with the U.S. Nuclear Regulatory Commission (NRC) regulations, the staff evaluated the hydrologic characteristics of the site and surrounding vicinity that may affect the safety of the proposed nuclear power plant. These site characteristics included the maximum flood elevation of surface water from precipitation, riverine processes (runoff, dam breach discharge, channel blockage or diversion), and combined events (e.g., from coincident wind waves). In addition, the staff reviewed the maximum elevation of groundwater and the characteristic ability of the site to attenuate a postulated accidental release of radiological material into surface water and groundwater. The surface-water hydrologic site characteristics determine the design basis flood for the proposed nuclear power plant (William States Lee III Nuclear Station (WLS) Units 1 and 2) and provide the basis for determining whether flood protection will be required. The groundwater hydrologic site characteristics determine the design basis groundwater loadings and provide the basis for radiological dose analysis for a potential receptor from the postulated accidental release of radioactive liquid effluents in surface and ground waters.

The staff prepared Sections 2.4.1 through 2.4.14 of this safety evaluation report (SER) in accordance with the review procedures described in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Sections 2.4.1

through 2.4.14, using information presented in Section 2.4, "Hydrologic Engineering," of the WLS Units 1 and 2 Final Safety Analysis Report (FSAR) Revisions 0 through 11, AP1000 Design Control Document (DCD) Revision 19, the applicant's responses to the staff's requests for additional information (RAIs), and generally available reference materials (e.g., those cited in applicable sections of NUREG-0800).

In Part 7 of the application, the applicant described an administrative standard departure (STD DEP) (Standard Departure 1.1-1) that remaps WLS COL FSAR section numbers to the associated AP1000 DCD section numbers. This standard departure affected Sections 2.4.1 and 2.4.15 in WLS COL FSAR Section 2.4. The staff determined that this departure has no safety significance.

The nominal proposed site grade for the WLS Units 1 and 2 nuclear power block is 180.4 meters (m) (592 feet (ft)) above mean sea level (MSL) with the nuclear island finished floor elevation[22] at 180.7 m (593 ft) above MSL. The ultimate heat sink (UHS) of the advanced passive pressurized water reactor (AP1000) design is the atmosphere. Therefore, hydrologic characteristics associated with conditions that would result in a loss of external water supply (e.g., low water, channel diversions) are not relevant for this particular design. Also, seismic design considerations of water-supply structures are not relevant for this particular design. Therefore, Regulatory Guide (RG) 1.27, "Ultimate Heat Sink for Nuclear Power Plants," and RG 1.29, "Seismic Design Classification," were not part of the regulatory basis for this Section 2.4 review.

As stated above, the site grade near the WLS Units 1 and 2 nuclear power block is 180.4 m (592 ft) above MSL. The following flooding hazard mechanisms, including associated effects, were computed and reported in the WLS COL FSAR.

Table 2.4.1-1 Flooding Hazard Mechanisms

Calculated Flooding Hazards and Associated Effects as Evaluated in WLS COL **FSAR**	**Water-Surface Elevation**	
	ft[(a)]	**m[(a)]**
Local Intense Precipitation and Associated Drainage	592.6	180.6
Flooding from Streams and Rivers	592.3[(b) (c)]	180.5[(b) (c)]
Failure of Dams and Onsite Water-Control/Storage Structures	592.3[(b) (c)]	180.5[(b) (c)]
Flooding from Storm Surge with Wave Runup	583.9	178
Flooding from Seiche[(c)]	--	--
Flooding from Tsunami[(d)]	--	--

[22] The nuclear island finished floor elevation, 180.7 m (593 ft) above MSL, corresponds to the AP1000 DCD reference floor elevation.

Ice-Induced Flooding[(d)]	--	--
Flooding from Channel Migrations or Diversions[(d)]	--	--
(a) above MSL (b) stillwater-surface elevation (c) bounded by other flooding mechanisms (d) not plausible at the site		

2.4.1 Hydrologic Description

2.4.1.1 *Introduction*

WLS COL FSAR Section 2.4.1 describes the site and all safety-related elevations, structures, and systems from the standpoint of hydrologic considerations and provided a topographic map showing the proposed changes to grading and to natural drainage features.

Section 2.4.1 of this report provides a review of the following specific areas: (1) interface of the plant with the hydrosphere including descriptions of site location, major hydrologic features in the site vicinity, surface-water and groundwater characteristics, and the proposed water supply to the plant; (2) hydrologic causal mechanisms that may require special plant design bases or operating limitations with regard to floods and water-supply requirements; (3) current and likely future surface and groundwater uses by the plant and water users in the vicinity of the site that may affect the safety of the plant; (4) available spatial and temporal data relevant for the site review; (5) alternate conceptual models of the hydrology of the site that reasonably bound hydrologic conditions at the site; (6) potential effects of seismic and non-seismic data on the postulated design bases and how they relate to the hydrology in the vicinity of the site and the site region; and (7) any additional information requirements prescribed within the “Contents of Application,” sections of the applicable Subparts to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants.”

As stated above in Section 2.4, hydrologic characteristics associated with conditions that would result in a loss of external water supply and seismic design considerations of water-supply structures are not relevant for the AP1000 design. Therefore, specific area (6), above, was not part of the staff’s review.

2.4.1.2 *Summary of Application*

This section of the WLS COL FSAR describes the site and all safety-related elevations, structures and systems from the standpoint of hydrologic considerations and provides a topographic map showing the proposed changes to grading and to natural drainage features.

The applicant addressed the COL information item identified in AP1000 DCD Tier 2, Section 2.4.1.1, Revision 19 related to hydrologic description as follows:

AP1000 COL Information Item

- WLS COL 2.4-1

In addition, this section addresses the following COL-specific information identified in AP1000 DCD Section 2.4.1.1, Revision 19.

> Combined License applicants referencing the AP1000 certified design will describe major hydrologic features on or in the vicinity of the site including critical elevations of the nuclear island and access routes to the plant.

2.4.1.3 *Regulatory Basis*

NRC regulations for the hydrologic description, and the associated acceptance criteria, are specified in NUREG-0800, Section 2.4.1.

The applicable regulatory requirements for identifying site location and describing the site hydrosphere are set forth in the following:

- 10 CFR 52.79(a)(1)(iii), "Contents of Applications; Technical Information in Final Safety Analysis Report," as it relates to the hydrologic characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

- 10 CFR Part 100, "Reactor Site Criteria," as it relates to identifying and evaluating hydrologic features of the site. The requirement to consider physical site characteristics in site evaluations is specified in 10 CFR 100,20(c).

The staff also used the following regulatory guides for the acceptance criteria identified in NUREG-0800, Section 2.4.1:

- RG 1.59, "Design Basis Floods for Nuclear Power Plants," supplemented by best current practices, as it relates to providing assurance that natural flooding phenomena that could potentially affect the site have been appropriately identified and characterized

- RG 1.102, "Flood Protection for Nuclear Power Plants," as it relates to providing assurance that structures, systems, and components (SSCs) important to safety have been designed to withstand the effects of natural flooding phenomena likely to occur at the site

- RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," as it relates to the contents of a COL application

2.4.1.4 *Technical Evaluation*

2.4.1.4.1 Site and Facilities

Information Submitted by the Applicant

The applicant stated in WLS COL FSAR Section 2.4.1.1 that the WLS site is located in eastern Cherokee County, SC, southwest of the Ninety-Nine Islands Reservoir, a portion of the Broad River, approximately 1.6 km (1 mi) northwest of the Ninety-Nine Islands Dam. The applicant noted that in addition to the Broad River and several tributaries, the Ninety-Nine Islands Reservoir, Make-Up Pond A, Make-Up Pond B, Make-Up Pond C, and Hold-Up Pond A make up the majority of the surface-water features in the vicinity of the WLS site. An embankment within Make-Up Pond B creates the Upper Arm Pond, which is connected to Make-Up Pond B by a culvert. The applicant stated that Make-Up Pond C is an offsite facility, located on a tributary of the Broad River, west of the WLS site.

The applicant selected the AP1000 certified plant design for the WLS application with the designed finished floor elevation at 180.7 m (593 ft) above MSL with the nuclear island basemat at 168.7 m (553.5 ft) above MSL. The WLS site grade would be at 180.4 m (592 ft) above MSL. The applicant also described the plant water systems including the water-consumption and water-treatment system, the intake system, which provides all raw water requirements for the plant, and the discharge system, which disperses cooling-tower blowdown into the Broad River.

Construction activities related to the abandoned Cherokee Nuclear Station resulted in extensive alteration of the site, including clearing of vegetation, construction of roads, a railroad spur, warehouses, power unit buildings, and other support buildings, and extensive excavation, and grading. Currently, the site is a partially developed industrial land.

The partially built reactor containment building of the abandoned Cherokee Nuclear Station would be removed during construction activities for the proposed units. The basemat slab and some warehouses would be retained. A new intake structure for the normal cooling-water supply is planned to be installed on the Broad River and the blowdown discharge would be placed on the upstream side of the Ninety-Nine Islands Dam near the turbine intakes.

Two new AP1000 units are proposed to be built near the abandoned Cherokee Units 1 and 3, at the west and east sides of the existing excavation, respectively (Figure 2.4.1-1 of this report). Each of the two proposed AP1000 units would use two mechanical draft cooling towers for its circulating-water system (CWS) cooling with makeup water supplied from the Broad River via Make-Up Pond A during normal river discharge conditions. Water stored in Make-Up Ponds B and C would be used, in that order, to provide water to Make-Up Pond A for CWS cooling water during low-flow conditions in the Broad River. The UHS for the AP1000 units is the atmosphere.

The AP1000 design consists of five principal structures: the nuclear island, the turbine building, the annex building, the diesel generator building, and the radioactive waste building. Only the nuclear island is designed as a Seismic Category I structure and contains all safety-related equipment.

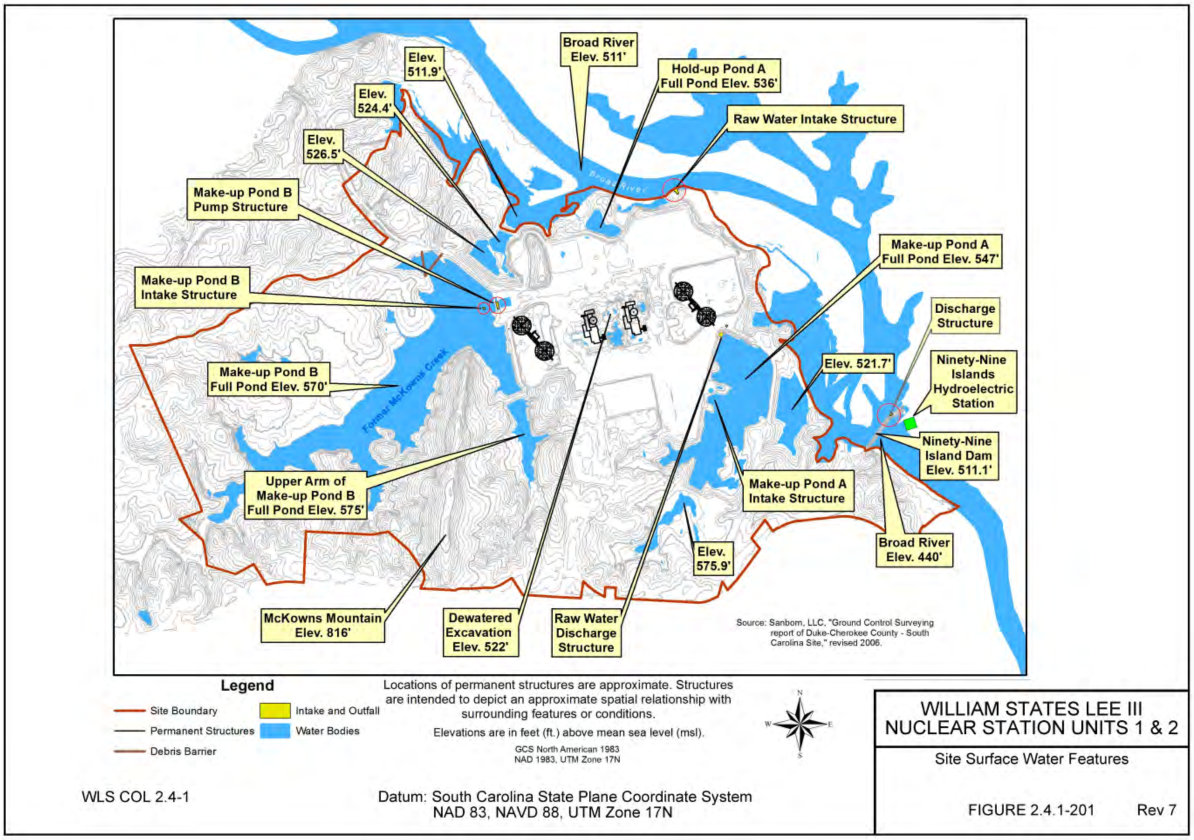


Figure 2.4.1-1 Map of the WLS site (adapted from WLS COL FSAR Revision 7, Figure 2.4.1-201)

The finished floor elevation of the nuclear island, or the AP1000 DCD reference floor elevation of 30.48 m (100 ft), would be placed at 180.7 m (593 ft) above MSL with the nuclear island basemat placed at 168.7 m (553.5 ft) above MSL. The plant grade elevation would be at 180.4 m (592 ft) above MSL.

The applicant stated that the intake system would provide all raw water required by the plant. Raw water would be pumped to Make-Up Pond A from the Broad River under the river's normal flow conditions. During low-flow conditions in the Broad River, water from Make-Up Pond B would be pumped to Make-Up Pond A to provide cooling water to the plant. When the storage in Make-Up Pond B is depleted and low-flow conditions in the Broad River persist, water from Make-Up Pond C would be pumped into Make-Up Pond B and subsequently into Make-Up Pond A to provide cooling water for the plant. Water from the Broad River can also be pumped directly into Make-Up Pond B from the river intake structure. After the Broad River flow returns to normal, Make-Up Pond B would be replenished from water withdrawn from the river. Normally, Make-Up Pond C would be refilled directly from the river intake structure, but an alternate path for refilling Make-Up Pond C would use water from the river intake structure pumped via Make-Up Ponds A and B, in that order.

The applicant stated that the discharge system would use a submerged pipe, perforated along its last portion that would be located near the hydroelectric generating station's intakes upstream of the Ninety-Nine Islands Dam. The discharge from WLS Units 1 and 2 would include non-radioactive and low-level radioactive wastes.

NRC Staff's Technical Evaluation

The staff reviewed the information related to COL Information Item 2.4-1 related to the provision of a description of all major hydrologic features on or in the vicinity of the site, including critical elevations of the nuclear island and access routes to the plant included under WLS COL FSAR Section 2.4.

In RAI 818, Question 02.04.01-1, the staff requested that the applicant clarify the process for determining the conceptual models of the interface of the plant with the hydrosphere and those of the hydrologic causal mechanisms and to ensure that the most conservative of plausible conceptual models has been identified. In a November 18, 2008, response, to RAI 818, Question 02.04.01-1, the applicant stated that the process followed to determine the conceptual models of the interface of the plant with the hydrosphere and those of the hydrologic causal mechanisms would be addressed in the responses to similar staff RAIs issued for the respective subsections of the WLS COL FSAR. The staff reviewed the applicant's responses to these individual RAIs issued for the respective subsections of the WLS COL FSAR and concluded that the applicant had responded to the individual RAIs and included sufficient information to address the staff's concerns. The staff's evaluations of these individual RAIs are presented in the respective sections of this report. Accordingly, the staff considers RAI 818, Question 02.04.01-1, resolved.

The staff conducted a hydrology site audit May 18 to 20, 2008. The site audit included a visit to the WLS site and a tour of the east end of the existing, dewatered excavation pit, the basemat of abandoned Cherokee Nuclear Station Unit 1 on the west end of the excavation pit, Make-Up Pond B, Hold-Up Pond A, the south end of the excavation pit, the dam impounding the south section of the east arm of Make-Up Pond B, the top of McKowns Mountain, the existing intake structure located on Make-Up Pond A, the proposed location of the new intake on the Broad River, and the Ninety-Nine Islands Dam. The applicant and the staff also reconnoitered the Broad River in the vicinity of the site during a boat tour. The staff reviewed the information regarding major hydrologic features in the vicinity of the site from publicly available sources of hydrologic data. The staff's review is described in the subsections below.

In RAI 818, Question 02.04.01-3, the staff requested that the applicant clarify an ambiguity related to the AP1000 DCD reference elevation and the WLS Units 1 and 2 finished floor elevation. In a November 18, 2008, response to RAI 818, Question 02.04.01-3, the applicant stated that the finished floor elevation refers to the AP1000 DCD reference floor elevation of 30.5 m (100 ft), as stated in AP1000 DCD Tier 2, Section 2.4. The applicant modified the WLS COL FSAR text to indicate that the AP1000 DCD reference floor elevation of 30.5 m (100 ft) corresponds to the site-specific nuclear island finished floor elevation. The staff concluded that the applicant provided sufficient information for staff to proceed with its review. Accordingly, the staff considers RAI 818, Question 02.04.01-3, resolved.

In RAI 818, Question 02.04.01-4, the staff requested that the applicant clarify Make-Up Ponds A and B relationship to safety. In a November 18, 2008, response to RAI 818, Question 02.04.01-4, the applicant confirmed that Make-Up Ponds A and B would not be used for any safety-related purpose. After responding to RAI 818, Question 02.04.01-4, the applicant decided to install Make-Up Pond C to address low-water conditions related to normal operations of the proposed units. The exchange of water among the three ponds is described in the "Information Submitted by the Applicant" section above. Since the water pumped from the Broad River to the three makeup ponds would only provide cooling during normal operations, the staff concluded that none of this water would be safety-related. The UHS for the AP1000 units is the atmosphere. The staff concluded that the applicant provided sufficient information related to the function of Make-Up Ponds A and B. Accordingly, the staff considers RAI 818, Question 02.04.01-4, resolved.

In a November 22, 2011, submittal, the applicant provided revisions to the CWS. The applicant stated that the revisions resulted from the change in the design of two cooling towers from the previous design that used three cooling towers per unit.

Based on a review of the material presented by the applicant in WLS COL FSAR Section 2.4.1 and the staff's observations made during the WLS site audit, and based on the reasons given above, the staff concluded that the applicant adequately considered the hydrologic characteristics of the WLS site as they relate to identifying and evaluating hydrologic features.

2.4.1.4.2 Hydrosphere

Information Submitted by the Applicant

The applicant provided a detailed description of the main hydrologic features in the vicinity of the WLS site including the Broad River and the Ninety-Nine Islands Reservoir. The applicant's description included a specific description of the site and all safety-related elevations, structures, exterior access, equipment, and systems from the standpoint of hydrology considerations. The applicant also described the current surface-water features on and off the site (Make-Up Ponds A, B, and C and Hold-Up Pond A) and the local groundwater conditions.

The WLS site is located within the Piedmont physiographic province, the non-mountainous portion of the older Appalachians. The main drainage in the region is the Broad River with tributaries that drain through deep and steep valleys. The Broad River Basin contains rolling hills and small floodplains.

The Broad River and most of its tributaries originate in the Blue Ridge Mountains of North Carolina. The WLS site is located in the Upper Broad River Basin (U.S. Geological Survey (USGS) Hydrologic Unit Code (HUC) 03050105), parts of which lie in both the states of North Carolina and South Carolina. The drainage basin above the Ninety-Nine Islands Dam, approximately 4,014 km^2 (1,550 mi^2) in size, contains tributaries to the Broad River, including the Green River, the First and Second Broad rivers, and Buffalo Creek. Elevations in the watershed upstream of the Ninety-Nine Islands Reservoir range from approximately 366 m (1,200 ft) above MSL at the headwaters of the First Broad River, to approximately 155.8 m (511 ft) above MSL above the Ninety-Nine Islands Dam, and approximately 134 m (440 ft) above MSL below the Ninety-Nine Islands Dam.

Discharge in the Broad River, recorded at USGS streamflow gauge 02153551, located just downstream of the Ninety-Nine Islands Dam, ranges from 3.9 cubic meters per second (m^3/s) (138 cubic feet per second (cfs) on September 14, 2002, to more than 1,699 m^3/s (60,000 cfs) in September 2004. The Gaffney USGS streamflow gauge 02133500, is located approximately 12.9 km (8 mi) upstream of the WLS site and has a contributing area of 155.4 km^2 (60 mi^2) less than that of the Broad River above the Ninety-Nine Islands Dam. The Gaffney USGS gauge recorded the highest discharge on record at 3,373 m^3/s (119,100 cfs) on August 14, 1940. Based on streamflow data at the USGS Gaffney gauge, the applicant estimated that the 100- and 500-year flood discharges in the Broad River are 2,722 to 3,956 m^3/s (97,900 to 127,000 cfs). The applicant also estimated the corresponding water-surface elevations based on the rating curve of the Ninety-Nine Islands Dam, assuming flashboard failure as being at 158.8 and 159.3 m (521 and 522.6 ft) above MSL, respectively.

The applicant filled in data gaps and extrapolated streamflow records from the Gaffney streamflow gauge to construct an 83-year record of streamflow discharge. The applicant estimated that the average annual flow in the Broad River is 70.8 m^3/s (2,500 cfs). The applicant also quantified the low-flow condition in the Broad River using the lowest consecutive 7-day streamflow likely to occur every 10 years, or 7Q10, as 12.4 m^3/s (439 cfs). The applicant stated that the monthly water temperature data from the USGS streamflow gauge 02156500, located near Carlisle, SC, for the period 1996 to 2006, ranged from 4.9ºC to 29.6ºC (40.8ºF to 85.3ºF).

Several small streams in the vicinity of the WLS site exist, including Cherokee Creek, Doolittle Creek, London Creek, McKowns Creek, and an intermittent stream that flows into Make-Up Pond A. McKowns Creek, which has a drainage area of approximately 661 hectares (ha) (1,633 acres (ac)), is impounded by a dam to create Make-Up Pond B on the WLS site. The intermittent stream that flows into Make-Up Pond A has a drainage area of approximately 156 ha (385 ac).

The dam impounding Make-Up Pond B, an earthen structure with its crest elevation at 179.8 m (590 ft) above MSL, was constructed during the 1970s for the abandoned Cherokee Nuclear Station. The spillway elevation is 173.7 m (570 ft) above MSL, which also indicates full pool elevation with a storage capacity of 4,933,927 m^3 (4000 ac-ft) at a surface area of 61 ha (150 ac). Make-Up Pond B is divided into two sections by a submerged dam with a crest elevation of approximately 164.6 m (540 ft) above MSL. Make-Up Pond B has an average depth of 9.6 m (31.4 ft) and a maximum depth of 18.1 m (59.3 ft).

The applicant stated that the outlet structure for Make-Up Pond B is adequately sized. Since the pond is not located on a large stream or river, the applicant expects minimal potential for significant blockage from debris collected at the outlet during flood events. The applicant reported that floating debris has not caused any problems in the past and the spillway has not been clogged. However, the applicant would create a shoreline management program along the banks of Make-Up Pond B. The program would consist of annual inspection of the shoreline and would remove trees that could potentially fall into the pond and trees that may be down on the ground. The applicant would also inspect the spillway for debris accumulation after rainfall events greater than 7.6 cm (3 in.) per hour. The applicant would also install a debris barrier system that rises and falls with the water level in the pond approximately 107 m (350 ft) from the spillway. The applicant stated that the debris barrier system is not considered a safety-related system.

An embankment within Make-Up Pond B creates the Upper Arm Pond, which is connected to Make-Up Pond B by a 137 centimeter (cm) (54 inch (in.)) culvert; the pond has a maximum depth of 9.8 m (32.2 ft), average depth of 9.6 m (31.4 ft), total storage capacity of 124,582 m^3 (101 acre-feet (ac-ft)), a normal pool elevation of 175.3 m (575 ft) above MSL, and a surface area of 3.7 ha (9.1 ac) at full pool. The embankment that impounds the Upper Arm Pond has a crest elevation of 179.8 m (590 ft) above MSL.

Make-Up Pond A was created in the 1970s during the construction of the abandoned Cherokee Nuclear Station by installing an earthen dam, which has a crest elevation varying from 169.9 to 169.2 m (557.5 to 555 ft) above MSL. The full pool elevation of Make-Up Pond A is 166.7 m (547 ft) above MSL; the pond has a surface area of about 25 ha (62 ac) and an estimated volume of 1,757,712 m^3 (1,425 ac-ft).

A small impoundment, named Hold-Up Pond A, is located north of the proposed reactor units and was also created in the 1970s by installing two dams with a crest elevation of 164.5 m (539.7 ft) above MSL. Hold-Up Pond A has a storage volume of about 69,568 m^3 (56.4 ac-ft) at a full pool elevation of 163.4 m (536 ft) above MSL and a surface area of 1.8 ha (4.4 ac).

Make-Up Pond C would be located approximately 2 mi west of the WLS site on London Creek, formed by an earthen dam impounding London Creek just upstream of its confluence with Little London Creek. The Make-Up Pond C Dam crest elevation would be at 201.2 m (660 ft) above MSL with a normal pool elevation of 198.1 m (650 ft) above MSL, a surface area of approximately 251 ha (620 ac), and a total storage capacity of 27,136,600 m^3 (22,000 ac-ft). The usable storage in Make-Up Pond C would be approximately 21,585,932 m^3 (17,500 ac-ft). The drainage area upstream of Make-Up Pond C would be 1,003 ha (2,479 ac).

There are about 132 dams upstream of the WLS site, six of which, Make-Up Pond C Dam, Whelchel Dam, Kings Mountain Reservoir or Moss Lake Dam, Lake Adger or Turner Shoals Dam, Lake Lure Dam, and Lake Summit Dam, impound about 88 percent of total storage in the Broad River Basin. Cherokee Falls Dam and Gaston Shoals Dam, both run-of-river structures, are located on the Broad River immediately upstream of the WLS site and are used for hydroelectric power generation but not for flood control. Cherokee Falls Dam was the first dam constructed in the Upper Broad River Basin in 1826. Two reservoirs located downstream of the WLS site are the Ninety-Nine Islands Reservoir and the Lockhart Reservoir. Dams impounding these two reservoirs are also run-of-river structures and not used for flood control.

The U.S. Army Corps of Engineers (USACE) and the Cleveland County Water (CCW) (previously Cleveland County Sanitary District [CCSD]) proposed to construct a dam on the First Broad River, a tributary of the Broad River upstream of the WLS site, approximately 1.6 km (1 mi) north of Lawndale, NC. The applicant reported that initial feasibility studies estimate that the dam may be approximately 25.3 m (83 ft) high with a 379.5-m (1,245 ft) bottom width and a 304.8 m (1,000 ft) wide spillway. The applicant stated that the surface area of the impoundment would be approximately 909 ha (2,245 ac), covering areas lower than 262.1 m (860 ft) above MSL.

The Piedmont aquifer system mainly consists of two sloped layers. The shallow unconfined layer, the water table aquifer, is composed of saprolite and residual soils. The deeper layer, the bedrock aquifer, is composed of weathered and unweathered crystalline igneous and metamorphic rocks. In the bedrock aquifer, water is stored and transmitted through fractures. The shallow aquifer is unconfined. The fracture system of the bedrock aquifer increases upward in prevalence of fractures as the crystalline rock transitions into saprolite. Due to the increased permeability of the transition zone, the saprolite and bedrock zones function as a single, interconnected aquifer system.

The applicant stated that although there is no single, widespread aquifer in the Piedmont region, local aquifer systems are hydraulically connected. The main source of recharge in the area is infiltration of local precipitation. Local groundwater flow directions can vary depending on topography, fracture characteristics and rock texture.

During the 1973 investigation for the abandoned Cherokee Nuclear Station, several springs and seeps were identified; they were located within valleys that directed surface runoff to the north and to the southeast. During site grading for construction of the abandoned Cherokee Nuclear Stations, these springs and seeps were cut and filled to level the site. Undisturbed topographic features on the WLS site are generally rounded hilltops and narrow valleys. Elevations on the site range from approximately 155.8 m (511 ft) above MSL at the Broad River to approximately 246.9 m (810 ft) above MSL on top of McKowns Mountain, located west of the nuclear power block area and between the two arms of Make-Up Pond B.

The applicant stated that the maximum water use from the Broad River during plant operation is estimated to be 1.8 m^3/s (63 cfs), which is approximately three percent of the average annual mean discharge in the river. The applicant also stated that groundwater would not be used as a primary source for any purpose and for any safety-related purposes for WLS and water for temporary fire protection, concrete batching, and other construction uses would be provided by the Draytonville Water District.

NRC Staff's Technical Evaluation

To review the hydrosphere description and to verify that the analyses in subsequent WLS COL FSAR sections are appropriate, in RAI 818, Question 02.04.01-2, the staff requested that the applicant provide spatially referenced data sets that were used to delineate subbasins and to derive surface and subsurface hydrologic and geologic properties.

In an October 10, 2008, response to RAI 818, Question 02.04.01-2, the applicant provided the geographic information system (GIS) layers for the Broad River watershed including sub-basins, major streams, stream gauge locations, and dam locations. The applicant also provided GIS layers for local site drainage including elevation contours and details of the plant layout. The staff used these data to review the applicant's method to determine flooding in the Broad River watershed, flooding in the drainage areas of Make-Up Ponds A, B, and C, and flooding during the local intense precipitation event. The staff's review is described in subsequent sections herein. The applicant also provided soils search results from the U.S. Department of Agriculture (USDA) Natural Resources Conservation Service (NRCS) Web Soil Survey website at and near the site. Based on its review of the data provided by the applicant, the staff concluded that the applicant provided sufficient information for staff to proceed with its review. Accordingly, the staff considers RAI 818, Question 02.04.01-2, resolved.

To review the appropriateness of COL applicant's methods, in RAI 818, Question 02.04.01-5, the staff requested that the applicant describe the method used to fill gaps in the 81-year streamflow record for the Broad River. In a November 18, 2008, response to RAI 818, Question 02.04.01-5, the applicant stated that it derived the 81-year daily average streamflow record for the Broad River near the WLS site from three USGS stream gauges: the Broad River gauge near Gaffney, SC, located just upstream of the site and with an associated drainage area of 1,490 mi^2; the Broad River gauge near Blacksburg, SC, located approximately 5.0 river km (3.1 river miles) upstream of the Gaffney gauge and with an associated drainage area of 3,341 km^2 (1,290 mi^2); and the Broad River gauge near Boiling Springs, NC, located approximately 26.1 river km (16.2 river miles) upstream of the Gaffney gauge and with an associated drainage area of 2,266 km^2 (875 mi^2). The applicant stated that streamflow data were available for the Gaffney gauge from 1938 to 1971 and from 1986 to 1990, for the Blacksburg gauge from 1997 to 2006, and for the Boiling Springs gauge from 1926 to 2006. The applicant stated that it estimated the 81-year streamflow record for the Gaffney gauge, from 1926 to 2006, by pro-rating the available streamflow data at other gauges by the respective drainage area ratios to fill data for absent years. The applicant estimated the streamflow data at the Gaffney gauge for absent years using the Blacksburg gauge first because that gauge is closer to the Gaffney gauge. The applicant used data from the Boiling Springs gauge for the years that did not have measurements at Gaffney and Blacksburg gauges.

The staff reviewed the applicant's method for estimating the 81-year streamflow record near the site and concluded that it is a commonly used method in hydrologic engineering and therefore, is acceptable. Accordingly, the staff considers RAI 818, Question 02.04.01-5, resolved.

To clarify the description of Make-Up Pond B and changes that may be made to it, the staff issued RAI 818, Question 02.04.01-6. In a November 18, 2008, response to RAI 818, Question 02.04.01-6, the applicant stated that the two sub-basins of the Make-Up Pond B refer to a bathymetric feature of the pond. During the construction of the pond in 1970s, a cofferdam was emplaced to facilitate construction of the Make-Up Pond B Dam. After the pond filled with water, the cofferdam was submerged, creating a division within the pond. The applicant stated that the cofferdam appears as two approximately parallel elevation contours at 164.6 m (540 ft) above MSL midway between the McKowns Mountain and the Make-Up Pond B Dam. The applicant also stated that this cofferdam would be removed during construction of WLS Units 1 and 2 to allow full communication between the two currently existing bathymetric divisions of Make-Up Pond B. The applicant stated that the Make-Up Pond B only provides water storage to support plant operations and has no safety-related function. The staff reviewed the applicant's response and concludes that the applicant provided sufficient information for staff to proceed with its review. Accordingly, the staff considers RAI 818, Question 02.04.01 6, resolved.

To gain a clear understanding of the significance of Hold-Up Pond A with respect to surface release pathways, the staff issued RAI 818, Question 02.04.01-7. In a November 18, 2008, response to RAI 818, Question 02.04.01-7, the applicant stated that the Hold-Up Pond A would be used as a settling pond for stormwater runoff during pre-construction and construction activities, as the discharge point for maintenance dewatering during plant construction, and as a stormwater detention basin during plant operations. The applicant reported that Hold-Up Pond A has a surface area of 1.7 ha (4.2 ac) with a watershed area of 0.08 km^2 (0.03 mi^2). In WLS COL FSAR Revision 2, the applicant reported that the surface area of Hold-Up Pond A is 1.8 ha (4.4 ac) with a storage volume of 69,568 m^3 (56.4 ac-ft) at a full pool elevation.

The applicant also stated that the underground piping constructed for the Cherokee project would be removed and backfilled during construction of WLS Units 1 and 2, which would eliminate man-made preferential groundwater pathways. The staff concluded that the applicant provided sufficient information for the staff to proceed with its review. Accordingly, the staff considers RAI 818, Question 02.04.01-7, resolved. The staff reviewed the technical validity of the applicant's response and its effects on groundwater pathways, and the staff's evaluation is presented in Section 2.4.12 of this report.

To facilitate direct comparison among several elevation values reported in the WLS COL FSAR, in RAI 818, Question 02.04.01-8, the staff requested that the applicant provide clarification regarding consistency of datums used throughout the WLS COL FSAR Section 2.4. In a November 18, 2009, response to RAI 818, Question 02.04.01-8, the applicant provided a list of datums used in WLS COL FSAR figures and has updated the WLS COL FSAR with this information. The staff concluded that the applicant provided sufficient information for staff to proceed with its review. Accordingly, the staff considers RAI 818, Question 02.04.01-8, resolved.

WLS COL FSAR Section 2.4.1.2.3 discusses the existing dams and reservoirs, and also refers to the proposed CCW Dam in WLS COL FSAR Section 2.4.1.2.3.3 "Water Management Changes." The staff subsequently requested additional information from the applicant, the USACE District Office in Wilmington, NC, and the CCW. The USACE provided some information about design details of the proposed reservoir.

In July 7, 2008, and July 11, 2008, emails to the staff, the USACE stated that the proposed CCW Dam would be an earth-fill structure approximately 25.3 m (83 ft) high and 379.5 m (1,245 ft) wide at the base with a 305 m (1,000 ft) wide emergency spillway. The dam would inundate areas below 262 m (860 ft) MSL upstream of it to create a reservoir with a surface area of approximately 526 ha (1,300 ac). The staff noted that the reservoir surface area value of 526 ha (1,300 ac), stated in the USACE email, differs from 909 ha (2,245 ac), the value stated in the *Federal Register* notice of intent pertaining to this proposed dam action. In WLS COL FSAR Section 2.4.4.1, the applicant stated that the estimated storage volume of the reservoir is approximately 58,590,386 m^3 (47,500 ac-ft). In a January 31, 2012, letter to the staff, McGill Associates provided currently available information regarding the proposed First Broad River Reservoir (impounded by the above-mentioned CCW Dam). The dam would be located approximately 5.6 river kilometers (3.5 river miles) upstream of North Carolina Highway 182 and 3.9 river kilometers (2.4 river miles) upstream of the confluence of Knob Creek with the First Broad River. The dam would be an earth-filled structure with the normal pool at 260.9 m (856 ft) above MSL; bottom of the dam at 242.3 m (795 ft) above MSL; crest of the dam, although not final yet, at approximately 266.7 m (875 ft) above MSL; spillway crest elevation at 261.5 m (858 ft) above MSL; and storage capacity at full pool elevation of 260.9 m (856 ft) above MSL of 26,153,778 m^3 (6,909,292,000 gal or 21,203 ac-ft). The spillway would be designed to safely pass 50 percent of the probable maximum precipitation (PMP) storm with a

0.9 m (3 ft) freeboard. The State of North Carolina may require the spillway to be designed to safely pass the PMP storm. The reservoir operation policy has not yet been determined.

On January 2, 2014, the staff received additional information regarding the CCW Dam and the First Broad River Reservoir from the USACE. The USACE informed the staff that the physical characteristics of the proposed dam and the First Broad River Reservoir have not changed. However, the USACE is still working to develop a plan for completing its review of the CCW proposal. The USACE also stated that its review of the CCW proposal may possibly find an alternative site that is more suitable in terms of environmental considerations. The staff's evaluation of the proposed CCW Dam and its potential effects on the safety of the WLS units is described in Section 2.4.4 of this report. As stated in Section 2.4.4.4.3 of this report, the staff concluded that a hypothetical failure of the proposed CCW Dam would not affect the safety of the WLS units.

Based on a review of the material presented by the applicant in WLS COL FSAR Section 2.4.1 and the staff's observations of the WLS site during the May 18 to 20, 2008, site audit, and based on the reasons given above, the staff concluded that the applicant has adequately considered the hydrosphere near the WLS site. Therefore, based on the reasons given above, the staff concluded that the applicant adequately identified the hydrologic characteristics of the proposed site with appropriate consideration of the most severe natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

2.4.1.5 *Post Combined License Activities*

There are no post COL activities related to this section.

2.4.1.6 *Conclusion*

The staff reviewed the application and confirmed that the applicant has demonstrated that the characteristics of the site fall within the site parameters specified in the Design Certification (DC) rule, and that no outstanding information is expected to be addressed in the WLS COL FSAR related to this section.

As set forth above, the applicant has presented and substantiated information to establish the site description. The staff reviewed the information provided and, for the reasons given above, concluded that the applicant has provided sufficient details about the site description to allow the staff to evaluate, as documented in Section 2.4.1 herein, that the applicant has met the relevant requirements of 10 CFR 52.79(a)(1) and 10 CFR Part 100 with respect to determining the acceptability of the site. This section addressed the major hydrologic features, satisfying COL Information Item 2.4-1. The staff concluded that the applicant provided sufficient information to satisfy the applicable requirements of 10 CFR Part 52 and 10 CFR Part 100.

2.4.2 Floods

2.4.2.1 *Introduction*

WLS COL FSAR Section 2.4.2 describes historical flooding at the proposed site or in the region contiguous with the WLS site. The information summarizes and identifies the individual types of flood-producing phenomena, and combinations of flood-producing phenomena considered in establishing the flood design bases for safety-related plant features. The discussion also covers the potential effects of local intense precipitation.

Section 2.4.2 of this report provides a review of the following specific areas: (1) a description of the flood history; (2) flood design considerations; and (3) the effects of local intense precipitation.

2.4.2.2 *Summary of Application*

This section of the WLS COL FSAR describes information about site-specific flooding. The applicant addressed the information item identified in AP1000 DCD Tier 2, Section 2.4.1.2, Revision 19 related to floods as follows:

AP1000 COL Information Item

- WLS COL 2.4-2 Floods

In addition, this section addresses the following COL-specific information identified in AP1000 DCD Tier 2, Section 2.4.1.2, Revision 19.

> Combined License applicants referencing the AP1000 certified design will address the following site-specific information about historical flooding and potential flooding factors, including the effects of local intense precipitation.
>
> - Probable Maximum Flood (PMF) on Streams and Rivers – Site-specific information that will be used to determine the design basis flooding at the site. This information will include the PMF on streams and rivers.
> - Dam Failures – Site-specific information about potential dam failures.
> - Probable Maximum Surge and Seiche Flooding – Site-specific information about probable maximum surge and seiche flooding.
> - Probable Maximum Tsunami Loading – Site-specific information about probable maximum tsunami loading.
> - Flood Protection Requirements – Site-specific information about flood protection requirements or verification that flood protection is not required to meet the site parameter for flood level.

In WLS COL FSAR Section 2.4.2, the applicant addresses the effects of local intense precipitation including the local PMF at the site. Other causes of floods and their effects are discussed in subsequent WLS COL FSAR sections. No further action is required for sites within the bounds of the site parameter for flood level.

2.4.2.3 *Regulatory Basis*

The relevant requirements of NRC regulations for the identification of floods and flood design considerations, and the associated acceptance criteria, are described in NUREG-0800, Section 2.4.2.

The applicable regulatory requirements for identifying floods are as follows:

- 10 CFR 52.79(a)(1)(iii), "Contents of applications; technical information in final safety analysis report," as it relates to the hydrologic characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

- 10 CFR Part 100, "Reactor Site Criteria," as it relates to identifying and evaluating hydrological features of the site. The requirement to consider physical site characteristics in site evaluations is specified in 10 CFR 100.20(c).

The staff also used the following regulatory guides for the acceptance criteria identified in NUREG-0800, Section 2.4.2:

- RG 1.59, "Design Basis Floods for Nuclear Power Plants," as supplemented by best current practices, as it relates to providing assurance that natural flooding phenomena that could potentially affect the site have been appropriately identified and characterized

- RG 1.102, "Flood Protection for Nuclear Power Plants," as it relates to providing assurance that SSCs important to safety have been designed to withstand the effects of natural flooding phenomena likely to occur at the site

- RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," as it relates to the contents of a COL application

2.4.2.4 *Technical Evaluation*

The staff reviewed COL Information Item 2.4-2 related to the provision of site-specific information about historical flooding and potential flooding factors at the plant site included under WLS COL FSAR Section 2.4. Additional aspects of this information item are addressed in Sections 2.4.3, 2.4.4, 2.4.5, 2.4.6, 2.4.7, and 2.4.10 of this report.

To ensure that the design basis flood is based on the most conservative of plausible conceptual models, in RAI 820, Question 02.04.02-1, the staff requested that the applicant describe the process followed to determine the conceptual models for floods from local intense precipitation, PMF in the drainage area upstream of the site, surges, seiche, tsunami, seismically induced dam failures, landslides, and ice effects. In an October 27, 2009, response to RAI 820, Question 02.04.02-1, the applicant stated that the conceptual models to determine the design basis flooding follow the recommendations of RG 1.206 and RG 1.59 and the design basis flooding was determined based on guidance provided in American National Standards Institute/American Nuclear Society standard, ANSI/ANS-2.8-1992, “Determining Design Basis Flooding at Power Reactor Sites.”

The applicant’s analysis of local intense precipitation used the Rational Method to determine runoff. This was appropriate because the area being analyzed is a small developed area. The applicant obtained the precipitation and intensity from the National Oceanic and Atmospheric Administration (NOAA) Hydrometeorological Reports (HMRs) 51 and 52 for a point and assumed the site drainage system would be non-functional during the local intense precipitation event. The applicant estimated the water-surface elevations using the USACE Hydrologic Engineering Center-River Analysis System (HEC-RAS) standard step backwater analysis software. The applicant also stated that flow restrictions were maximized by representing the building structures as obstructions to flow and assuming that they do not provide any flood storage. The applicant also performed a sensitivity analysis by varying the values of Manning’s roughness coefficient.

The applicant stated that the process followed to determine conceptual models for floods from a PMF in the drainage area upstream of the site would be discussed in response to RAI 821, Question 02.04.03-1, those for floods from surges and seiches in response to RAI 823, Question 02.04.05-1, those for floods from tsunami and landslides in response to RAI 824, Question 02.04.06-1, and those for floods from dam failures in response to RAI 822, Question 02.04.04-1.

The applicant consulted the USACE ice jam database and obtained river temperature data from the USGS stream gauge database to determine the conceptual models for floods from ice effects. The applicant stated that because no water is required from the Broad River or from Make-Up Ponds A and B to support safety-related functions of the two proposed units, any potential icing of water supply would have no effects on safety-related facilities.

Subsequently, the applicant decided to install Make-Up Pond C to address low-water conditions related to normal operations of the proposed units. Water stored in Make-Up Ponds B and C would be used, in that order, to provide water to Make-Up Pond A for CWS cooling water during low-flow conditions in the Broad River. Since the water pumped from the Broad River to the three Make-Up Ponds would only provide cooling during normal operations, the staff concluded that none of this water is safety-related. The UHS for the AP1000 units is the atmosphere.

The staff reviewed the applicant’s response to RAI 820, Question 02.04.02-1, and concluded that the applicant provided sufficient description of the process followed to determine conceptual models of flooding at and near the site. Therefore, the staff considers RAI 820, Question 02.04.02-1, resolved. The staff used the information provided by the applicant to evaluate the flooding analyses in subsequent sections of this report.

2.4.2.4.1 Flood History

Information Submitted by the Applicant

The applicant stated that floods on the Broad River occur primarily as a result of precipitation runoff over the watershed. Since dams located upstream of the WLS site are used for water supply and not for flood control, peak discharges in the Broad River near the WLS site are not affected significantly by the dams.

The Gaffney USGS streamflow gauge, (Gauge Number 02153500 with contributing area of 3,859 km^2 (1,490 mi^2)), is located about 8 river km (5 river mi) upstream of the WLS site. The applicant reported that the contributing area of the Broad River upstream of the WLS site is about 4,014 km^2 (1,550 mi^2). The applicant stated that the highest water elevation near the WLS site caused by historical floods (159.3 m (522.5 ft) at USGS Gaffney Station on August 14, 1940, and 156.6 m (513.6 ft) at the Ninety-Nine Islands Reservoir during May 1972) are substantially below the designed nuclear island finished floor elevation of 180.7 m (593 ft) above MSL. The applicant also stated that no historical data exist regarding flooding due to surges, seiches, tsunamis, dam failures, or landslides.

NRC Staff's Technical Evaluation

The staff reviewed the data presented by the applicant in WLS COL FSAR Section 2.4.2 regarding historical flooding and conducted its own analysis. The staff obtained peak streamflow data for USGS streamflow gauges in the Upper Broad River Basin. There are 30 gauges where peak streamflow data are available in the Upper Broad River Basin. Of these 30 gauges, 14 are located upstream of the site (Figure 2.4.2-1 of this report). Table 2.4.2-1 of this report shows the historical maximum peak streamflow reported for each of these streamflow gauges and the corresponding date. The historical maximum peak discharge at the USGS Gaffney Station, 3,373 m^3/s (119,100 cfs), occurred on August 14, 1940. Based on these data, the staff concluded that the flood history presented by the applicant is accurate.

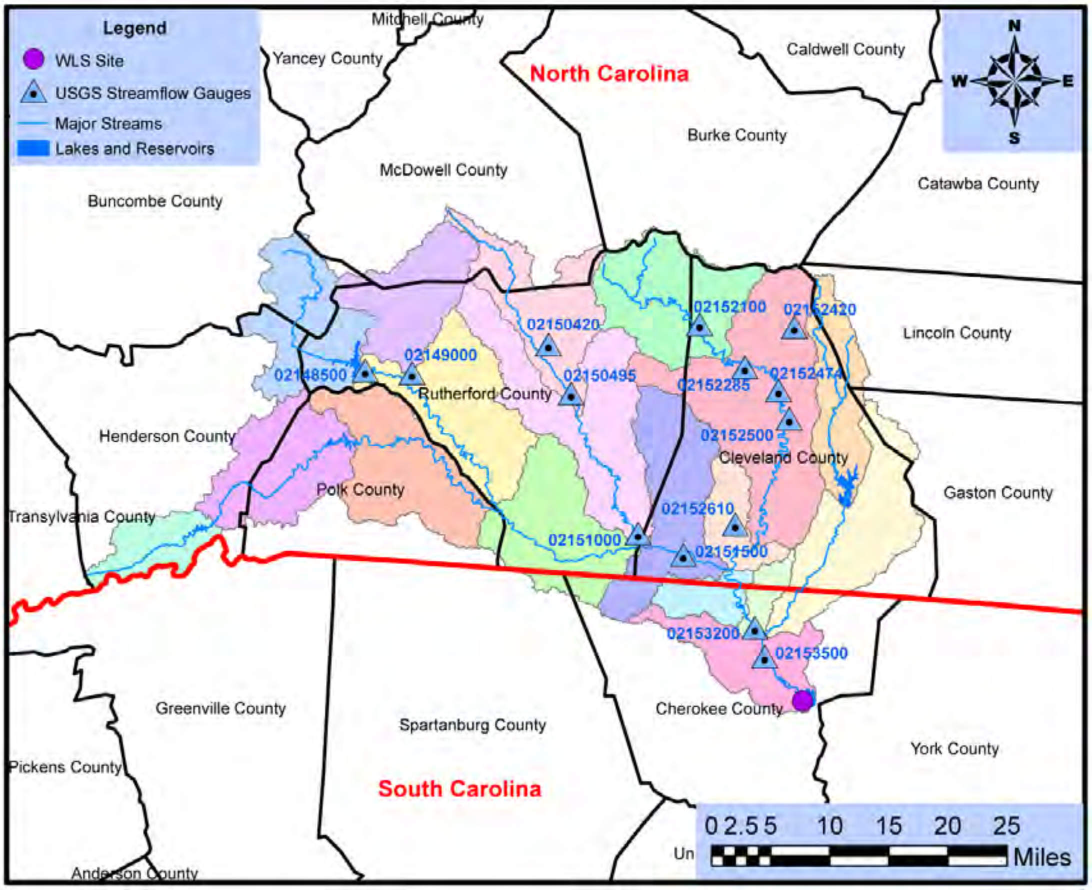


Figure 2.4.2-1 Map of the Upper Broad River Basin Showing Streamflow Stations Used to Describe Historical Flood in the Basin and Showing Adjacent Counties

Table 2.4.2-1 Historical Maximum Peak Streamflow at USGS Streamflow Gauges Upstream of the Site

USGS Streamflow Gauge	Peak Streamflow (m^3/s (cfs))	Date
02148500 Broad River near Chimney Rock, NC	736 (26,000)	1928-08-15
02149000 Cove Creek near Lake Lure, NC	200 (7,050)	1957-06-05
02150420 Camp Creek near Rutherfordton, NC	38 (1,350)	1957-06-04
02150495 Second Broad River near Logan, NC	136 (4,810)	2010-01-25
02151000 Second Broad River at Cliffside, NC	425 (15,000)	1940-08-14
02151500 Broad River near Boiling Springs, NC	2,076 (73,300)	1928-08-16
02152100 First Broad River near Casar, NC	354 (12,500)	2004-09-08
02152285 First Broad River at Sr1512 near Lawndale, NC	84 (2,970)	2009-01-07
02152420 Big Knob Creek near Fallston, NC	96 (3,400)	1970-08-10
02152474 First Broad River at Lawndale, NC	246 (8,690)	2010-01-25
02152500 First Broad River near Lawndale, NC	920 (32,500)	1940-08-14
02152610 Sugar Branch near Boiling Springs, NC	31 (1,110)	1971-10-16
02153200 Broad River near Blacksburg, SC	949 (33,500)	2010-01-25
02153500 Broad River near Gaffney, SC	3,370 (119,000)	1940-08-14

The staff examined a hazard mitigation study conducted by Gaston County (2005). The hazard mitigation study mentions that within Gaston County tsunami and storm surge events are unlikely, landslides (mass earth movements) are unlikely, and snow and ice events are possible; however, with an average annual snowfall of approximately 7.8 cm (3 in.) flooding due to snow and ice events are unlikely. Based on this study and its independent review of historical data, the staff agreed with the applicant that floods due to tsunami, storm surge, landslide, or snow and ice events are unlikely to pose credible flooding hazards at the proposed WLS nuclear reactor site.

The staff reviewed the flood history information provided by the applicant in WLS COL FSAR Section 2.4.2 and independently obtained data from USGS to conclude that the information provided is sufficient to establish the history of flooding at and near the WLS site. Therefore, based on the reasons given above, the staff concluded that the applicant adequately identified and evaluated hydrological features of the site. Since the flood history provided by the applicant can be used as baseline information to compare with the estimated design bases for safety-related SSCs, the staff also concluded that the applicant adequately determined that the hydrologic characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been recorded are met.

2.4.2.4.2 Flood Design Considerations

Information Submitted by the Applicant

The applicant stated that the design basis flood elevation at the WLS site was determined from several scenarios including the effects of local intense precipitation, PMF on streams and rivers, and potential dam failures, and that these flood scenarios are described in their respective WLS COL FSAR sections. The applicant also considered the combinations of appropriate conditions with flooding scenarios such as wind-generated waves. The applicant stated that because of the inland location of the WLS site, consideration of ocean-front surges, seiches, and tsunamis is not necessary. The applicant also stated that consideration of snowmelt and ice effects is not necessary because the WLS site is located in a temperate region.

The applicant estimated that the maximum flood water-surface elevation at the WLS site would result from a local intense precipitation event; the maximum flood water-surface elevation at the WLS site would be 180.6 m (592.6 ft) above MSL. The safety-related plant elevation is 180.7 m (593 ft) above MSL. The applicant has stated that this water-surface elevation, 179.72 m (589.62 ft) above MSL, is identified as a site characteristic.

NRC Staff's Technical Evaluation

The staff reviewed the description of flooding mechanisms provided by the applicant in WLS COL FSAR Sections 2.4.2, 2.4.3, 2.4.4, 2.4.5, 2.4.6, and 2.4.7. The staff's review of these individual flooding mechanisms and their flooding potential is described in detail in the associated sections herein. The staff observed that in the November 22, 2011, revised flooding analysis the applicant changed the design basis flood source mechanism from probable maximum flooding in the watershed of Make-Up Pond B with coincident wind waves to onsite flooding from local intense precipitation. The staff agreed with the applicant that the design basis flood elevation would be caused by local intense precipitation near the site and would be below ground-floor elevation of safety-related SSCs at the WLS site. More details of the design basis flood are described in Section 2.4.3.4.5 of this report.

Based on a review of the applicant's information contained in the WLS COL FSAR, the staff concluded that the applicant appropriately considered flood-causing phenomena and their combinations that are relevant for the WLS site. Based on the reasons given above, the staff found that the requirements of 10 CFR 100.20(c), as they relate to identifying and evaluating hydrological features of the site, are met. The staff agreed that the combinations of flood-causing phenomena considered by the applicant are appropriate for the WLS site. Based on the reasons given above, the staff concluded that the requirements of 10 CFR 52.79(a)(1)(iii) are met, as they relate to the determination of hydrologic characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

2.4.2.4.3 Effects of Local Intense Precipitation

Information Submitted by the Applicant

The applicant estimated the local intense PMP for the WLS site by following the guidance in HMRs 51 and 52. The applicant obtained the 2.6 km^2 (1 mi^2) PMP values from HMR 52 for durations of 1 hour and less. For durations of 6 to 72 hours, the applicant obtained the PMP values from HMR 51 for a 25.6 km^2 (10 mi^2) area. The applicant's estimate of local intense precipitation at the WLS site is shown in Table 2.4.2-2 below.

Table 2.4.2-2 COL Applicant's Estimates of Cumulative Local Intense Precipitation Depths at the WLS Site

	Duration								
	5 min	15 min	30 min	1 hr	6 hr	12 hr	24 hr	48 hr	72 hr
PMP (cm [in.])	15.7 (6.2)	24.6 (9.7)	35.6 (14.0)	48.0 (18.9)	75.9 (29.9)	90.2 (35.5)	102.6 (40.4)	112.5 (44.3)	118.9 (46.8)

The applicant noted that the AP1000 site parameter for PMP is 52.6 cm (20.7 in.) per hour. Therefore, the local intense precipitation depth at the WLS site is lower than the AP1000 site parameter.

The applicant stated that elevations immediately adjacent to the nuclear power block areas enclosed by a roadway, range from the nominal plant grade of 180.4 m (592 ft) to 179.8 m (590 ft) above MSL. Farther away from this area, the grade is flat from the roadway to the plant side of the vehicle barrier system at 179.8 m (590 ft) above MSL. The outer bank of the vehicle barrier system is at an elevation of 179.2 m (588 ft) above MSL. Farther outside the vehicle barrier system, the WLS site is flat at an approximate elevation of 179.2 m (588 ft) above MSL before steeper slopes that form the banks of the adjacent water bodies including Make-Up Ponds A and B, Hold-Up Pond A, and the Broad River. The WLS site drainage area is shown in Figure 2.4.2-2 of this report.

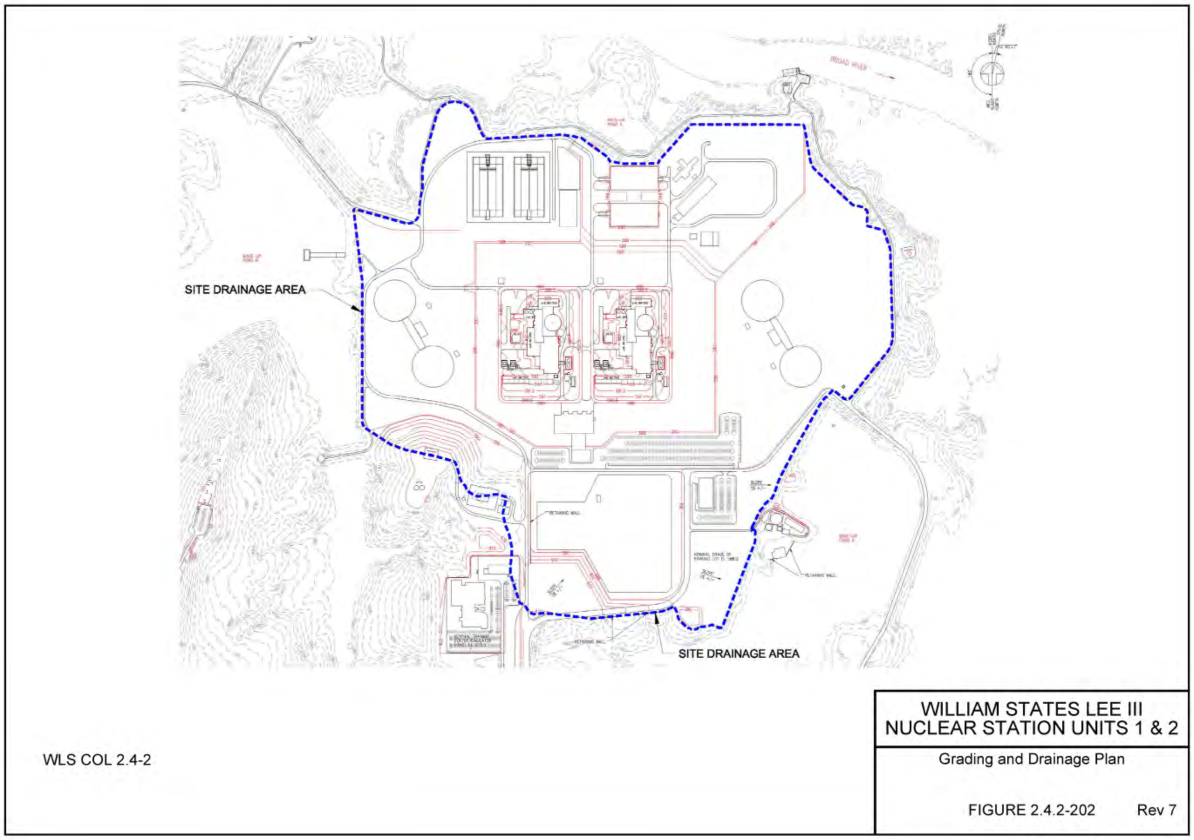


Figure 2.4.2-2 The Site Drainage Area at the WLS Site (Adapted from WLS COL FSAR Revision 7, Figure 2.4.2-202)

The applicant represented the WLS site areas at an elevation of 179.2 m (588 ft) above MSL as an idealized dry reservoir with an elevation-discharge-storage relationship. The applicant refers to this idealized reservoir as the idealized reservoir for the overall site. The storage was estimated using the elevation-area relationship within the drainage area excluding nuclear power block areas that are bounded by the vehicle barrier system, an area north of WLS Unit 2 that slopes from 179.8 m (590 ft) to 179.2 m (588 ft) above MSL, and areas where plant structures and the switchyard are located. The applicant developed the discharge relationship for this reservoir by representing its outer boundary, at an elevation of 179.2 m (588 ft) above MSL, as a broad-crested weir. The weir length is the total length of the 179.2 m (588 ft) above MSL contour minus the length of sections that are deemed ineffective because of less steep downstream slopes. The water-surface elevations in the downstream water bodies provide the downstream boundary conditions for the weir flow. The applicant stated that tailwater conditions would not affect the discharge over the weir; however, the applicant stated that it selected a conservatively low weir discharge coefficient of 2.0. The idealized dry reservoir model was implemented in the USACE Hydrologic Engineering Center Hydrologic Modeling System (HEC-HMS) software Version 3.5.

The applicant simulated two local intense precipitation storms. The first storm used PMP depths for a 72-hour duration with a precipitation interval of 1 hour and the second storm used PMP depths for a 6-hour duration with a precipitation interval of 5 minutes. The applicant estimated the runoff discharge by multiplying the precipitation intensity during each interval by the drainage area; the applicant noted that this approach is equivalent to using the Rational Method with a runoff coefficient of 1 where no losses occur. The applicant used the estimated runoff discharge as inflow to the overall site reservoir model and simulated the resulting water-surface elevation in the reservoir using level-pool routing with the outflow from the reservoir determined by the broad-crested weir equation. The applicant used the water-surface elevation resulting from the 72-hour duration storm as the starting reservoir elevation for the 6-hour duration storm. The applicant reported that the maximum water-surface elevation in the overall site idealized reservoir would be 179.5 m (588.8 ft) above MSL. The applicant used this water-surface elevation as the downstream boundary condition to analyze the WLS site area upstream of the vehicle barrier system.

Similar to the overall site idealized reservoir, the applicant developed an idealized reservoir model for the WLS site area upstream of the vehicle barrier system. The nuclear power block areas bounded by the 179.8 m (590 ft) above MSL contour are not included and all structures provide no storage. The applicant used the broad-crested weir equation to estimate the outflow from the reservoir with the length of the weir determined by the length of the 178.9 m (590 ft) above MSL contour, reduced by the lengths of sections that were deemed ineffective because of the presence of structures. The applicant used a weir discharge coefficient of 2.0. In its analysis, the applicant did not identify any tailwater effects.

Similar to the overall site idealized reservoir, the applicant used two local intense precipitation storms for the idealized reservoir model representing the WLS site area upstream of the vehicle barrier system. Estimation of runoff discharge included no losses. The applicant used the water-surface elevation resulting from the 72-hour duration storm as the starting reservoir elevation for the 6-hour duration storm. Using level-pool routing, the applicant estimated that the maximum water-surface elevation in the reservoir for the 6-hour duration storm would be 180 m (590.6 ft) above MSL. The applicant used this water-surface elevation as the downstream boundary condition to analyze the nuclear power block area of the two units.

For the nuclear power block areas of the two units, the applicant analyzed the runoff from local intense precipitation using four channels within each nuclear power block area (Figure 2.4.2-3 of this report), using a steady-state, backwater analysis in HEC-RAS software version 4.1.0. The applicant did not allow any precipitation losses; the peak runoff discharge used in the steady-state, backwater analysis results from a PMP intensity of 15.7 cm (6.2 in.) in 5 min. The applicant obtained the four channels' cross-section characteristics from the site grading and drainage plan. Structures in the nuclear power block area provide no storage and act as obstructions to open-channel flow. The applicant used a Manning's roughness coefficient of 0.026, appropriate for gravel-lined channels, as a bounding value for ground-cover types in the nuclear power block areas. The applicant reported that the maximum water-surface elevation at the upstream cross section occurred in channels B1 and B2 of the two nuclear power block areas, respectively. The applicant estimated the maximum water-surface elevation to be 180.6 m (592.6 ft) above MSL. The applicant noted that all safety-related structures are located at or above an elevation of 180.7 m (593 ft) above MSL.

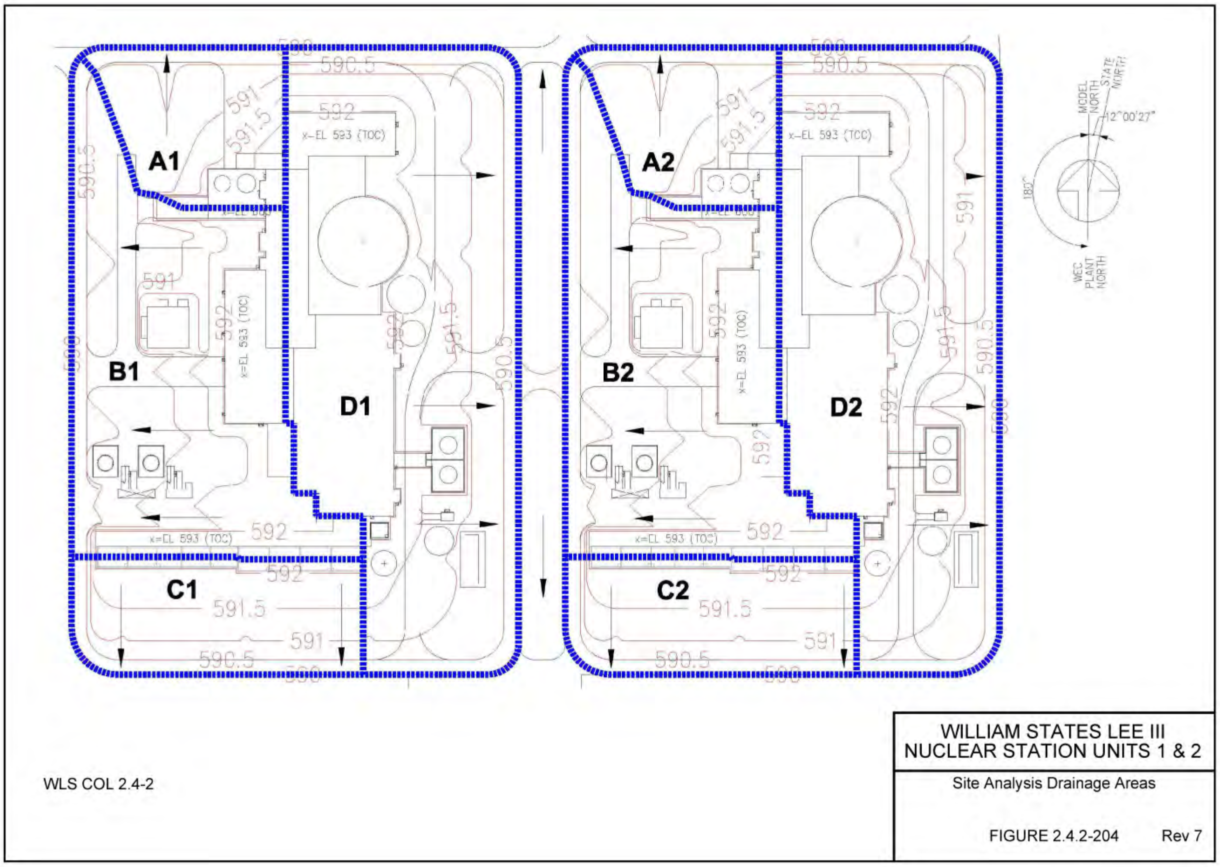


Figure 2.4.2-3 Drainage Pattern within the Nuclear Power Block Areas of the Two Units

NRC Staff's Technical Evaluation

The staff independently estimated the local intense precipitation at the site from NOAA HMRs 51 and 52. The staff-estimated values are shown in Table 2.4.2-3 of this report. The staff compared its independently estimated values of the local intense precipitation at the site to those stated by the applicant and determined that there are small differences in the two estimates. The largest of these differences, approximately 1.6 percent, occurred for the cumulative precipitation depth corresponding to the duration of 48 hours. The staff concluded that these differences are minor and would not significantly affect the estimation of water-surface elevations during site flooding under the local intense precipitation. The staff concluded, therefore, that the applicant's estimate of the local intense precipitation, shown in Table 2.4.2-2 of this report, is acceptable and would be used as a site characteristic.

Table 2.4.2-3 Cumulative Local Intense Precipitation Depths Independently Estimated by the Staff

	Duration								
	5 min	15 min	30 min	1 hr	6 hr	12 hr	24 hr	48 hr	72 hr
PMP (cm [in.])	15.7 (6.2)	24.4 (9.6)	35.6 (14.0)	48.0 (18.9)	76.2 (30.0)	90.7 (35.7)	104.1 (41.0)	114.3 (45.0)	119.4 (47.0)

The following discussion tracks the staff's review of the applicant's local intense precipitation-induced analysis in WLS COL FSAR Revisions 0 through 11. As described above in the subsection, "Information Submitted by the Applicant," the applicant has updated its analysis of flooding in the WLS site area under local intense precipitation after the nominal site grade for WLS Units 1 and 2 nuclear power block and the nuclear island finished floor elevations were raised to 180.4 m (592 ft) and 180.7 m (593 ft) above MSL, respectively.

To understand the analysis performed by the applicant related to the runoff from and water-surface elevations on site drainage areas under local intense precipitation, in RAI 820, Question 02.04.02-2, the staff requested that the applicant: (1) provide input files used in the HEC-RAS analysis described in WLS COL FSAR Section 2.4.2.3; (2) provide details of the iterative process used with the HEC-RAS model to determine water-surface elevations during the local intense precipitation event described in WLS COL FSAR Section 2.4.2.3; (3) provide details of how the times of concentration for the site drainage areas were determined; and (4) provide the locations of safety-related structures where the maximum water-surface elevations for each of the site drainage areas occurred. In an October 27, 2008, response to RAI 820, Question 02.04.02-2, the applicant stated that HEC-RAS and HEC-HMS electronic input and output files were provided to the NRC. The staff used these files to review the applicant's analysis described in WLS COL FSAR Section 2.4.2 and to perform independent confirmatory analyses. The applicant stated that the AP1000 safety-related structures are the containment building and the auxiliary building. The applicant identified these structures on a figure provided with the RAI response.

The staff evaluated the applicant's responses to the RAIs and determined that the applicant has adopted a reasonable approach to identify the effects of local intense precipitation. The staff reviewed the proposed post-construction elevation contour map of the site extending out from the nuclear power block area and determined that the identification of the four subareas based on the site grading plan and the location of safety-related SSCs conform to the elevation contours. The applicant used the Rational Method with an assumption of no losses to estimate the discharges in the four subareas. The applicant used the HEC-RAS hydraulic simulation software to estimate water-surface elevations under steady conditions. The staff concluded that these are conservative approaches and therefore, the applicant's methods for determination of discharges and water-surface elevations during the local intense precipitation event are adequate. Accordingly, the staff considers RAI 820, Question 02.04.02-2, resolved.

However, the staff noted that the applicant's use of the values of Manning's roughness coefficients for grass and for gravel, 0.035 and 0.015, respectively, were potentially not conservative for the corresponding groundcover. A higher value of Manning's roughness coefficient typically results in a greater depth of flow for the same discharge and same channel geometry. Chow (1959) suggested that the value of Manning's roughness coefficient ranges from 0.025 to 0.035 for short grass and from 0.030 to 0.050 for high grass. Since the applicant did not specify which grass type would be prevalent in the grassed areas on the site, the staff conservatively selected a Manning's n value of 0.048. Similarly, because the recommended value for lined or built-up channels with gravel bottom and formed concrete side ranges from 0.017 to 0.025 (1959), the staff selected the value of 0.025 as a conservative value. The applicant used values of 0.035 for grass and 0.015 for paved surfaces. The staff independently performed a HEC-RAS simulation for the northeast subarea of the local site drainage using the input files provided by the applicant. In this analysis, the staff started with the final, converged HEC-RAS simulation performed by the applicant. The only changes made to the HEC-RAS inputs were the Manning's roughness coefficients for grass and for gravel. The staff did not perform any iteration to accurately estimate the time of concentration. The staff's analysis was only meant to estimate how much, if any, effect conservatively selected Manning's roughness coefficients may have on the simulated velocities and water-surface elevations. Table 2.4.2-4 of this report shows the difference in HEC-RAS predictions for flow velocities and those for water-surface elevations for all subareas at cross sections near which safety-related SSC would be located.

Table 2.4.2-4 Difference in Simulated Flow Velocities and Water-Surface Elevations at Cross Sections Near Safety-Related SSCs Due to Change in Manning's Roughness Coefficient

Sub-area	Cross Section	Flow Velocities (m/s [(ft/s])		Water-Surface Elevations (m [ft] Above MSL)	
		COL Applicant	Staff	COL Applicant	Staff
NE	XS10	0.08 (0.25)	0.06 (0.2)	179.47 (588.82)	179.53 (589.01)
NW	XS7	0.17 (0.55)	0.14 (0.45)	179.52 (588.96)	179.58 (589.18)
SE	XS9	0.11 (0.37)	0.10 (0.33)	179.44 (588.7)	179.49 (588.87)
SW	XS8	0.02 (0.06)	0.02 (0.05)	179.63 (589.34)	179.70 (589.58)

As shown in Table 2.4.2-4 above, although the water-surface elevations did not change significantly, the velocities did change by up to 20 percent (at cross-section XS10 in the northeast (NE) subarea), which in turn would require iterations to refine the estimate of the time of concentration. The staff noted, based on the results described above, that the effects of local intense precipitation on the safety-related facilities should be re-evaluated by the applicant using more appropriate values for Manning's roughness coefficient or provide a justification why the base values of Manning's roughness coefficient used in the WLS COL FSAR analysis are conservative. Therefore, in supplemental RAI 72, Question 02.04.02-3, the staff requested that the applicant re-evaluate the effects of local intense precipitation based on more appropriate values of Manning's roughness coefficient, or to justify why the base values of Manning's roughness coefficient used in the WLS COL FSAR analysis were conservative.

In a July 17, 2008, response to RAI 72, Question 02.04.02-3, the applicant stated that a re-evaluation of the effects of the local intense precipitation analysis had been performed with a Manning's roughness coefficient value of 0.050 for grass cover areas and a value of 0.025 for paved and gravel cover areas. The applicant acknowledged the minor increase of the maximum water-surface elevation (from 179.63 m (589.34 ft) above MSL to 179.70 m (589.57 ft) above MSL for the southwest drainage area) due to the local intense precipitation effects. The re-estimated maximum water-surface elevation remained below the plant elevation of all WLS safety-related structures (179.8 m (590 ft) above MSL).

The staff evaluated the applicant's responses to the RAIs and concluded that the applicant has adopted a conservative approach by using the larger values for Manning's roughness coefficient in the HEC-RAS model. Therefore, the staff considers RAI 72, Question 02.04.02-3, resolved.

As described above, the applicant revised the analysis for site flooding under local intense precipitation in response to the staff's RAI 484, Question 10.04.05-2. The staff reviewed the applicant's description of changes to site grading and the applicant's calculations to support the updated site flood analysis. The staff concluded that the applicant's revised analysis that idealizes the site as a dry and shallow reservoir is reasonable because the site area is characterized by gentle slopes and the nuclear power block areas are surrounded by raised roadway that the accumulated runoff would overtop. The staff also concluded that the applicant's analysis to determine water-surface elevations in the nuclear power block areas adjacent to SSCs important to safety is conservative because of the following assumptions and their effects on the water-surface elevation in the idealized reservoir.

- The estimation of runoff ignores all precipitation losses to maximize volume of storage and corresponding water-surface elevation in the idealized reservoir.

- Depression storage was ignored, thereby maximizing the volume of runoff and increase water-surface elevation in the idealized reservoir.

- Plant structures provided obstruction to the flow but no storage, thereby minimizing surface area of the idealized reservoir that would result in higher water-surface elevation for the same storage volume.

- The sheet flow after overtopping could run off the site in all directions, but was assumed to be restricted to three channels that convey runoff to Make-Up Pond B, and the Broad River would result in reduced conveyance and therefore increase storage and corresponding water-surface elevation in the idealized reservoir.

- The downstream boundary conditions in Make-Up Pond B and the Broad River were assumed to be results of their respective PMF events even though the two PMF events would be extremely unlikely to occur concurrently.

The staff also reviewed the parameters the applicant used in the analysis to specify downstream boundary conditions (Manning's n and weir discharge coefficients) and finds them reasonable and conservative. The applicant also used multiple temporal distributions for the local intense precipitation and chose the most conservative results. Therefore, the staff concluded that the applicant's revised analysis for the effects of local intense precipitation on the water-surface elevation within and near the nuclear power block areas is reasonable and conservative. The staff concluded that the applicant-estimated water-surface elevation of 179.72 m (589.62 ft) above MSL that would occur on the west side of each of the proposed units in the area between the Annex Building and the Diesel Generator Building is conservative and acceptable as a site characteristic maximum water-surface elevation. For comparison, this estimate is slightly higher than that reported by the applicant, 179.70 m (589.57 ft) above MSL, in its original analysis.

After raising the nominal site grade for WLS Units 1 and 2 nuclear power block and the nuclear island finished floor elevation to 180.4 m (592 ft) and 180.7 m (593 ft) above MSL, respectively, the applicant updated its analysis of flooding in the WLS site area under a local intense precipitation event. The staff described the applicant's updated analysis above in the subsection, "Information Submitted by the Applicant." The staff reviewed the information provided in the WLS COL FSAR and reviewed the applicant's calculation packages in the reading room.

The staff reviewed the applicant's description of the revised analysis and concluded that the applicant implemented the WLS site area in hydrologic and hydraulic modeling software that is currently accepted in standard engineering practice. The applicant's models incorporate several conservative assumptions:

- No runoff losses are allowed, thereby maximizing the runoff volume, discharge, and the resulting water-surface elevations.
- The two areas modeled using idealized dry reservoirs promote detention of runoff within the reservoirs resulting in higher water-surface elevation.
- The selected value of the weir discharge coefficient would result in smaller discharge from the reservoirs and therefore maximize water-surface elevations within the reservoirs.
- The initial water-surface elevations for the two idealized reservoirs were set to an elevation that would result from a 72-hour-duration local intense precipitation before using the 6-hour, 5-minute local intense precipitation inflow into the reservoirs; this assumption is equivalent to assuming that a 72-hour PMP event was followed by a 6-hour PMP event at the WLS site, which is extremely conservative because it assumes two extreme precipitation events occurring in sequence.
- The Manning's roughness coefficient value of 0.026 is an appropriate maximum value for gravel bottom built-up channels with sides that have random stones in mortar; this assumption would result in a higher water-surface elevation within the nuclear power block areas.

Therefore, the staff concluded that the applicant's revised local intense precipitation-induced flood analysis is reasonable and conservative and would result in conservatively estimated water-surface elevations near the safety-related SSCs. Based on a review of the applicant's information in WLS COL FSAR Revisions 0 through 11, the staff concluded that the applicant has appropriately considered flood-causing phenomena related to local intense precipitation for the WLS site. Therefore, based on the reasons given above, the staff concluded that the applicant adequately identified and evaluated hydrological features of the site. The staff agreed that the flood-causing phenomena associated with local intense precipitation considered by the applicant are appropriate for the WLS site. Therefore, based on the reasons given above, the staff concluded that the applicant adequately determined hydrologic characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

2.4.2.5 *Post Combined License Activities*

There are no post COL activities related to this section.

2.4.2.6 *Conclusion*

The staff reviewed the application and confirmed that the applicant has addressed the information related to individual types of flood-producing phenomena, and combinations of flood-producing phenomena, considered in establishing the flood design bases for safety-related plant features. The information also covered the potential effects of local intense precipitation. The staff also confirmed that there is no outstanding information required to be addressed in the WLS COL FSAR related to this section.

As set forth above, the applicant has presented and substantiated information to establish the site description. The staff reviewed the information provided and, for the reasons given above, concluded that the applicant has provided sufficient details about the site description to allow the staff to evaluate, as documented in Section 2.4.2 of this report, whether the applicant has met the relevant requirements of 10 CFR 52.79(a)(1) and 10 CFR Part 100 with respect to determining the acceptability of the site. This partially addresses COL Information Item 2.4-2. The staff concludes that the applicant has provided sufficient information with respect to the flood history, flood design considerations, and the effects of local intense precipitation to satisfy the applicable requirements of 10 CFR Part 52 and 10 CFR Part 100.

2.4.3 Probable Maximum Flood (PMF) on Streams and Rivers

2.4.3.1 *Introduction*

WLS COL FSAR Section 2.4.3 describes the hydrologic site characteristics affecting any potential hazard to the plant's safety-related facilities as a result of the effect of the PMF on streams and rivers.

Section 2.4.3 herein provides a review of the following specific areas: (1) regional PMPs and their losses; (2) runoff and stream course models; (3) PMF; (4) flood water-surface elevations including effects of coincident wind waves; (5) consideration of other site-related evaluation criteria; and (6) any additional information requirements prescribed in the "Contents of Application" sections of the applicable subparts of 10 CFR Part 52.

2.4.3.2 *Summary of Application*

This section of the WLS COL FSAR describes the site-specific PMFs on streams and rivers. The applicant addressed the information item identified in AP1000 DCD Tier 2, Section 2.4.1.2, Revision 19 related to PMF as follows:

AP1000 COL Information Item

- WLS COL 2.4-2

In addition, this section addresses the following COL-specific information identified in AP1000 DCD Section 2.4.1.2, Revision 19.

> Combined License applicants referencing the AP1000 certified design will address the following site-specific information about historical flooding and potential flooding factors, including the effects of local intense precipitation.
>
> - PMF on Streams and Rivers – Site-specific information that will be used to determine the design basis flooding at the site. This information will include the PMF on streams and rivers.
> - Dam Failures – Site-specific information about potential dam failures.
> - Probable Maximum Surge and Seiche Flooding – Site-specific information about probable maximum surge and seiche flooding.
> - Probable Maximum Tsunami Loading – Site-specific information about probable maximum tsunami loading.
> - Flood Protection Requirements – Site-specific information about flood protection requirements or verification that flood protection is not required to meet the site parameter for flood level.

In WLS COL FSAR Section 2.4.3, the applicant addressed the effects of PMF on streams and rivers. Other causes of floods and their effects are discussed in related WLS COL FSAR sections. No further action is required for sites within the bounds of the site parameter for flood level.

2.4.3.3 *Regulatory Basis*

The relevant requirements of NRC regulations for the identification of floods and flood design considerations, and the associated acceptance criteria, are described in NUREG-0800, Section 2.4.3.

The applicable regulatory requirements for identifying probable maximum flooding on streams and rivers are as follows:

- 10 CFR 52.79(a)(1)(iii), "Contents of applications; technical information in final safety analysis report," as it relates to identifying hydrologic site characteristics with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

- 10 CFR Part 100, "Reactor Site Criteria," as it relates to identifying and evaluating hydrological features of the site. The requirements to consider physical site characteristics in site evaluations are specified in 10 CFR 100.20(c).

- 10 CFR 100.23(d), "Geologic and Seismic Siting Factors," sets forth the criteria to determine the siting factors for plant design bases with respect to seismically induced floods and water waves at the site.

The staff also used the following regulatory guides for the acceptance criteria identified in NUREG-0800, Section 2.4.3:

- RG 1.59, "Design Basis Floods for Nuclear Power Plants," supplemented by best current practices, as it relates to providing assurance that natural flooding phenomena that could potentially affect the site have been appropriately identified and characterized

- RG 1.102, "Flood Protection for Nuclear Power Plants," as it relates to providing assurance that SSCs important to safety have been designed to withstand the effects of natural flooding phenomena likely to occur at the site

- RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," as it relates to the contents of a COL application

2.4.3.4 *Technical Evaluation*

In this section, the staff reviewed the applicant's analysis of PMF on streams and rivers near the WLS site including in the Broad River, in the McKowns Creek and Make-Up Pond B, in the Intermittent Stream and Make-Up Pond A, and in the London Creek and Make-Up Pond C. The staff's independent analysis is also described.

The staff reviewed COL Information Item 2.4-2 related to the provision of site-specific information about the PMF at the plant site included under WLS COL FSAR Section 2.4. Additional aspects of this information item are addressed in Sections 2.4.2. 2.4.4, 2.4.5, 2.4.6, 2.4.7, and 2.4.10 of this report.

To ensure that the design basis flood is based on the most conservative of plausible conceptual models, in RAI 821, Question 02.04.03-1, the staff requested that the applicant describe the process followed to determine the conceptual models for floods in streams and rivers and in the site drainage system. In an October 27, 2008, response to RAI 821, Question 02.04.03-1, the applicant stated that conceptual models used to determine the design basis flood are consistent with the guidance of RG 1.206 and RG 1.59. The applicant stated that the flood estimation approach is consistent with the current state of the practice guidance described in ANSI/ANS-2.8-1992.

The applicant estimated the effects of flooding in Make-Up Ponds A and B using the point PMP values from HMRs 51 and 52. The applicant analyzed several time distributions of the PMP to estimate the most severe flooding effects in the makeup ponds. The applicant used the Soil Conservation Service (SCS) unit hydrograph method to estimate runoff. The applicant maximized the runoff by assuming wet antecedent conditions and no precipitation losses.

The applicant stated that the PMP for the Broad River Basin was estimated using HMRs 51 and 52 and that the PMP over the Broad River Basin was maximized by evaluating several storm centers, storm sizes, and storm orientations. The applicant derived the unit hydrographs for use in the PMF estimation from USGS unit hydrographs for the region. The applicant stated that antecedent storm conditions were chosen to maximize runoff and noted that a PMF in the Broad River Basin was combined with coincident wind-wave activity.

The staff reviewed the applicant's October 27, 2008, response to RAI 821, Question 02.04.03-1, and concluded that the description of the process followed to arrive at conceptual models of floods in streams and rivers is adequate. Therefore, the staff considers RAI 821, Question 02.04.031, resolved. The staff's review of flooding on the WLS site and the nuclear power block areas due to local intense precipitation is described in Section 2.4.2 of this report.

2.4.3.4.1 Probable Maximum Precipitation

Information Submitted by the Applicant

The applicant estimated the PMP for the watershed above the WLS site defined by HMRs 51 and 52 based on an existing study for Ninety-Nine Islands Dam. The applicant optimized the orientation of the PMP storm over the Broad River Basin using the HMR 52 computer software. The applicant used 18 sub-basins in the Broad River Basin upstream of the WLS site. The applicant modified the PMP analysis to include antecedent storm conditions, as specified by RG 1.59, Appendix A and estimated the critical 72-hour storm PMP rainfall total to be 64.7 cm (25.5 in.) for the entire watershed. The applicant used HMR 53 to estimate the winter PMP for the Broad River Basin. The applicant examined the combined event of winter PMP coincident with a 100-year snowpack and stated that snowmelt is not considered to be a factor in modeling the PMF event because the sum of winter PMP and 100-year snowpack is approximately 70 percent of the all-season PMP.

The applicant stated that the PMP for the McKowns Creek and Make-Up Pond B watershed, the Intermittent Stream and Make-Up Pond A watershed, and the London Creek and Make-Up Pond C watershed is the local intense precipitation previously estimated in WLS COL FSAR Section 2.4.2.3.

NRC Staff's Technical Evaluation

To understand the analysis performed by the applicant related to the estimation of PMP on the drainage basin above the Ninety-Nine Islands Dam, in RAI 821, Question 02.04.03-2, the staff requested that the applicant describe the relevance of HMR 53 for determination of the PMP in the Broad River Basin.

In an October 27, 2008, response to RAI 821, Question 02.04.03-2, the applicant stated that the winter PMP for the Broad River Basin is estimated by multiplying the all-season PMP value obtained from HMR 52 by the ratio of the 25.9 km^2 (10 mi^2) winter PMP obtained from HMR 53 to the 25.9 km^2 (10 mi^2) all-season PMP obtained from HMR 51. The applicant estimated the ratio to be approximately 0.567 (66.3 cm/116.8 cm [26.1 in/46.0 in]). Using this ratio, the applicant estimated that the winter PMP over the Broad River Basin for a 72-hour duration would approximately be 36.7 cm (14.5 in.) (0.567 × 64.7 cm (25.5 in.)). The applicant further stated that the 100-year snowpack is estimated to be 43.2 cm (17.0 in.) of snow with a water equivalent of 8.6 cm (3.4 in.) and is described in WLS COL FSAR Section 2.3.1.2.7.1. The applicant subsequently estimated, assuming that the 100-year antecedent snowpack would completely melt during the winter PMP event, that the combined potential runoff during the PMP-on-snowpack event would be 45.5 cm (17.9 in.) (36.0 cm (14.5 in.) winter PMP + 8.6 cm (3.4 in.) 100-year snowpack water equivalent), which is approximately 70 percent of the estimated all-season, 72-hour PMP depth of 64.7 cm (25.5 in.) over the whole Broad River Basin. The applicant concluded, therefore, that snowmelt would not be a significant factor in generation of a controlling PMF event in the Broad River Basin.

The staff evaluated the procedure used by the applicant for the estimation of winter PMP in the Broad River Basin. The applicant's approach is based on the assumption that the ratio of the 25.9 km^2 (10 mi^2) winter precipitation (given in HMR 53) to the 25.9 km^2 (10 mi^2) all-season PMP (given in HMR 51) at a given location remains the same for other drainage areas and times of occurrences. The staff independently obtained the 25.9 km^2 (10 mi^2) PMP depths for the months of December through April from HMR 53 for durations equal to 6, 24, and 72 hours at the WLS site. The staff also independently obtained the all-season PMP depths from HMR 51 for the same durations at the WLS site. These values are shown in Table 2.4.3-1. The staff estimated the ratios of HMR 53 PMP for the months of December through April to the HMR 51 all-season PMP depth. These values are shown in Table 2.4.3-2 of this report.

Based on the data shown in Tables 2.4.3-1 and 2.4.3-2 of this report, the staff concluded that the month-wise ratios of HMR 53 PMP depend on the selected duration of the PMP. There is a clear increasing trend for the value of the ratio with increasing duration of the PMP. The staff also noted that the mean of these ratios for the winter months for the 72-hour duration, 0.569, is very close to that used by the applicant, 0.567.

Table 2.4.3-1 Seasonal PMP Depths from HMR 53 and All-Season PMP Depths from HMR 51 at the WLS Site

Duration (hr)	HMR 53 PMP Depth (cm [in.])					HMR 51 All-Season PMP Depth (cm [in.])
	December	January	February	March	April	
6	36.3 (14.3)	33.0 (13.0)	33.0 (13.0)	36.3 (14.3)	43.2 (17.0)	75.7 (29.8)
24	53.6 (21.1)	50.8 (20.0)	50.8 (20.0)	54.9 (21.6)	61.0 (24.0)	102.1 (40.2)
72	68.1 (26.8)	63.5 (25.0)	63.5 (25.0)	68.1 (26.8)	74.2 (29.2)	118.6 (46.7)

Table 2.4.3-2 Ratio of Seasonal PMP Depths from HMR 53 to the All-Season PMP Depths from HMR 51 at the WLS Site

Duration (hr)	HMR 53 to HMR 51 PMP Depth Ratio					
	December	January	February	March	April	Mean
6	0.480	0.436	0.436	0.480	0.570	0.481
24	0.525	0.498	0.498	0.537	0.597	0.531
72	0.574	0.535	0.535	0.574	0.625	0.569

The staff also noted that the winter PMP values obtained from HMR 53 are appropriate for a 25.9 km^2 (10 mi^2) drainage area. Since the PMP depth for a given duration at a given location decreases with increasing drainage area, the staff concluded that the use of 25.9 km^2 (10 mi^2) winter PMP depths for the Broad River Basin, which is approximately 4,014 km^2 (1,550 mi^2) in size above the Ninety-Nine Islands Dam, is conservative. Therefore, the staff concluded that the applicant's approach for estimation of winter PMP depths over the Broad River Basin is conservative. Accordingly, the staff considers RAI 821, Question 02.04.03-2, resolved.

The staff also determined that the addition of the 100-year snowpack, which is assumed to completely melt during the winter PMP event, to the winter PMP depth over the Broad River Basin results in a combined PMP-on-snow event depth that is less than the all-season PMP depth estimated below. Therefore, the staff agreed with the applicant that the flood generated by a PMP-on-snow event would be less severe than the all-season PMF estimated for the Broad River Basin and does not require further consideration.

The staff independently estimated the all-season PMP over the Broad River Basin using the HMR 52 software developed by the USACE. The staff used a GIS layer of the Broad River Basin and its sub-basin boundaries to estimate normalized coordinates for input to the HMR 52 software. The depth-area-duration values from the all-season PMP for durations of 6, 12, 24, 48, and 72 hours, and drainage areas of 25.9, 518, 2,590, 12,950, 25,900, 51,800 km^2 (10, 200, 1,000, 5,000, 10,000 and 20,000 mi^2) from HMR 51 were also input into the software. The

HMR 52 software automatically finds the optimal orientation of the PMP storm pattern over the input drainage basin. The staff-estimated PMP storm pattern and PMP depths closely agreed with those reported by the applicant. Therefore, the staff considers the applicant's PMP estimates acceptable.

Based on a review of the applicant's information in WLS COL FSAR Revisions 0 through 11, the staff concluded that the applicant has appropriately considered the local intense precipitation for the WLS site. Therefore, based on the reasons given above, the staff concluded that the applicant adequately identified and evaluated extreme precipitation events at the site. The staff agreed that the local intense precipitation considered by the applicant is appropriate for the WLS site. Therefore, based on the reasons given above, the staff concluded that the applicant adequately determinated hydrologic characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

2.4.3.4.2 Precipitation Losses

Information Submitted by the Applicant

The applicant estimated the precipitation losses for Broad River sub-basins and London Creek watershed upstream of Make-Up Pond C based on an existing study using the USDA SCS (now the NRCS) curve number method. The precipitation losses for each sub-basin are provided in the WLS COL FSAR. The applicant's precipitation losses range from approximately 2.5 to 7.3 cm (1 to 2.9 in.) during the 72-hour PMP event. For McKowns Creek watershed upstream of Make-Up Pond B and the Intermittent Stream watershed upstream of Make-Up Pond A, the applicant assumed that no precipitation loss occurs and therefore all rainfall is transformed to runoff.

NRC Staff's Technical Evaluation

The staff reviewed the method followed by the applicant to estimate precipitation loss rates for the PMF estimation in the Broad River Basin and in the watersheds of Make-Up Ponds A, B, and C. The staff determined that the method used to estimate loss rate for the Broad River Basin and the London Creek watershed upstream of Pond C is commonly used in practice. However, the applicant stated that this method resulted in rainfall losses ranging from 37 to 71 percent with a mean of 51 percent during the antecedent storm and in precipitation losses ranging from 3 to 19 percent with an average of 8 percent during the full PMP storm.

The staff evaluated the effect of loss rates estimated by the applicant on the predicted PMF water-surface elevations near the site by independently estimating the sensitivity of the predictions to this parameter. The staff used the HEC-HMS input files provided by the applicant to perform an analysis for the Broad River Basin such that no precipitation losses were allowed. The staff's analysis resulted in a peak discharge of 22,342 m^3/s (789,000 cfs) at the Ninety-Nine Islands Dam compared to the approximately 21,011 m^3/s (742,000 cfs) discharge estimated by the applicant. The staff used the discharges estimated by HEC-HMS model in the HEC-RAS model provided by the applicant to estimate the corresponding water-surface elevations under the no-loss scenario. The staff's evaluation of the Broad River PMF water-surface elevations is described in Section 2.4.3.4.5 of this report.

The staff agreed with the applicant that the no-loss approach used for the drainage areas of Make-Up Ponds A and B is an appropriately conservative approach for determination of the PMF water-surface elevations in the makeup ponds.

Based on a review of the applicant's information in WLS COL FSAR Revision 11, the staff concluded that the applicant has appropriately considered precipitation losses at and near the WLS site. Therefore, based on the reasons given above, the staff concluded that the applicant adequately identified and evaluated extreme flood events at the site. The staff agreed that the precipitation losses considered by the applicant are appropriate for the WLS site. Therefore, based on the reasons given above, the staff concluded that the applicant adequately determined hydrologic characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

2.4.3.4.3 Runoff and Stream Course Models

Information Submitted by the Applicant

The applicant developed the Broad River runoff and stream course model based on an existing HEC-1 study with modifications to include the antecedent rainfall conditions. The applicant used the USACE HEC-HMS Version 3.0.1 modeling software for estimating the runoff and routing calculations and USACE HEC-RAS Version 3.1.3 modeling software to route hydrographs from above Gaston Shoals Dam to Lockhart Dam.

To account for nonlinear basin response at high rainfall rates, the applicant increased the peak of the unit hydrograph by 20 percent and reduced the time to peak by approximately 33 percent. The applicant used the SCS unit hydrograph method as a basis for a modified unit hydrograph to transform rainfall to runoff for the Make-Up Pond C sub-basin. The applicant obtained the HEC-RAS cross sections from an existing study and modified as necessary and used Manning's roughness coefficients given in published tables by Chow.

For McKowns Creek and Make-Up Pond B, the applicant used the USACE HEC-HMS modeling software for estimating runoff and storage routing calculations. To account for nonlinear basin response at high rainfall rates, the applicant increased the peak of the unit hydrograph by 20 percent and reduced the time to peak by approximately 33 percent. The applicant used the SCS unit hydrograph method as a basis for a modified unit hydrograph to transform rainfall to runoff. The applicant estimated the drainage area, length of watercourse, and average slope of the watershed from aerial topography of the area and calculated the lag time using the standard SCS curve number regression equation. The applicant estimated the base flow to have a constant rate of 0.05 m^3/s (1.8 cfs) using the minimum average monthly flow of the Gaffney and Ninety-Nine Island gauges (USGS No. 02153500 and 02153551) with correction on the basis of a ratio of drainage basin areas. The applicant developed the Make-Up Pond B outflow structure rating curve using standard weir and orifice flow equations with coefficients of 3.5 and 0.8, respectively and estimated the available storage based on aerial topography. The applicant assumed a full pond elevation of 174 m (570 ft) above MSL for antecedent conditions.

For McKowns Creek and Make-Up Pond B, the applicant used the USACE HEC-HMS modeling software for estimating runoff and storage routing calculations. To account for nonlinear basin response at high rainfall rates, the applicant increased the peak of the unit hydrograph by 20 percent and reduced the time to peak by approximately 33 percent. The applicant used the SCS unit hydrograph method as a basis for a modified unit hydrograph to transform rainfall to runoff. The applicant estimated the drainage area, length of watercourse, and average slope of the watershed from aerial topography of the area and calculated the lag time using the standard SCS curve number regression equation. The applicant estimated the base flow to have a constant rate of 0.05 m^3/s (1.8 cfs) using the minimum average monthly flow of the Gaffney and Ninety-Nine Island gauges (USGS No. 02153500 and 02153551) with correction on the basis of a ratio of drainage basin areas. The applicant developed the Make-Up Pond B outflow structure rating curve using standard weir and orifice flow equations with coefficients of 3.5 and 0.8, respectively and estimated the available storage based on aerial topography. The applicant assumed a full pond elevation of 174 m (570 ft) above MSL for antecedent conditions.

NRC Staff's Technical Evaluation

In RAI 821, Question 02.04.03-3, the staff requested that the applicant explain why unit hydrographs calibrated using observed runoff events produced by precipitation depths much smaller than the PMP event in the Broad River Basin were appropriate to estimate the PMF in the basin or to update the PMF analysis with techniques recommended by other Federal agencies or those used in standard practice.

In an October 28, 2008, response to RAI 821, Question 02.04.03-3, the applicant stated that the impact of nonlinear basin response is examined as a sensitivity study for Make-Up Pond B watershed. The applicant stated that the SCS unit hydrograph, used to transform rainfall to runoff, was modified to increase the peak discharge by 20 percent and to decrease the time base by approximately 33 percent. The applicant stated that the intermediate ordinates of the unit hydrograph were adjusted to maintain the area under the curve equal to 2.5 cm (1 in.) of rainfall excess. The applicant estimated that the maximum water-surface elevation in Make-Up Pond B would be higher using the modified unit hydrographs. The applicant stated that there would be no significant change in wind-wave activity because the increase in the water-surface elevation in Make-Up Pond B was not large enough to significantly affect wind-wave characteristics. The applicant stated that the higher maximum water-surface elevation estimate in Make-Up Pond B for PMP and coincident wind-wave effects was only provided as a sensitivity analysis and would not supersede the flood elevation previously reported.

The staff evaluated the applicant's response to RAI 821, Question 02.04.03-3, and determined that accounting for nonlinearity of basin response should not be viewed only as a sensitivity effect. The final water-surface elevation used for the design basis must be based on an appropriate analysis that includes the effects of nonlinear basin response to account for the most conservative plausible runoff generation scenario. In RAI 69, Question 02.04.03-6, the staff requested that the applicant provide an analysis of Make-Up Pond B flood water-surface elevation that included the effects of nonlinear basin response.

The staff also noted that the coincident wind-wave activity described above was based on the previously estimated water-surface elevation in Make-Up Pond B. The staff concluded that the new estimate of water-surface elevation should be used to re-estimate wind waves. Therefore, in RAI 69, Question 02.04.03-7, the staff requested that the applicant re-estimate coincident wind waves with the flood stillwater-surface elevation estimated from the analysis that includes the effects of nonlinear basin response. In an October 27, 2008, response to RAI 821, Question 02.04.03-3, the applicant stated that a comparison of maximum flood water-surface elevation in the Broad River and the Make-Up Pond B with the maximum flood water-surface elevation in Make-Up Pond B showed a significant amount of freeboard. Based on this significant amount of freeboard, the applicant concluded that accounting for nonlinear basin response in the Broad River Basin would not increase the maximum flood water-surface elevation in the river or in the Make-Up Pond A enough to exceed the maximum flood water-surface elevation estimated for the Make-Up Pond B.

The staff agreed with the applicant that there appears to be a significant amount of freeboard available between the maximum flood water-surface elevation estimated for the Make-Up Pond B and the maximum flood water-surface elevations estimated for the Broad River and the Make-Up Pond A. The staff determined that it is likely, if nonlinear basin response were accounted for in runoff generation during the Broad River Basin PMF event that the flood water-surface elevation would be higher and therefore, would result in a longer fetch length in the inundated areas of the river floodplain. The staff concluded therefore, that a more detailed analysis that includes the effects of nonlinear basin response and the effects of coincident wind waves is needed to conclusively determine that the design basis flood would result from a PMF in the Make-Up Pond B watershed and not in the Broad River Basin. Therefore, in RAI 69, Question 02.04.03-8, the staff requested that the applicant re-estimate wind waves in the Broad River and in Make-Up Pond A with the flood stillwater-surface elevation that accounted for the nonlinear basin response.

In a June 19, 2009, response to RAI 69, Questions 02.04.03-6, 02.04.03-7, and 02.04.03-8, the applicant re-analyzed the flooding scenario for Make-Up Pond B by calculating the effects of wind-driven waves from a still water elevation that includes the nonlinear basin response effects. The staff examined the applicant’s analysis and concluded that the applicant’s new flood analysis includes nonlinear basin response for Make-Up Pond A and coincident wind-wave effects and, therefore, finds that the applicant’s analysis is conservative. Accordingly, the staff considers RAI 820, Question 02.04.03-3 and RAI 69, Questions 02.04.03-6, 02.04.03-7, and 02.04.03-8, resolved.

Based on a review of the applicant’s information in WLS COL FSAR Revision 11, the staff concluded that the applicant has appropriately considered the characteristics of the streams and rivers near the WLS site. Therefore, based on the reasons given above, the staff concluded that the applicant adequately identified and evaluated extreme flood events at the site. The staff agreed that the local intense precipitation considered by the applicant is appropriate for the WLS site. Therefore, based on the reasons given above, the staff concluded that the applicant adequately determined hydrologic characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

2.4.3.4.4 Probable Maximum Flood Flow

Information Submitted by the Applicant

The applicant estimated the peak PMF discharge at the WLS site as 23,311 m^3/s (823,212 cfs) resulting from the 2,590 km^2 (1,000 mi^2) storm centered near the centroid of the Gaston Shoals Dam drainage basin.

The applicant estimated the peak PMF runoff to be 567 m^3/s (20,039 cfs) and the routed discharge to be 183 m^3/s (6,471 cfs) from a 6-hour two-thirds peaking storm event for the McKowns Creek watershed upstream of Make-Up Pond B. However, the applicant stated that the controlling water-surface elevation occurred during a 72-hour end-peaking storm event with a peak PMF runoff of 536 m^3/s (18,937 cfs) and a routed discharge of 237 m^3/s (8,386 cfs).

The applicant estimated the peak PMF runoff to be 330 m^3/s (11,644 cfs) and a routed discharge to be 279 m^3/s (9,847 cfs) from a 6-hour storm event for the Intermittent Stream watershed upstream of Make-Up Pond A.

The applicant estimated the peak PMF runoff to be 826 m^3/s (29,167 cfs) and a routed discharge to be 300 m^3/s (10,577 cfs) from a 72-hour end-peaking storm event for the London Creek watershed upstream of Make-Up Pond C.

NRC Staff's Technical Evaluation

The staff reviewed the HEC-HMS model files provided by the applicant and determined that the approach used by the applicant for estimating the PMF flood discharge is appropriate. The staff performed a sensitivity analysis on the PMF discharge by changing the loss rate for the Broad River Basin PMF estimation as described in Section 2.4.3.4.2 of this report. As expected, the PMF discharge near the site increased under the no-loss scenario investigated by the staff. The water-surface elevations corresponding to the no-loss scenario were also estimated by the staff. The staff's evaluation is described in Section 2.4.3.4.5 of this report.

Based on a review of the applicant's information in WLS COL FSAR Revision 11, the staff concluded that the applicant has appropriately estimated the PMF discharge in streams and rivers near the WLS site using approaches currently used in standard practice. Therefore, based on the reasons given above, the staff concluded that the applicant adequately identified and evaluated extreme flood events at the site. The staff agreed that the PMF discharge estimated by the applicant is appropriate for the WLS site. Therefore, based on the reasons given above, the staff concluded that the applicant adequately determined hydrologic characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

2.4.3.4.5 Water-Elevation Determinations

Information Submitted by the Applicant

The applicant estimated that the maximum stillwater flood elevation near the WLS site from a PMF in the Broad River Basin would be 168.1 m (551.5 ft) above MSL. The applicant estimated that the maximum flood elevation near the WLS site from a PMF in the McKowns Creek watershed upstream of Make-Up Pond B would be 178.1 m (584.4 ft) above MSL. The applicant stated that Make-Up Pond B is not located on a large river or stream and that the impoundment is confined within a small watershed area. The applicant also stated that it did not consider blockage of the outlet structure in its analysis of the PMF routed through Make-Up Pond B because the outlet structure is sized adequately and there is minimal potential for significant amount of debris to be picked up by floodwaters during the PMF event and subsequently transported to the outlet structure. The applicant stated that its shoreline management program would consist of removing trees from the area around the perimeter of Make-Up Pond B extending 15.2 m (50 ft) beyond contour elevation of 178.3 m (585 ft) above MSL. The applicant stated that this area would be maintained as a grassed, paved, or covered by some other suitable material throughout the operational life of the plant. The maximum water-surface elevation in Make-Up Pond B results from the Upper Arm culvert being non-functional while allowing overtopping and discharge into Make-Up Pond B. During the same event, the applicant estimated that the maximum water-surface elevation in the Upper Arm Pond would be 180.5 m (592.3 ft). The ridges on the east of the Upper Arm Pond separate it from the WLS site.

The applicant estimated the maximum water-surface elevation of Make-Up Pond A from a PMF in the Intermittent Stream watershed as 170.1 m (558.2 ft). The applicant stated that because the PMF discharge flow from Make-Up Pond C is bounded by the PMF discharge in the Broad River Basin, spillover from Make-Up Pond C during a PMF event is not a limiting event for flooding at the WLS site when taken as an isolated event.

NRC Staff's Technical Evaluation

To understand the applicant's rationale behind the selection of the design basis flood elevation at the WLS site and whether any special safety-related structures or systems, particularly a debris collection boom near the Make-Up Pond B spillway, may be needed for flood protection, the staff issued RAI 821, Question 02.04.03-4. In an October 27, 2008, response to RAI 821, Question 02.04.03-4, the applicant stated that the maximum flood water-surface elevation from a PMF event in the watershed of Make-Up Pond B and the coincident effects of wind-induced waves would be significantly below the WLS site grade. The applicant also provided a description of a shoreline management plan with this RAI response. The applicant stated that the shoreline management program would remove trees from the water's edge around Make-Up Pond B. The applicant also stated that this area would be maintained as grassy, paved, or with other suitable cover throughout the operational life of the plant. The applicant concluded that because of the shoreline management plan, blockage of Make-Up Pond B's outlet structure from debris during a PMF event would not be credible.

The staff evaluated the applicant's October 27, 2008, response to RAI 821, Question 02.04.03-4, and concluded that the proposed shoreline management program would limit debris accumulation at the outlet structure during the PMF event in the watershed of Make-Up Pond B. Therefore, the staff concluded that the approach and the method adopted for estimation of the PMF in the watershed of Make-Up Pond B is adequate. However, the staff noted that the applicant needed to include details of the Shoreline Management Plan in WLS COL FSAR Section 2.4.3 because it provides important information related to the justification of the PMF estimation approach for the watershed of Make-Up Pond B. Therefore, in RAI 69, Question 02.04.03-10, the staff requested that the applicant address this need. In a June 19, 2009, response to RAI 69, Question 02.04.03-10, the applicant stated that the details provided in response to the staff's RAI 821, Question 02.04.03-4, were included in WLS COL FSAR Revision 1. The applicant stated that the shoreline management program would remove all trees from the water's edge. The applicant stated that this area would be maintained as paved or grassed or covered with other suitable alternative material throughout the operational life of the plant. The staff reviewed the applicant's response and determined that sufficient detail regarding the shoreline management program has not been provided in the WLS COL FSAR. Because the success of the shoreline management program is essential for maintaining the PMF water-surface elevation in Make-Up Pond B below the AP1000 DCD maximum flood level, the staff determined that the applicant needs to provide additional details of the shoreline management program in the WLS COL FSAR including frequency of inspection, criteria for determination of the need to perform maintenance, and the frequency of maintenance throughout the operational life of the plant. The applicant did not provide any additional details of the shoreline management program in Revision 5 of the WLS COL FSAR. In Revision 7 of the WLS COL FSAR, the applicant stated that the shoreline management program would annually inspect the shoreline around Make-Up Pond B and remove trees that may have fallen on the ground and trees that may be about to fall. The applicant would also inspect the spillway for debris after rainfall events that exceed an intensity of 7.6 cm/hr (3 in/hr). The applicant would install a secondary debris barrier system, approximately 107 m (350 ft) from the Make-Up Pond B spillway. The staff reviewed the details of the applicant's shoreline management program and the secondary barrier system. Since an effective implementation of the shoreline management program would minimize debris available for transport with the PMF and the secondary debris barrier system would be in place, the staff concluded that the combination of the shoreline management program and the debris barrier system would prevent the blockage of the Make-Up Pond B spillway. The staff considers implementation of the shoreline management program and the installation of debris barrier system commitments on the part of the applicant. Accordingly, the staff considers RAI 821, Question 02.04.03-4, and RAI 69, Question 02.04.03-10, resolved.

The staff estimated the PMF discharges in the Broad River Basin under a no precipitation loss HEC-HMS scenario performed to investigate the sensitivity of the PMF water-surface elevations near the site to the value of the selected loss rate. The discharge hydrographs predicted by HEC-HMS were input by the staff into the HEC-RAS model of the Broad River Basin provided by the applicant. The staff's HEC-RAS simulation used combined inflow into the Gaston Shoals Dam as the upstream unsteady boundary condition. Flood discharges from reaches and sub-basins located downstream from the Gaston Shoals Dam were input into the HEC-RAS model as lateral inflows. The staff was able to successfully perform the unsteady simulation. However, based on the description of the HEC-RAS model in the WLS COL FSAR and the additional details provided in RAI responses, the staff was unable to determine location of the cross section near the site. The staff required more information regarding the HEC-RAS model setup to verify the sensitivity of the PMF water-surface elevation to precipitation loss rates. Therefore, in RAI 69, Question 02.04.03-9, the staff requested that the applicant identify the cross section in the HEC-RAS setup that is located directly across from the WLS site.

In a June 19, 2008, response to RAI 69, Question 02.04.03-9, and in a subsequent telephone conversation with the applicant, the staff obtained clarification on the location of the above-mentioned cross section and withdrew RAI 69, Question 02.04.03-9. Accordingly, the staff considers RAI 69, Question 02.04.03-9, resolved. During the staff's simulation of a no precipitation loss scenario, the maximum PMF water-surface elevation in the Broad River at the HEC-RAS cross section located directly across from the WLS site was 168.7 m (553.4 ft) above MSL. The increase in the PMF water-surface elevation near the WLS site under the no precipitation loss scenario was relatively minor, about 0.6 m (1.9 ft), compared to the difference between the PMF water-surface elevation and the WLS site grade, which is about 12.2 m (40 ft). Therefore, the staff concluded that the PMF in the Broad River would not cause water-surface elevations near the WLS site that would affect the safety of the WLS units.

Based on a review of the applicant's information in WLS COL FSAR Revision 11, the staff concluded that the applicant has appropriately estimated the PMF water-surface elevations in streams and rivers near the WLS site using approaches currently used in standard practice. Therefore, based on the reasons given above, the staff concluded that the applicant adequately identified and evaluated extreme flood events at the site. The staff agreed that the PMF water-surface elevations estimated by the applicant are appropriate for the WLS site. Therefore, based on the reasons given above, the staff concluded that the applicant adequately determined hydrologic characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

2.4.3.4.6 Coincident Wind-Wave Activity

Information Submitted by the Applicant

The applicant evaluated the coincident wind-wave activity for the Broad River, Make-Up Pond A, Make-Up Pond B, and Make-Up Pond C. The applicant stated that effects of wind waves for Broad River, Make-Up Pond A, and Make-Up Pond B are addressed in WLS COL FSAR Section 2.4.4.

The applicant stated that Make-Up Pond C is located on a tributary of the Broad River such that wind waves in the pond would not affect the WLS site. The applicant reported that wind-wave activity coincident with PMF in the Broad River Basin and a concurrent failure of Make-Up Pond C Dam is evaluated in WLS COL FSAR Section 2.4.4, which bounds the wind-wave activity in Make-Up Pond C.

NRC Staff's Technical Evaluation

To obtain estimates of coincident wind-wave activity to be included in the estimation of the design basis flood water-surface elevation, the staff issued RAI 821, Question 02.04.03-5.

In an October 27, 2008, response to RAI 821, Question 02.04.03-5, the applicant stated that the effects of wind-wave activity were estimated for flood events in the Broad River and Make-Up Ponds A and B.

The applicant provided an update to the WLS COL FSAR Section 2.4.3.6. The applicant estimated fetch lengths using the longest straight-line fetch from USGS topographic maps. The applicant stated that wave height, wave setup, and wave runup were estimated using the procedures described by USACE. The applicant estimated the 2-year annual extreme mile wind speed using ANSI/ANS-2.8-1992 guidance to be 22.4 m/s (50 mph). The applicant updated the WLS COL FSAR text to reflect changes in wind-wave estimation in response to RAI 69, Questions 02.04.03-6, 02.04.03-7, and 02.04.03-8. The staff evaluated the applicant's response to RAI 821, Question 02.04.03-5, and concluded that the process used to estimate the coincident wind-wave activity is currently used in standard engineering practice and is therefore adequate. Therefore, the staff considers RAI 821, Question 02.04.03-5, resolved.

Based on a review of the applicant's information in WLS COL FSAR Revision 11, the staff concluded that the applicant has appropriately estimated the effects of coincident wind waves in streams and rivers near the WLS site using approaches currently used in standard practice. Therefore, based on the reasons given above, the staff concluded that the applicant adequately identified and evaluated extreme flood events at the site. The staff agreed that the coincident wind-wave activity estimated by the applicant is appropriate for the WLS site. Therefore, based on the reasons given above, the staff concluded that the applicant adequately determined hydrologic characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

2.4.3.5 *Post Combined License Activities*

There are no post COL activities related to this section.

2.4.3.6 *Conclusion*

The staff reviewed the application and confirmed that the applicant has addressed the information relevant to PMF on streams and rivers, and that there is no outstanding information required to be addressed in the WLS COL FSAR related to this section.

As set forth above, the applicant has presented and substantiated information to establish the site description. The staff reviewed the information provided and, for the reasons given above, concluded that the applicant has provided sufficient details about the site description to allow the staff to evaluate, as documented in Section 2.4.3 of this report, whether the applicant has met the relevant requirements of 10 CFR 52.79(a)(1) and 10 CFR Part 100 with respect to determining the acceptability of the site. This addresses COL Information Item 2.4-2 with regard to the PMF. The staff concludes that the applicant has provided sufficient information on hydrologic site characteristics affecting any potential hazard to the plant's safety-related facilities as a result of the effect of the PMF on streams and rivers to satisfy the applicable requirements of 10 CFR Part 52 and 10 CFR Part 100.

2.4.4 Potential Dam Failures

2.4.4.1 *Introduction*

WLS COL FSAR Section 2.4.4 describes potential dam failures to ensure that any potential hazard to safety-related structures due to failure of onsite, upstream, and downstream water-control structures is considered in the plant design.

Section 2.4.4 of this report presents the staff's review of the estimation of flood level caused by different dam failures. The specific areas of review are as follows: (1) dam-failure permutations; (2) unsteady-flow analysis of potential dam failures; (3) water-level determination; and (4) any additional information requirements prescribed in the "Contents of Application" sections of the applicable subparts to 10 CFR Part 52.

2.4.4.2 *Summary of Application*

This section of the WLS COL FSAR describes the site-specific information about floods from potential dam failures. The applicant addressed the information item identified in AP1000 DCD Tier 2, Section 2.4.1.2, Revision 19 related to potential dam failures as follows:

AP1000 COL Information Item

- WLS COL 2.4-2

In addition, this section addresses the following COL Information Item 2.4-2 (COL Action Item 2.4.1-1) identified in AP1000 DCD Section 2.4.1.2.

> Combined License applicants referencing the AP1000 certified design will address the following site-specific information on historical flooding and potential flooding factors, including the effects of local intense precipitation.
>
> - PMF on Streams and Rivers – Site-specific information that will be used to determine the design basis flooding at the site. This information will include the PMF on streams and rivers.
> - Dam Failures – Site-specific information on potential dam failures.
> - Probable Maximum Surge and Seiche Flooding – Site-specific information on probable maximum surge and seiche flooding.
> - Probable Maximum Tsunami Loading – Site-specific information on probable maximum tsunami loading.
> - Flood Protection Requirements – Site-specific information on flood protection requirements or verification that flood protection is not required to meet the site parameter for flood level.

In WLS COL FSAR Section 2.4.4, the applicant addressed the effects of floods caused by potential dam failures. Other causes of floods and their effects are discussed in related WLS COL FSAR sections. No further action is required for sites within the bounds of the site parameter for flood level.

2.4.4.3 *Regulatory Basis*

The relevant requirements of NRC regulations for the identification of floods, flood design considerations and potential dam failures, and the associated acceptance criteria, are described in NUREG-0800, Section 2.4.4.

The applicable regulatory requirements for identifying the effects of dam failures are as follows:

- 10 CFR 52.79(a)(1)(iii), “Contents of applications; technical information in final safety analysis report,” as it relates to identifying hydrologic site characteristics with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

- 10 CFR Part 100, “Reactor Site Criteria,” as it relates to identifying and evaluating hydrological features of the site. The requirement to consider physical site characteristics in site evaluations is specified in 10 CFR 100.20(c).

- 10 CFR 100.23(d), “Geologic and seismic siting factors,” sets forth the criteria to determine the siting factors for plant design bases with respect to seismically induced floods and water waves at the site.

The staff also used the following regulatory guides for the acceptance criteria identified in NUREG-0800, Section 2.4.4:

- RG 1.59, “Design Basis Floods for Nuclear Power Plants,” supplemented by best current practices, as it relates to providing assurance that natural flooding phenomena that could potentially affect the site have been appropriately identified and characterized

- RG 1.102, “Flood Protection for Nuclear Power Plants,” as it relates to providing assurance that SSCs important to safety have been designed to withstand the effects of natural flooding phenomena likely to occur at the site

- RG 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition),” as it relates to the contents of a COL application

2.4.4.4 *Technical Evaluation*

The staff reviewed WLS COL FSAR Section 2.4.4 and checked the referenced AP1000 DCD to ensure that the combination of the AP1000 DCD and the COL application represents the complete scope of information relating to this review topic. The staff’s review confirmed that the information in the application and incorporated by reference addresses the required information relating to the potential dam failure.

The staff reviewed the information in the WLS COL FSAR:

AP1000 COL Information Item

- WLS COL 2.4-2

In its independent review, the staff also considered a breach of the proposed CCW Dam and subsequent flows and resulting water-surface elevation in the Broad River adjacent to the site. The staff concluded that these scenarios were conservative and neither scenario would exceed the design-basis flood elevation. The applicant and the staff used the guidance provided in ANSI/ANS-2.8-1992 to quantify flood water elevations at the site resulting from postulated dam failures.

To ensure that the most conservative of plausible conceptual models has been identified, in RAI 822, Question 02.04.04-1, the staff requested that the applicant describe the process to determine the conceptual models for flood waves from severe breaching of upstream dams, domino-type or cascading failures of dams, dynamic effects on safety-related SSCs, loss of safety-related water supplies, sediment deposition and erosion, and failure of onsite water control or storage structures.

In an October 27, 2008, response to RAI 822, Question 02.04.04-1, the applicant stated that the conceptual models to determine flood waves generated from failures of water-control structures conform to the guidance of RG 1.206 and RG 1.59 and the determination of the flood follows the recommendations of ANSI/ANS-2.8-1992.

The applicant maximized the flood generated by upstream failure of dams by assuming that the failures were coincident with the peak discharge of the PMF and were further enhanced by coincident wind-wave activity. The applicant stated that it selected the dam breach parameters conservatively based on USACE guidance. The applicant assumed that Tuxedo and Turner Shoals dams would fail in a cascade because they are located on the same tributary.

The applicant noted that no water is required for any safety-related purpose for the proposed units from the Broad River or from Make-Up Ponds A, B, and C and therefore, no safety-related facilities would be affected by sediment deposition and erosion. The applicant also stated that there are no onsite water-control or -storage structures, failures of which may produce floods that may affect safety-related SSCs.

The staff reviewed the applicant's October 27, 2008, response to RAI 822, Question 02.04.04-1, and concluded that the approach for determination of upstream single and cascading dam-failure-generated floods near the site is sufficiently conservative because the applicant assumed that the dam failures would coincide with the peak discharge at those locations during the PMF event. The staff also concluded that the applicant's evaluation of cascading failure combinations is adequate because a cascading failure of dams on the same tributary is used. Accordingly, the staff considers RAI 822, Question 02.04.04-1, resolved.

The staff conducted a hydrology site audit during the period of May 18 to 20, 2008. The site audit included a visit to the WLS site and a tour of Make-Up Pond B, the dam impounding the south section of the east arm of Make-Up Pond B, Make-Up Pond A, and the Ninety-Nine Islands Dam. The applicant and staff also reconnoitered the Broad River in the vicinity of the site during a boat tour. The staff did not visit any of the dams located upstream of the WLS site in the Broad River Basin.

2.4.4.4.1 Dam-Failure Permutation

Information Submitted by the Applicant

The applicant stated that the overtopping failure of upstream dams during the PMF event in the Broad River Basin would result in more severe flooding than that caused by seismic failures (failure of dams under safe shutdown earthquake coincident with a 25-year flood and failure of dams under the operating basis earthquake coincident with a one-half PMF or the 500-year flood) because the PMF event is a more severe flood than the coincident floods used for seismic failure scenarios. Therefore, the applicant did not evaluate floods resulting from seismic failure of dams coincident with floods that are lesser in magnitude than the PMF.

The applicant assumed that both Cherokee Falls and Gaston Shoals dams would be overtopped during the PMF event and the dam failure would coincide with the peak PMF discharge. The applicant assumed Cherokee Falls Dam to fail completely in 0.5 hour and the middle section of the Gaston Shoals Dam to fail in 0.5 hour along with the failure of embankment abutments separating the latter dam's three sections.

The applicant estimated that failure of the Gaston Shoals Dam only, coincident with the peak PMF discharge, would result in a discharge of 2,333 m^3/s (824,000 cfs) and a corresponding water-surface elevation of 168.1 m (551.5 ft) above MSL at the WLS site. The applicant also estimated that failures of both Gaston Shoals and the Cherokee Dams, coincident with the peak PMF discharge, would result in the same discharge and water-surface elevation at the WLS site because of small reservoir volumes compared to the PMF discharge.

The applicant reported that major upstream structures include Lake Lure, impounded by a dam on the Broad River; Lake Summit, impounded by the Tuxedo Dam on Green River; Lake Adger, impounded by the Turner Shoals Dam; Kings Mountain Reservoir Dam or Moss Lake Dam located on Buffalo Creek; and Lake Whelchel Dam located approximately 12.9 km (8 mi) northwest of the WLS site on Cherokee Creek. The applicant stated that Lake Lure Dam, Tuxedo Dam, and Turner Shoals Dam are designed to withstand overtopping and assumed that dam failure during overtopping would coincide with the peak PMF discharge. For Kings Mountain Reservoir Dam, the applicant postulated a piping failure with the consequent dam failure coinciding with the peak PMF discharge.

The CCW has applied for a permit to construct a water-supply reservoir on the First Broad River approximately 1.6 km (1 mi) north of Lawndale, North Carolina, approximately 42 km (26 mi) northeast of the WLS site. The proposed dam is expected to be approximately 379 m (1,245 ft) long and 25 m (83 ft) high with an impoundment surface area of approximately 908.5 hectare (2,245 ac) and may inundate areas lower in elevation than 262 m (860 ft) above MSL. The applicant stated that its storage would be approximately 58,590,386 m^3 (47,500 ac-ft). The applicant assumed that the dam would be designed to not fail during a PMF event. Since the proposed dam and reservoir are comparable to the Kings Mountain Reservoir, the applicant assumed that seismic failure of the proposed dam coincident with floods of magnitudes lesser than the PMF would be no worse than that estimated for the Kings Mountain Reservoir. Since the dam-failure analysis includes the failure of Kings Mountain Reservoir, the applicant assumed that the effects of failure of the proposed dam would be less than those estimated for existing dams.

The applicant stated that Make-Up Pond C is located on London Creek. The applicant also stated that Lake Cherokee is located on a tributary to London Creek upstream of Make-Up Pond C. The applicant considered the dams impounding Lake Cherokee and Make-Up Pond C to fail in a cascade. The applicant has considered the failures of Lake Lure Dam, Tuxedo Dam, Turner Shoals Dam, Lake Whelchel Dam, Kings Mountain Reservoir Dam, Lake Cherokee Dam, and Make-Up Pond C Dam coincident with the PMF as the critical dam-failure event in the Broad River Basin upstream of the WLS site. The applicant estimated that the peak dam-failure discharge coincident with a PMF event in the Broad River Basin would be 52,386 m^3/s (1,850,000 cfs).

The applicant also analyzed the failures of Lake Cherokee Dam and Make-Up Pond C Dam during a PMF event in the Make-Up Pond C watershed. The PMF event in the Make-Up Pond C watershed would result from a more intense PMP event compared to the Broad River Basin PMP because of the smaller drainage area of Make-Up Pond C watershed. The applicant determined that the peak discharge during this dam-failure permutation would be 37,831 m^3/s (1,336,000 cfs), which is less than the peak discharge estimated for the Broad River with the postulated upstream dam failures. Therefore, the applicant did not consider the cascading failure of Lake Cherokee and Make-Up Pond C Dams further. The applicant considered the failure of Upper Arm Pond Dam located within Make-Up Pond B. The maximum peak discharge resulting from a 6-hour tail end-peaking storm was 672 m^3/s (23,726 cfs).

The applicant stated that there are no safety-related facilities that would be affected by loss of water supply due to failures of dams and there are no onsite water-control or -storage structures that are located above site grade.

NRC Staff's Technical Evaluation

The staff evaluated the applicant's October 27, 2008, response to RAI 822, Question 02.04.04-1, and concluded that the process used by the applicant to determine dam-failure permutations upstream of the site is adequate. Lake Summit Dam (also called the Tuxedo Dam) and Lake Adger Dam (also called the Turner Shoals Dam) are located on the Green River and their impoundments have storage capacities of 10,410,586 m^3 (8,440 ac-ft) and 14,431,737 m^3 (11,700 ac-ft), respectively at their corresponding normal pool elevations. Lake Lure Dam is located on the Broad River with a storage of approximately 39,835,295 m^3 (32,295 ac-ft) at normal pool elevation. Green River flows downstream from Lakes Summit and Adger and joins the Broad River that flows downstream of Lake Lure. Therefore, the staff concluded that a cascading failure of Lake Summit and Lake Adger dams on the Green River is possible.

Downstream of the confluence of the Green River with the Broad River, Gaston Shoals Dam and Cherokee Falls Dam are located on the Broad River with storage capacities of approximately 3,083,705 m^3 (2,500 ac-ft) and 246,696 m^3 (200 ac-ft), respectively. Two more dams, the Kings Mountain Reservoir Dam on Buffalo Creek, with a storage capacity of approximately 54,273,199 m^3 (44,000 ac-ft) and the Lake Whelchel Dam on Cherokee Creek, with a storage capacity of approximately 7,154,194 m^3 (5,800 ac-ft), are located upstream of the Cherokee Falls Dam. However, the creeks that flow downstream from these two dams join the Broad River downstream of the Gaston Shoals Dam but upstream of the Cherokee Falls Dam. Therefore, staff noted that cascading failures of Kings Mountain Reservoir Dam and Cherokee Falls Dam and those of Lake Whelchel Dam and Cherokee Falls Dam are also plausible. However, the staff concluded that the small impoundment capacity of Cherokee Falls Dam would not significantly enhance the cascading effects of any flood generated by a dam failure upstream of it.

Since the dam impounding Lake Cherokee is located upstream on a tributary to London Creek on which Make-Up Pond C is located, the staff concluded that a cascading failure of these two dams is possible. On the mainstem of the Broad River, a cascading failure of Gaston Shoals and Cherokee Falls Dams is also possible. However, because these dams have small storage impoundments, the increase in the discharge that causes overtopping, especially that of the PMF would be minor. Therefore, the staff concluded that the dam-failure scenario analyzed by the applicant, in which Lake Lure Dam, Lake Summit Dam, Lake Adger Dam, Kings Mountain Reservoir Dam, and Lake Whelchel Dam failed coincident with peak discharges at those locations during the PMF event, is conservative.

The applicant considered the failure of the Upper Arm Pond Dam, which is located within Make-Up Pond B. Since there are no other dams upstream of Make-Up Pond B dam, no other dam-failure permutations need evaluation. Based on the above review, the staff considers RAI 822, Question 02.04.04-1, resolved.

Based on a review of the applicant's information in WLS COL FSAR Revision 11, the staff concluded that the applicant has appropriately considered dam-failure permutations in the Broad River upstream of the WLS site using approaches currently used in standard practice. Therefore, based on the reasons given above, the staff concluded that the applicant adequately identified and evaluated extreme flood events at the site. The staff agreed that the dam-failure permutations considered by the applicant are appropriate for the WLS site. Therefore, based on the reasons given above, the staff concluded that the applicant adequately determined hydrologic characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

2.4.4.4.2 Unsteady-Flow Analysis of Potential Dam Failures

Information Submitted by the Applicant

The applicant used the same HEC-RAS unsteady-flow model previously used in the PMF water-surface elevation calculations, but modified it to use the HEC-RAS dam breach feature. The applicant used the peak PMF discharge, estimated by the HEC-HMS model at the Ninety-Nine

Islands Dam, as input into the HEC-RAS model to perform a steady-state analysis to determine the water-surface elevation at the WLS site.

NRC Staff's Technical Evaluation

The steady-state simulations performed by the applicant with the peak PMF discharge would result in a water-surface elevation that would be asymptotically approached by an unsteady-flow simulation performed with the complete discharge hydrograph. Therefore, the staff determined that the methods used by the applicant are adequately conservative and the water-surface elevation in the Broad River near the site would be conservatively determined by the approach adopted by the applicant.

Based on a review of the applicant's information in WLS COL FSAR Revision 11, the staff concluded that the applicant has appropriately analyzed dam-failure flood flow in the Broad River upstream of the WLS site using approaches currently used in standard practice. Therefore, based on the reasons given above, the staff concluded that the applicant adequately identified and evaluated extreme flood events at the site. The staff agreed that the dam-failure flood flow analyzed by the applicant is appropriate for the WLS site. Therefore, based on the reasons given above, the staff concluded that the applicant adequately determined hydrologic characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

2.4.4.4.3 Water Level at the Plant Site

Information Submitted by the Applicant

The applicant used the HEC-RAS model described above to estimate the water-surface elevation near the WLS site corresponding to a steady-state discharge of 52,386 m^3/s (1,850,000 cfs) in the Broad River near the WLS site. The applicant reported an estimated stillwater-surface elevation of 175.7 m (576.5 ft) above MSL, approximately 5 m (16.5 ft) below the safety-related plant grade of 180.7 m (593 ft) above MSL.

The applicant evaluated the wind-wave activity during the PMF event in the Broad River near the WLS site using the USACE Coastal Engineering Manual. The applicant used a 2-year coincident wind speed of 80 km/h (50 mph), a fetch length of 4.5 km (2.8 mi), and a slope of 40 percent to estimate a combined flood water-surface elevation of 178.2 m (584.8 ft) above MSL.

The applicant used the HEC-HMS model to determine the maximum water-surface elevation in Make-Up Pond B considering failure of the Upper Arm Pond Dam. The maximum water-surface elevations in Make-Up Pond B and Upper Arm Pond were 178.3 m (585.1 ft) and 180.5 m (592.3 ft) above MSL, respectively. The applicant stated that the ridge on the east of Upper Arm Pond separates it from the WLS site and at water-surface elevations above 179.8 m (590 ft) above MSL, discharge from Upper Arm Pond occurs directly to Make-Up Pond B.

NRC Staff's Technical Evaluation

As stated above, the staff concluded that the methods used by the applicant are adequately conservative and the water-surface elevation in the Broad River near the site would be conservatively determined by the approach adopted by the applicant. To assess the impact of wind-induced waves on the flood water-surface elevation under the multiple dam-failure scenarios described by the applicant, in RAI 822, Question 02.04.04-2, the staff requested that the applicant provide wind-induced wave heights coincident with the controlling dam breach flooding scenario.

In an October 27, 2008, response to RAI 822, Question 02.04.04-2, the applicant stated that WLS COL FSAR Section 2.4.4.3 was updated to provide the effects of wind-wave activity coincident with the dam breach scenario investigated for the Broad River. The applicant also included two new figures in the WLS COL FSAR that showed the fetch lengths for the Broad River and Make-Up Pond A during the dam breach flooding scenario.

The applicant stated in the updated WLS COL FSAR text that the wind-wave activity was evaluated for the Broad River to coincide with the PMF including the effects of dam failures. The applicant estimated the critical, longest straight-line fetch length to be approximately 4.5 km (2.8 mi). The applicant stated that the maximum water-surface elevation in the Broad River from the PMF, including the effect of dam failures and the effect of wind-wave activity, would be approximately 178.2 m (584.8 ft) above MSL. Since the safety-related plant grade is at 180.7 m (593 ft) above MSL, the applicant concluded that the site would be safe from flooding due to a PMF including the effects of upstream dam failures and coincident wind-wave activity in the Broad River.

The applicant further stated that during severe flooding events, Make-Up Pond A is inundated by backwaters of the Broad River. Therefore, the applicant evaluated the effects of wind-wave activity in Make-Up Pond A during the PMF event that included the effects of dam failure. The applicant estimated the critical, straight-line fetch length to be approximately 4.3 km (2.7 mi). The applicant reported the 2-year annual extreme mile wind speed to be 22.4 m/s (50 mph) based on the recommendations of ANSI/ANS-2.8-1992. The applicant estimated the critical duration to be 53 minutes and adjusted the wind speed to be 22.3 m/s (49.9 mph). The applicant estimated the significant wave height to be 0.8 m (2.8 ft) and the maximum wave height to be 1.4 m (4.6 ft). The applicant estimated the wind setup to be approximately 0.02 m (0.07 ft) and the maximum runup, using the 47 percent slope along the banks of the Make-Up Pond A near the site, to be approximately 2.88 m (8.9 ft). Therefore, the applicant considered the total wind-wave activity to be approximately 2.7 m (9.53 ft) for Make-Up Pond A. The applicant stated that the maximum water-surface elevation in the Make-Up Pond A from the Broad River PMF including the effect of dam failures and the effect of wind-wave activity would be approximately 178.4 m (585.4 ft) above MSL. Since the safety-related plant grade is at 180.7 m (593 ft) above MSL, the applicant concluded that the site would be safe from flooding due to the Broad River PMF including the effects of upstream dam failures and coincident wind-wave activity in the Make-Up Pond A.

The staff evaluated the applicant's October 27, 2008, response to RAI 822, Question 02.04.04-2, and subsequent changes to the WLS COL FSAR up to and including Revision 7 with respect to the methods used to determine the maximum flood water-surface elevation in the Broad River. The staff concluded that the applicant has used appropriate and conservative methods to estimate the flood water-surface elevation in the Broad River. Accordingly, the staff considers RAI 822, Question 02.04.04-2, resolved. Since the dam failures were postulated to coincide with the peak discharge of the PMF, the staff concluded that this scenario is more conservative than the flood generated only by dam failures or only by a PMF in the Broad River Basin.

The staff noted that a dam is proposed by the CCW (the CCW Dam) on the First Broad River in Cleveland County, NC. The estimated storage of the CCW Dam impoundment would be approximately 58,590,386 m^3 (47,500 ac-ft). Downstream of the proposed CCW Dam, the First Broad River joins the Broad River upstream of the Gaston Shoals Dam. The applicant assumed that the dam would be designed to not fail during a PMF event. The applicant stated in the WLS COL FSAR that because the proposed dam and reservoir are comparable in size to the Kings Mountain Reservoir, a hypothetical seismic failure of the proposed dam coincident with floods of magnitudes lesser than the PMF would be no worse than those estimated for the Kings Mountain Reservoir. The applicant assumed that the effects of failure of the proposed dam would be less than those estimated for existing dams. The staff investigated a hypothetical dam-failure scenario for the proposed CCW Dam, similar to that used for the other dams in the Broad River Basin, as a sensitivity study to determine if the flood water-surface elevation in the river near the site would be significantly affected. As stated in Section 2.4.1.4.2 of this report, the staff received information regarding the proposed CCW Dam from McGill Associates on January 31, 2012, and from the USACE on January 2, 2014.

The staff reviewed the HEC-HMS and HEC-RAS files provided by the applicant and performed an independent simulation to estimate the sensitivity of the flood water-surface elevation by specifying no precipitation loss during the PMP event to maximize the PMF. The staff also used the modified unit hydrographs for the sub-basins of the Broad River Basin to account for nonlinear runoff generation response during the PMF and included a hypothetical failure scenario for the proposed CCW Dam.

Using the HEC-HMS input files provided by the applicant, the staff added a hypothetical CCW Dam located on the First Broad River to modify the Broad River model used by the applicant for dam breach simulations. The staff used the applicant-reported dam height of 25.3 m (83 ft) in the simplified dam breach equation to estimate the peak dam breach discharge of 20,763 m^3/s (733,242 cfs). The height of the CCW Dam according to information received by the staff from McGill Associates on January 31, 2012, is approximately 24.4 m (80 ft). Since the staff used a larger value of dam height, 25.3 m (83 ft), the estimate of peak dam breach discharge is conservative. The staff simulated the modified Broad River dam breach model with concurrent PMF conditions similar to the applicant's approach. The other dams in the Broad River Basin upstream of the WLS site were assumed to fail in a manner identical to the approach used by the applicant.

The staff independently estimated the peak outflow near the WLS site due to dam breaches concurrent with a PMF in the Broad River Basin to be approximately 1,019 m^3/s (36,000 cfs) larger than that estimated by the applicant. The staff used the HEC-RAS model to simulate the water-surface elevation near the WLS site resulting from a steady-state, but increased, discharge. The staff-estimated stillwater elevation near the WLS site was approximately 0.3 m (1 ft) higher than the applicant's estimate.

The staff also evaluated the coincident wind-wave activity for the Broad River with the maximum water-surface elevation resulting from the dam failures concurrent with a PMF event. Using the USGS topographic quadrangles near the WLS site, the staff estimated the longest straight-line fetch length as 4.7 km (2.9 mi).

Using the procedures described by USACE, the staff estimated the maximum wave runup to be 1.6 m (5.1 ft) and a wind setup to be 0.03 (0.1 ft). Therefore, the staff estimated the total dam failure with concurrent PMF water-surface elevation including the effects of coincident wind-wave activity to be 1.9 m (6.1 ft) higher than the applicant's estimate. The staff's conservatively estimated water-surface elevation in the Broad River during the postulated dam-failure permutation, combined with the postulated failure of the CCW Dam and wind waves, was 3.2 m (10.4 ft) lower than the safety-related WLS site grade of 180.7 m (593 ft) above MSL.

As stated in Section 2.4.1.4.2 of this report, the USACE is reviewing the CCW proposal at this time. As stated in this section, the staff evaluated the CCW's proposed site for the First Broad River Reservoir. In a June 1, 2009, letter to the CCW, the USACE identified three alternative reservoir locations in addition to the proposed CCW location. These alternative locations were on Knob Creek, Upper Crooked Run Creek, and Lower Crooked Run Creek. In a more recent communication to the staff, the USACE stated that in its current review, an alternative side stream reservoir on the Upper Crooked Run Creek is being evaluated.

It is possible that the USACE, during its review of the CCW proposal and preparation of an environmental impact statement, may find the alternative Upper Crooked Run Creek site to be environmentally preferable to the CCW's proposed site. To allow for this possibility, the staff evaluated the potential flood at the WLS site produced by a hypothetical failure of a dam similar in characteristics to the CCW Dam located in the Upper Crooked Run Creek drainage. Since the proposed location of the CCW Dam is below the confluence of the Upper Crooked Run Creek and the First Broad River, the staff concluded that the alternative dam location would be farther upstream than the proposed location of the CCW Dam in the First Broad River drainage. The staff also noted that the alternative reservoir being evaluated by the USACE would be an off-stream storage reservoir and therefore may not require construction of a dam on the Upper Crooked Run Creek. However, the staff conservatively assumed that a dam similar to the proposed CCW Dam would be constructed on the Upper Crooked Run Creek because a hypothetical failure of a dam on the Upper Crooked Run Creek would generate a more severe flood in the First Broad River and at the WLS site.

Since the alternative dam location on the Upper Crooked Run Creek is upstream of the CCW Dam's proposed location that the staff analyzed previously in this section and the characteristics of the dam at the two locations are assumed to be similar, the dam failure-generated flood discharge at the two locations would be similar. However, the dam failure-generated flood discharge would need to travel downstream from the two locations to the WLS site via the First Broad River and the Broad River. Since the distance the flood wave would have to travel to the WLS site is greater for the Upper Crooked Run Creek alternative site than for the CCW's proposed site, the attenuation of the peak discharge for the Upper Crooked Run Creek alternative site would be greater. Therefore, the flood effects would be greater if the CCW Dam were located at the CCW's proposed site. Based on this review and analysis, the staff concluded that its analysis of the flood effects from a hypothetical failure of a future CCW Dam described earlier in this section is conservative.

Since the conservatively estimated flood water-surface elevation near the WLS site from dam failures with a concurrent PMF event and coincident wind waves in the Broad River Basin is lower than the safety-related site grade of 180.7 m (593 ft) above MSL, the staff concluded that the safety-related SSCs of the proposed units at the WLS site are unaffected by flooding due to dam failures.

The staff reviewed the applicant's analysis of the failure of the Upper Arm Pond Dam. The staff determined that the applicant used conservative assumptions to estimate the highest water-surface elevations in Make-Up Pond B and the Upper Arm Pond. Since the ridge on the east side of the Upper Arm Pond prevents flood waters from spilling over into the WLS site area without first draining over the Upper Arm Pond Dam into Make-Up Pond B, the staff concluded that the failure of the culvert or overtopping failure of the Upper Arm Pond would not affect the WLS site.

Based on a review of the applicant's information in WLS COL FSAR Revision 11, the staff concluded that the applicant has appropriately analyzed dam-failure flood water-surface elevations in the Broad River upstream of the WLS site, Make-Up Pond B, and Upper Arm Pond using approaches currently used in standard practice. Therefore, based on the reasons given above, the staff concluded that the applicant adequately identified and evaluated extreme flood events at the site. The staff agreed that the dam-failure flood water-surface elevations estimated by the applicant are appropriate for the WLS site. Therefore, based on the reasons given above, the staff concluded that the applicant adequately determined hydrologic characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

2.4.4.5 *Post Combined License Activities*

There are no post COL activities related to this section.

2.4.4.6 *Conclusion*

The staff reviewed the application and confirmed that the applicant has addressed the information relevant to potential dam failures, and that no outstanding information is expected to be addressed in the WLS COL FSAR related to this section.

As set forth above, the applicant has presented and substantiated information to establish the site description. The staff reviewed the information provided and, for the reasons given above, concluded that the applicant has provided sufficient details about the site description to allow the staff to evaluate, as documented in Section 2.4.4 of this report, whether the applicant has met the relevant requirements of 10 CFR 52.79(a)(1) and 10 CFR Part 100 with respect to determining the acceptability of the site. This addressed potential dam failures in COL Information Item 2.4-2. The staff concludes that the applicant provided sufficient information on hazards from extreme flooding events at the site to satisfy the applicable requirements of 10 CFR Part 52 and 10 CFR Part 100.

2.4.5 Probable Maximum Surge and Seiche Flooding

2.4.5.1 *Introduction*

WLS COL FSAR Section 2.4.5 describes probable maximum surge and seiche flooding to ensure that any potential hazard to the safety-related SSCs at the proposed site has been considered in compliance with NRC regulations.

Section 2.4.5 of this report presents the evaluation of the following topics based on data provided by the applicant in the WLS COL FSAR and information available from other sources: (1) probable maximum hurricane (PMH) that causes the probable maximum surge as it approaches the site along a critical path at an optimum rate of movement; (2) probable maximum wind storm (PMWS) from a hypothetical extratropical cyclone or a moving squall line that approaches the site along a critical path at an optimum rate of movement; (3) a seiche near the site, and the potential for seiche wave oscillations at the natural periodicity of a water body that may affect flood water-surface elevations near the site or cause a low water-surface elevation affecting safety-related water supplies; (4) the potential effects of seismic and non-seismic information on the postulated design bases and how they relate to a surge and seiche in the vicinity of the site and the site region; and (5) any additional information requirements prescribed in the "Contents of Application" sections of the applicable subparts to 10 CFR Part 52.

As stated in Section 2.4 above, hydrologic characteristics associated with conditions that would result in a loss of external water supply and seismic design considerations of water-supply structures are not relevant for the AP1000 design. Therefore, the low-water aspect of item (3) above was not part of the staff's review.

2.4.5.2 *Summary of Application*

This section of the WLS COL FSAR describes the site-specific information on probable maximum surge and seiche flooding in terms of effects on structures and water supply. The applicant addressed the information item identified in AP1000 DCD Tier 2, Section 2.4.1.2, Revision 19 related to probable maximum surge and seiche flooding as follows:

<u>AP1000 COL Information Item</u>

- WLS COL 2.4-2

In addition, this section addresses the following COL-specific information identified in AP1000 DCD Section 2.4.1.2, Revision 19.

> Combined License applicants referencing the AP1000 certified design will address the following site-specific information about historical flooding and potential flooding factors, including the effects of local intense precipitation.
>
> - PMF on Streams and Rivers – Site-specific information that will be used to determine the design basis flooding at the site. This information will include the PMF on streams and rivers.

- Dam Failures – Site-specific information about potential dam failures.

- Probable Maximum Surge and Seiche Flooding – Site-specific information about probable maximum surge and seiche flooding.

- Probable Maximum Tsunami Loading – Site-specific information about probable maximum tsunami loading.

- Flood Protection Requirements – Site-specific information about flood protection requirements or verification that flood protection is not required to meet the site parameter for flood level.

In WLS COL FSAR Section 2.4.5, the applicant addressed the effects of floods caused by probable maximum surge and seiche. Other causes of floods and their effects are discussed in related WLS COL FSAR sections. No further action is required for sites within the bounds of the site parameter for flood level.

2.4.5.3 *Regulatory Basis*

The relevant requirements of NRC regulations for consideration of the effects of probable maximum surge and seiche, and the associated acceptance criteria, are described in NUREG-0800, Section 2.4.5.

The applicable regulatory requirements for identifying surge and seiche hazards are as follows:

- 10 CFR 52.79(a)(1)(iii), "Contents of applications; technical information in final safety analysis report," as it relates to identifying hydrologic site characteristics with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

- 10 CFR Part 100, "Reactor Site Criteria," as it relates to identifying and evaluating hydrological features of the site. The requirement to consider physical site characteristics in site evaluations is specified in 10 CFR 100.20(c).

- 10 CFR 100.23(d), "Geologic and seismic siting factors," sets forth the criteria to determine the siting factors for plant design bases with respect to water levels at the site.

The staff also used the following regulatory guides for the acceptance criteria identified in NUREG-0800, Section 2.4.5:

- RG 1.59, "Design Basis Floods for Nuclear Power Plants," supplemented by best current practices, as it relates to providing assurance that natural flooding phenomena that could potentially affect the site have been appropriately identified and characterized

- RG 1.102, "Flood Protection for Nuclear Power Plants," as it relates to providing assurance that SSCs important to safety have been designed to withstand the effects of natural flooding phenomena likely to occur at the site

- RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," as it relates to the contents of a COL application

2.4.5.4 *Technical Evaluation*

In this section of the report, the staff reviewed the applicant's assessment of the probable maximum surge and seiche flooding. The staff's independent analysis, where needed for the review, is also described.

Information Submitted by the Applicant

The applicant followed the regulatory guidance prescribed by RG 1.59, which describes the probable maximum surge and seiche based on a PMH, a PMWS, or a moving squall line. The applicant stated that RG 1.59 recommends consideration of a PMH for areas within 322 km (200 mi) of coastal areas and the WLS site is located approximately 282 km (175 mi) from the Atlantic Ocean on the southeast bank of the Broad River.

Following the recommendation for Folly Island in RG 1.59, the applicant estimated the PMH storm surge to be 8.1 m (26.5 ft) above MSL, which includes a 0.3 m (1.0 ft) sea level anomaly known to occur for predicted tides at Charleston, SC. The applicant also reported that the maximum storm surge along the Atlantic Coast since 1975, the most recent year for which data was included in RG 1.59, caused by hurricane Hugo, was approximately 6.1 m (20 ft) high.

Since the safety-related plant grade at 180.7 m (593 ft) above MSL is approximately 25 m (81.9 ft) higher than the normal water-surface elevation in the Broad River near the WLS site, (511.1 ft) above MSL, the applicant concluded that the PMH surge of 8.1 m (26.5 ft) would not flood the site, even if the surge were to translate to the site from the coast without any attenuation. The applicant also concluded that based on the location of the site and its elevation, safety-related facilities at the WLS site would not be affected by surge and seiche flooding.

The applicant stated resonance at natural periodicity, lake reflection, and harbor resonance are characteristics of harbors, estuaries, and large lakes and are generally not associated with river sites. The COL applicant did not present any specific descriptions and analyses of wave action during surge and seiche events.

The applicant used the USACE approach to estimate surge in Make-Up Ponds A and B under extreme winds. The COL applicant used the 100-year water-surface elevations, 169.5 m (556.1 ft) and 175.6 m (576.2 ft) above MSL for Make-Up Ponds A and B, respectively, as the initial elevations. Using the maximum wind speeds identified in WLS COL FSAR Section 2.3.1.2.8, the applicant estimated maximum crest-to-trough wave heights of 1.2 m (3.8 ft) and 2.1 m (6.9 ft) for Make-Up Ponds A and B, respectively. The total high-speed wind-wave activity for Make-Up Ponds A and B were 1.7 m (5.6 ft) and 2.4 m (7.8 ft), respectively. Therefore, the applicant estimated that the flood water-surface elevation in Make-Up Ponds A and B during extreme wind events would be 171.2 m (561.7 ft) and 178 m (583.9 ft) above MSL, respectively. The applicant concluded that the high-speed wind-wave activity would not affect the WLS site because the maximum water-surface elevations in Make-Up Ponds A and B during the extreme wind events would be below the safety-related plant grade of 180.7 m (593 ft) above MSL.

The applicant stated that Make-Up Pond C is located on a tributary of the Broad River, west of the WLS site. The applicant further stated that a postulated failure of the dam impounding Make-Up Pond C would release waters to the Broad River and not directly to the WLS site and therefore flooding from surges and seiches in Make-Up Pond C would be bounded by the postulated failure of Make-Up Pond C Dam described in WLS COL FSAR Section 2.4.4.

NRC Staff's Technical Evaluation

The staff reviewed COL Information Item 2.4-2 related to the provision of site-specific information about the PMF at the plant site included under WLS COL FSAR Section 2.4. Additional aspects of this information item are addressed in Sections 2.4.2. 2.4.3, 2.4.4, 2.4.6, 2.4.7, and 2.4.10 of this report. To ensure that the most conservative of plausible conceptual models has been identified, in RAI 823, Question 02.04.05-1, the staff requested that the applicant describe the process followed to determine the conceptual models for PMH, PMWS, seiche and resonance, wave runup, and sediment erosion and deposition.

In an October27, 2008, response to RAI 823, Question 02.04.05-1, the applicant stated that the conceptual models to determine floods from PMH, PMWS, seiche and resonance, and wave runup are consistent with the guidance of RG 1.206 and RG 1.59 and the estimation methods used conform to the guidance provided by ANSI/ANS-2.8-1992. The applicant stated that the maximum hurricane storm surge produced on the Atlantic Coast was transposed to the site without accounting for the travel distance and the presence of any instream structures. The applicant stated that maximum wind speeds were used in the determination of wind-wave effects for water bodies adjacent to the site. The applicant stated that the resulting flood waves did not exceed the water-surface elevation produced by the design basis flood.

The applicant also stated that the natural fundamental periods of oscillation of the water bodies adjacent to the site are determined to be significantly shorter than meteorologically induced wave periods. The applicant concluded, therefore, that there is no potential for resonance of the seiche in the water bodies near the site that could result in flooding at the site. The applicant further stated that no safety-related water is required by the proposed units from the Broad River or from Make-Up Ponds A and B. The applicant concluded, therefore, that no safety-related SSCs of the proposed units would be affected by sediment deposition or erosion.

The staff evaluated the applicant's October 27, 2008, response to RAI 823, Question 02.04.05-1, and concluded that the approach adopted by the applicant in the evaluation of storm surge and seiche hazards at the site is appropriate and sufficiently conservative. Accordingly, the staff considers RAI 823, Question 02.04.05-1, resolved.

Seiches can be caused in lakes and reservoirs by meteorological or seismic forcing. To review the effects of meteorologically or seismically induced seiches near the WLS site, in RAI 823, Question 02.04.05-1, the staff requested that the applicant provide an assessment of meteorologically and seismically induced seiches in Make-Up Ponds A and B. In and October 27, 2008, response to RAI 823, Question 02.04.05-2, the applicant stated that an assessment of meteorologically induced waves in Make-Up Ponds A and B is carried out by estimating the wave periods for high-speed wind waves and the natural fundamental periods of oscillation for the two ponds. The applicant stated that the natural fundamental periods of oscillation of Make-Up Ponds A and B are 2.7 and 8 minutes, respectively. The applicant also reported that the wave periods of high-speed wind waves for Make-Up Ponds A and B, estimated coincident with a 100-year water-surface elevation within the ponds, are 1.8 and 2.7 seconds, respectively. The applicant stated that the wave periods are much shorter than the natural fundamental period of oscillation of the two ponds.

The applicant further stated that the natural fundamental periods of oscillation of the two ponds are significantly shorter than meteorologically induced wave periods such as synoptic storm pattern frequency and dramatic reversal in steady wind direction. The applicant concluded therefore, that a meteorologically induced seiche would not be set up in the two makeup ponds. The applicant also considered the possibility of seismically induced seiches in Make-Up Ponds A and B in response to the staff's RAI 823, Question 02.04.05-2, but provided the details with its response to the staff's RAI 824, Question 02.04.06-2,. The applicant stated that there are no capable tectonic sources within a 25 mile radius of the site as described in WLS COL FSAR Section 2.5.3. The applicant concluded, therefore, that a seismically generated seiche in Make-Up Ponds A and B is unlikely.

The staff evaluated the October 27, 2008, response to RAI 823, Question 02.04.05-2, and updates to the WLS COL FSAR up to and including Revision 7 to determine that the methods used by the applicant to postulate seiching mechanisms and resulting water-surface elevations near the site are adequate. The staff agreed with the applicant that if the natural fundamental periods of oscillation of the two ponds are significantly different from those that can be induced by meteorological events, that meteorologically induced seiches in the two ponds are not likely. The staff agreed with the applicant that in absence of a nearby capable tectonic source, a seismically generated seiche in the makeup ponds is unlikely.

The staff concluded that, due to the location of the site, approximately 282 km (175 mi) inland from the Atlantic Coast any hurricane-induced storm surge in the Atlantic Ocean would dissipate before reaching the site and, therefore, would not affect the site. Due to of the location of Make-Up Pond C on a tributary of the Broad River away from the WLS site, the staff also agreed with the applicant that surge and seiche flooding in the pond would be bounded by the flood generated by a postulated failure of its dam. Accordingly, the staff considers RAI 823, Question 02.04.05-2, resolved.

Based on a review of the applicant's information in WLS COL FSAR Revision 11, the staff concluded that the applicant has appropriately analyzed flood water-surface elevations caused by surge and seiche near the WLS site using approaches currently used in standard practice. Therefore, based on the reasons given above, the staff concluded that the applicant adequately identified and evaluated extreme flood events at the site. The staff agreed that the surge and seiche flood water-surface elevations estimated by the applicant are appropriate for the WLS site. Therefore, based on the reasons given above, the staff concluded that the applicant adequately determined hydrologic characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

2.4.5.5 *Post Combined License Activities*

There are no post COL activities related to this section

2.4.5.6 *Conclusion*

The staff reviewed the application and confirmed that the applicant has addressed the information relevant to probable maximum surge and seiche flooding, and that there is no outstanding information required to be addressed in the WLS COL FSAR related to this section/

As set forth above, the applicant has presented and substantiated information to establish the site description. The staff reviewed the information provided and, for the reasons given above, concluded that the applicant has provided sufficient details about the site description to allow the staff to evaluate, as documented in Section 2.4.5, herein, whether the applicant has met the relevant requirements of 10 CFR 52.79(a)(1) and 10 CFR Part 100 with respect to determining the acceptability of the site. This addressed the probable maximum surge and seiche flooding in WLS COL Information Item 2.4-2. The staff concludes that the applicant provided sufficient information on flood hazards related to storm surge and seiches to satisfy the applicable requirements of 10 CFR Part 52 and 10 CFR Part 100.

2.4.6 Probable Maximum Tsunami Hazards

2.4.6.1 *Introduction*

WLS COL FSAR Section 2.4.6 describes probable maximum tsunami hazards to ensure that any potential tsunami hazard to the safety-related SSCs at the proposed site has been considered in compliance with NRC regulations.

Section 2.4.6 of this report presents an evaluation of the following topics based on data provided by the applicant in the WLS COL FSAR and information available from other sources: (1) historical tsunami data; (2) probable maximum tsunami (PMT) that may pose hazards to the site; and (3) any additional information requirements prescribed in the “Contents of Application” sections of the applicable subparts to 10 CFR Part 52.

2.4.6.2 *Summary of Application*

This section of the WLS COL FSAR addresses the site-specific information about PMT hazards in terms of effects on structures and water supply. The applicant addressed the information item identified in AP1000 DCD Tier 2, Section 2.4.1.2, Revision 19 related to PMT hazards as follows:

AP1000 COL Information Item

- WLS COL 2.4-2

In addition, this section of the WLS COL FSAR addresses the following COL-specific information identified in AP1000 DCD Section 2.4.1.2, Revision 19.

> Combined License applicants referencing the AP1000 certified design will address the following site-specific information about historical flooding and potential flooding factors, including the effects of local intense precipitation.
>
> - PMF on Streams and Rivers – Site-specific information that will be used to determine the design basis flooding at the site. This information will include the PMF on streams and rivers.
> - Dam Failures – Site-specific information about potential dam failures.
> - Probable Maximum Surge and Seiche Flooding – Site-specific information about probable maximum surge and seiche flooding.
> - PMT Loading – Site-specific information about PMT loading.
> - Flood Protection Requirements – Site-specific information about flood protection requirements or verification that flood protection is not required to meet the site parameter for flood level.

In WLS COL FSAR Section 2.4.6, the applicant addressed the effects of floods caused by PMT. Other causes of floods and their effects are discussed in related WLS COL FSAR sections. No further action is required for sites within the bounds of the site parameter for flood level.

2.4.6.3 *Regulatory Basis*

The relevant requirements of NRC regulations on consideration of the effects of PMT hazards, and the associated acceptance criteria, are described in NUREG-0800, Section 2.4.6.

The applicable regulatory requirements for tsunami hazards are as follows:

- 10 CFR 52.79(a)(1)(iii), "Contents of applications; technical information in final safety analysis report," as it relates to identifying hydrologic site characteristics with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.
- 10 CFR Part 100, "Reactor Site Criteria," as it relates to identifying and evaluating hydrological features of the site. The requirement to consider physical site characteristics in site evaluations is specified in 10 CFR 100.20(c).
- 10 CFR 100.23(d), "Geologic and seismic siting factors," sets forth the criteria to determine the siting factors for plant design bases with respect to water levels at the site.

The staff also used the following regulatory guides for the acceptance criteria identified in NUREG-0800, Section 2.4.6:

- RG 1.59, “Design Basis Floods for Nuclear Power Plants,” supplemented by best current practices, as it relates to providing assurance that natural flooding phenomena that could potentially affect the site have been appropriately identified and characterized

- RG 1.102, “Flood Protection for Nuclear Power Plants,” as it relates to providing assurance that SSCs important to safety have been designed to withstand the effects of natural flooding phenomena likely to occur at the site

- RG 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition),” as it relates to the contents of a COL application

2.4.6.4 *Technical Evaluation*

In this section of the report, the staff reviewed the applicant’s analysis of PMT. The staff’s independent analysis, where needed for the review, is also described.

Information Submitted by the Applicant

The applicant stated that no specific tsunami hazard maps are available for the east coast of the United States. The applicant used a general tsunami risk map developed by the USACE to identify a tsunami wave height of 1.5 m (5 ft) for the east coast of the United States. The applicant reported that NOAA tsunami database includes a maximum recorded tsunami height of 6.1 m (20 ft) at Daytona Beach, FL, on July 3, 1992, which was probably meteorologically induced. The applicant stated that the WLS site is located approximately 282 km (175 mi) inland from the Atlantic Ocean and the safety-related plant elevation is 180.7 m (593 ft) above MSL. Based on historical tsunami data and the location and elevation of the WLS site, the applicant concluded that safety-related facilities would not be affected by tsunami flooding.

The applicant stated that hill-slope failure-induced landslides that generate significant waves in Make-Up Ponds A and B are implausible. The applicant stated that field investigations reported in WLS COL FSAR Section 2.5.1.1 noted no irregular weathering conditions or natural landslide hazards and there is no documented evidence of significant landslides at the WLS site or adjacent to the Make-Up Ponds. The applicant stated that landslides of limited size in shallow soil or fill may occur, but would be of insufficient volume and would occur at too low of velocities to cause any significant water waves. The applicant further stated that the slopes around Make-Up Ponds A and B are either natural and have existed since the Holocene age or resulted from cut and fill during the Cherokee Nuclear Station construction. The applicant stated that these slopes are stable and show no visual evidence of groundwater seepage, past failures, or movement or creep.

The applicant stated that because there are no capable tectonic sources within 40 km (25 mi) of the WLS site, surface fault ruptures from seismic waves are not plausible. The applicant concluded that seismically generated water waves in Make-Up Ponds A and B would be insignificant compared to the freeboard in these ponds.

The applicant stated that Make-Up Pond C is located on a tributary of the Broad River, west of the WLS site. The applicant further stated that a postulated failure of the dam impounding Make-Up Pond C would release waters to the Broad River and not directly to the WLS site and, therefore, flooding from seismically induced water waves in Make-Up Pond C would be bounded by the postulated failure of Make-Up Pond C Dam described in WLS COL FSAR Section 2.4.4.

NRC Staff's Technical Evaluation

The staff reviewed COL Information Item 2.4-2 related to the provision of site-specific information about the PMF at the plant site included under WLS COL FSAR Section 2.4. Additional aspects of this information item are addressed in Sections 2.4.2. 2.4.3, 2.4.4, 2.4.5, 2.4.7, and 2.4.10, herein.

To ensure that the most conservative of plausible conceptual models have been identified, in RAI 824, Question 02.04.06-1, the staff requested that the applicant describe the process followed to determine the conceptual models for PMT, tsunami propagation, wave runup, inundation and drawdown, hydrostatic and hydrodynamic forces, debris and water-borne projectiles, and sediment erosion and deposition. In an October 27, 2008, response to RAI 824, Question 02.04.06-1, the applicant stated that the conceptual models used to determine flood waves generated by PMT and other tsunami-like waves follow the requirements of RG 1.206 and RG 1.59. The applicant compared the historical maximum recorded tsunami wave heights and the maximum wave heights reported in a tsunami risk map for the east coast of the U.S. to the available freeboard of the WLS site above the Broad River. The applicant concluded that the resulting flood waves are less than the design-basis flood. Therefore, the applicant concluded that there is no potential for inundation, hydrostatic and hydrodynamic forces, debris, or water-borne projectiles that may affect safety-related facilities.

The applicant also stated that there are no capable tectonic sources in the vicinity of the site as described in WLS COL FSAR Section 2.5.3. The applicant concluded, therefore, that a seismically induced water wave near the site is not plausible. The applicant further stated that there are no irregular conditions or natural landslide hazards in the vicinity of the site. The applicant concluded, therefore, that landslide-induced tsunami-like waves near the WLS site are not plausible.

Based on a review of the applicant's October 27, 2008, response to RAI 824, Question 02.04.06-1, the staff concluded that the process used to determine the conceptual models for tsunamis and tsunami-like waves in the vicinity of the WLS site is adequately described. Accordingly, the staff considers RAI 824, Question 02.04.06-1, resolved. The staff used this information in its safety review combined with observations made during the site audit.

The staff performed a search of the NOAA National Geophysical Data Center (NGDC) Historical Tsunami Database for tsunami runup events reported on the east coast of the U.S. The maximum runup reported in the database on the east coast of the United States is 6.0 m (19.7 ft) above MSL at Daytona Beach, FL, on July 3, 1992, due to a meteorological cause. The staff concluded, based on its independent search and the applicant's RAI responses that credible seismic and landslide tsunamigenic sources in the vicinity of the site do not exist. The staff also determined that the site is located approximately 282 km (175 mi) from the Atlantic Ocean. For a tsunami generated by a near or a far-field oceanic source to affect the site, the runup would need to travel this distance, 282 km (175 mi), inland from the coast. Historically, maximum horizontal extent of inundation is reported to be less than approximately 8.1 km (5 mi). Since the site is located at least one order of magnitude farther than the maximum horizontal inland distance reported for historical tsunamis, the staff concluded that a PMT in the Atlantic Ocean would not pose a hazard at the site.

In RAI 824, Question 02.04.06-2, the staff requested that the applicant provide an assessment of landslide and slope-failure potential on the shores of Make-Up Ponds A and B and an assessment of tsunami-like waves that may be generated by the potential landslides or slope failures in these ponds. In an October 27, 2008, response to RAI 824, Question 02.04.06-2, the applicant stated that no irregular weathering or natural landslides were identified during field investigations in the region. Therefore, the applicant concluded that landslide-generated tsunami-like waves are not plausible for Make-Up Ponds A and B.

The staff determined that there are no active or passive volcanoes located near the site the eruption of and resulting pyroclastic flows from which may generate a tsunami-like wave in any water bodies near the site. The staff evaluated the applicant’s information in the WLS COL FSAR and concluded that the process used by the applicant to estimate source generator characteristics with respect to landslide-generated tsunami-like waves is adequate. The staff also agreed with the applicant’s assessment that a landslide-generated tsunami-like wave in the water bodies near the site is not likely. The staff concluded that a tsunami or a tsunami-like wave in the vicinity of the site is an unlikely event. Therefore, the staff determined that a more detailed tsunami analysis is not needed. Based on its review, the staff considers RAI 824, Question 02.04.06-2, resolved.

Based on a review of the applicant’s information in WLS COL FSAR Revision 11, the staff concluded that the applicant has appropriately analyzed the potential for floods caused by tsunamis near the WLS site using approaches currently used in standard practice. Therefore, based on the reasons given above, the staff concluded that the applicant adequately identified and evaluated extreme flood events at the site. Therefore, based on the reasons given above, the staff concluded that the applicant adequately determined hydrologic characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

2.4.6.5 *Post Combined License Activities*

There are no post COL activities related to this section.

2.4.6.6 *Conclusion*

The staff reviewed the application and confirmed that the applicant has addressed the information relevant to PMT hazards, and that there is no outstanding information required to be addressed in the WLS COL FSAR related to this section.

As set forth above, the applicant has presented and substantiated information to establish the site description. The staff reviewed the information provided and, for the reasons given above, concluded that the applicant has provided sufficient details about the site description to allow the staff to evaluate, as documented in Section 2.4.6 of this report, whether the applicant has met the relevant requirements of 10 CFR 52.79(a) (1) and 10 CFR Part 100 with respect to determining the acceptability of the site. This addressed the probable maximum tsunami hazard in COL Information Item 2.4-2. The staff concludes that the applicant has provided sufficient information on PMT hazards to satisfy the applicable requirements of 10 CFR Part 52 and 10 CFR Part 100.

2.4.7 Ice Effects

2.4.7.1 *Introduction*

WLS COL FSAR Section 2.4.7 describes ice effects to ensure that safety-related facilities and water supply are not affected by ice-induced hazards.

Section 2.4.7 of this report presents an evaluation of the following topics based on data provided by the applicant in the WLS COL FSAR and information available from other sources: (1) regional history and types of historical ice accumulations (i.e., ice jams, wind-driven ice ridges, floes, frazil ice formation, etc.); (2) potential effects of ice-induced, high- or low-flow levels on safety-related facilities and water supplies; (3) potential effects of ice to produce forces on, or cause blockage of, safety-related facilities; and (4) any additional information requirements prescribed in the “Contents of Application” sections of the applicable subparts to 10 CFR Part 52.

As stated in Section 2.4 above, hydrologic characteristics associated with conditions that would result in a loss of external water supply and seismic design considerations of water-supply structures are not relevant for the AP1000 design. Therefore, low-water conditions related to topics listed above were not part of the staff’s review.

2.4.7.2 *Summary of Application*

This section of the WLS COL FSAR addresses the site-specific information on ice effects. The applicant addressed the information item identified in AP1000 DCD Tier 2, Section 2.4.1.2, Revision 19 related to ice effects as follows:

AP1000 COL Information Item

- WLS COL 2.4-2

In addition, this section addresses the following COL-specific information identified in AP1000 DCD Section 2.4.1.2, Revision 19:

> Combined License applicants referencing the AP1000 certified design will address the following site-specific information about historical flooding and potential flooding factors, including the effects of local intense precipitation.

- Probable Maximum Flood (PMF) on Streams and Rivers – Site-specific information that will be used to determine the design basis flooding at the site. This information will include the PMF on streams and rivers.

- Dam Failures – Site-specific information about potential dam failures.

- Probable Maximum Surge and Seiche Flooding – Site-specific information about probable maximum surge and seiche flooding.

- Probable Maximum Tsunami Loading – Site-specific information about probable maximum tsunami loading.

- Flood Protection Requirements – Site-specific information about flood protection requirements or verification that flood protection is not required to meet the site parameter for flood level.

In WLS COL FSAR Section 2.4.7, the applicant addressed ice effects on high water at the site. Other causes of floods and their effects were discussed in subsequent WLS COL FSAR sections. No further action is required for sites within the bounds of the site parameter for flood level.

2.4.7.3 *Regulatory Basis*

The relevant requirements of NRC regulations for the identification and evaluation of ice effects, and the associated acceptance criteria, are described in NUREG—800, Section 2.4.7.

The applicable regulatory requirements for identifying ice effects are as follows:

- 10 CFR 52.79(a)(1)(iii), "Contents of applications; technical information in final safety analysis report," as it relates to identifying hydrologic site characteristics with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

- 10 CFR Part 100, "Reactor Site Criteria," as it relates to identifying and evaluating hydrological features of the site. The requirement to consider physical site characteristics in site evaluations is specified in 10 CFR 100.20(c).

- 10 CFR 100.23(d), "Geologic and seismic siting factors," sets forth the criteria to determine the siting factors for plant design bases with respect to water levels at the site.

The staff also used the following regulatory guides for the acceptance criteria identified in NUREG-0800, Section 2.4.7:

- RG 1.59, "Design Basis Floods for Nuclear Power Plants," supplemented by best current practices, as it relates to providing assurance that natural flooding phenomena that could potentially affect the site have been appropriately identified and characterized

- RG 1.102, “Flood Protection for Nuclear Power Plants,” as it relates to providing assurance that SSCs important to safety have been designed to withstand the effects of natural flooding phenomena likely to occur at the site

- RG 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition),” as it relates to the contents of a COL application

2.4.7.4 *Technical Evaluation*

In this section of the report, the staff reviewed the applicant’s analyses regarding ice effects. The staff’s independent analysis, where needed for the review, is also described.

Information Submitted by the Applicant

The applicant reported that winter water temperatures range from 0 to 9 ºC (32 to 48.2 ºF) between 1962 and 1981 at 10 USGS gauging stations on the Broad River and tributaries upstream of the WLS site. The applicant also reported that the lowest recorded water temperature near the WLS site from 1959 to 2005 is 2 ºC (35.6 ºF) according to the EPA STORET database and the values vary from 1 to 4 ºC (33.8 to 39.2 ºF) from 1995 to 2000 as stated in the measurements by the North Carolina Department of Environmental and Natural Resources at nine stations located near the 10 USGS stations. Based on these water temperature records, the applicant concluded that historical observations suggest the Broad River water temperatures consistently remain above freezing. The applicant also concluded that flooding of the WLS site from an ice jam is a remote possibility based on historical data from the USACE Cold Regions Research and Engineering Laboratory historical ice jam database. Since no safety-related water storage is required for WLS, the applicant concluded that ice-induced low flow would not affect its safety.

NRC Staff’s Technical Evaluation

The staff reviewed COL Information Item 2.4-2 related to the provision of site-specific information about the PMF at the plant site included under WLS COL FSAR Section 2.4. Additional aspects of this information item are addressed in Sections 2.4.2. 2.4.3, 2.4.4, 2.4.5, 2.4.6, and 2.4.10 of this report.

The staff searched the NOAA National Climatic Data Center (NCDC) Storm Events database to search for snow and ice events for the counties in which the majority of the Upper Broad River Basin lies. This search yielded the storm and ice events listed in Table 2.4.7-1 of this report.

Table 2.4.7-1 Snow and Ice Events in the Upper Broad River Basin and Its Adjoining Areas Between 1993 and 2008

County	Number of Snow and Ice Events	Number of Ice Storms
Buncombe Co., NC	144	8
Henderson Co., NC	87	13
Polk Co., NC	50	7
McDowell Co., NC	79	12
Rutherford Co., NC	51	6
Cleveland Co., NC	44	7
Gaston Co., NC	37	8
Lincoln Co., NC	43	9
Cherokee Co., SC	50	10

Based on the storm and ice events in the Upper Broad River Basin, the staff determined that ice events are frequent during winters near the WLS site. The staff also searched the USACE ice jam database for ice jam or ice dam formation on the Broad River. No historical records exist of ice jams or dams on the Broad River. As stated in the USACE ice jam database, there are only two records of ice jam or ice dam formation in the State of North Carolina. An ice jam formed in the Neuse River upstream of the USGS streamflow gauge at Kinston, NC, on January 26, 1940, that lasted for 4 days. An ice gorge was reported in the Missouri River at Williston, NC, on February 24, 1925. The ice jam database does not list any events for the State of South Carolina.

Based on the search of the USACE ice jam database, the staff concluded that formation of ice jams and ice dams has not been reported for the Broad River. Therefore, the staff concluded that ice dams and ice jams are not a credible hazard near the WLS site.

The staff downloaded air temperature data for three cooperative stations located near the WLS site from NOAA NCDC. These three stations are Shelby 2 NNE, Gastonia, and Ninety-Nine Islands. The period of record at these stations are 1893−1895 and 1936-present (Shelby), 1930-present (Gastonia), and 1960-present (Ninety-Nine Islands). Using this air temperature data, the staff performed independent analysis to determine some characteristics of minimum daily mean air temperature near the site. These characteristics for the three stations are shown in Table 2.4.7-2 of this report.

Table 2.4.7-2 Some Characteristics of Mean Daily Air Temperature at NOAA NCDC Cooperative Stations Near the Site

Characteristic	Shelby 2 NNE	Gastonia	Ninety-Nine Islands
Minimum daily mean air temperature (date)	-13ºC (8ºF) (02/08/1895)	-12ºC (11ºF) (01/21/1985)	-12ºC (11ºF) (01/21/1985)
Number of days below freezing (total number of days for which data are available)	768 (27,818)	1076 (26,678)	740 (17,566)
Maximum number of consecutive days daily mean air temperature remains at freezing or below	13	11	9
Maximum number of consecutive days daily mean air temperature remains at 18ºF or below	3	2	2

The minimum daily mean air temperature at Shelby, Gastonia, and the Ninety-Nine Islands stations are -13ºC, -12ºC, -12ºC (8ºF, 11ºF, and 11ºF), respectively. The average number of days per year that the stations' daily mean air temperature was below freezing are approximately 10, 15, and 15, respectively. The longest sequences of days that the daily mean air temperature was at or below freezing for the three stations are 13, 11, and 9 days for Shelby, Gastonia, and Ninety-Nine Islands stations, respectively. Based on these characteristics of the daily mean air temperature, the staff concluded that the air temperature near the WLS site can fall below freezing for moderately long periods of time. However, the sequences are not long enough to cause extensive freezing of water bodies near the WLS site.

The staff also concluded the length of sequences of days during which the daily mean air temperature remained at or below -8ºC (18ºF). The reason for choosing the threshold of -8ºC (18ºF) is that frazil ice forms in turbulent, non-snow covered waters that undergo supercooling when the air temperature falls to or below -8ºC (18ºF). The staff noted that at Gastonia and Ninety-Nine Islands stations, the maximum span of such sequences was just 2 days and at Shelby it was 3 days. Based on this data, the staff concluded that frazil ice formation at and near the WLS site is possible, although unlikely, given the short duration during which supercooling can occur. However, the proposed reactors at the WLS site would not depend on any external source of water supply for safe shutdown. Therefore, the staff concluded that the safety of the plants will not be affected by formation of frazil ice at and near the WLS site.

Based on a review of the applicant's information in WLS COL FSAR Revision 11, the staff concluded that the applicant has appropriately analyzed the potential for ice effects near the WLS site using approaches currently used in standard practice. Therefore, based on the reasons given above, the staff concluded that the applicant adequately identified and evaluated extreme flood events at the site. Therefore, based on the reasons given above, the staff concluded that the applicant adequately determined hydrologic characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

2.4.7.5 *Post Combined License Activities*

There are no post COL activities related to this section.

2.4.7.6 *Conclusion*

The staff reviewed the application and confirmed that the applicant has addressed site characteristics and other hydrometeorological parameters related to ice formation at or near the plant site, and that there is no outstanding information required to be addressed in the WLS COL FSAR related to this section.

As set forth above, the applicant has presented and substantiated information to establish the site description. The staff reviewed the information provided and, for the reasons given above, concluded that the applicant has provided sufficient details about the site description to allow the staff to evaluate, as documented in Section 2.4.7 of this report, whether the applicant has met the relevant requirements of 10 CFR 52.79(a)(1) and 10 CFR Part 100 with respect to determining the acceptability of the site. This addressed ice effects in COL Information Item 2.4-2. The staff concludes that the applicant has provided sufficient information on flood hazards from ice effects to satisfy the applicable requirements of 10 CFR Part 52 and 10 CFR Part 100.

2.4.8 Cooling Water Canals and Reservoirs

2.4.8.1 *Introduction*

WLS COL FSAR Section 2.4.8 describes cooling-water supply, made up of canals and reservoirs, used to transport and impound water supplied to the safety-related SSCs.

Section 2.4.8 of this report presents an evaluation of the applicant's submittal related to cooling-water canals and reservoirs. As stated in Section 2.4 above, hydrologic characteristics associated with conditions that would result in a loss of external water-supply and seismic design considerations of water-supply structures are not relevant for the AP1000 design. Therefore, flooding and low-water conditions for canals and reservoirs were not part of the staff's review.

2.4.8.2 *Summary of Application*

This section of the WLS COL FSAR addresses the site-specific information on cooling-water canals and reservoirs. The applicant addressed the information item identified in AP1000 DCD Tier 2, Section 2.4.1.3, Revision 19 related to cooling-water canals and reservoirs as follows:

AP1000 COL Information Item

- WLS COL 2.4-3

In addition, this section addresses the following COL-specific information identified in AP1000 DCD Tier 2, Section 2.4.1.3, Revision 19.

> Combined License applicants will address the water supply sources to provide makeup water to the service water system cooling tower.

2.4.8.3 *Regulatory Basis*

The relevant requirements of NRC regulations for the identification of design considerations for cooling-water canals and reservoirs, and the associated acceptance criteria, are described in NUREG-0800, Section 2.4.8.

The applicable regulatory requirements for cooling-water canals and reservoirs are as follows:

- 10 CFR 52.79(a)(1)(iii), "Contents of applications; technical information in final safety analysis report," as it relates to identifying hydrologic site characteristics with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

- 10 CFR Part 100, "Reactor Site Criteria," as it relates to identifying and evaluating hydrological features of the site. The requirement to consider physical site characteristics in site evaluations is specified in 10 CFR 100.20(c).

- 10 CFR 100.23(d), "Geologic and seismic siting factors," sets forth the criteria to determine the siting factors for plant design bases with respect to water levels at the site.

The staff also used the following regulatory guides for the acceptance criteria identified in NUREG-0800, Section 2.4.8:

- RG 1.59, "Design Basis Floods for Nuclear Power Plants," supplemented by best current practices, as it relates to providing assurance that natural flooding phenomena that could potentially affect the site have been appropriately identified and characterized

- RG1.102, "Flood Protection for Nuclear Power Plants" as it relates to providing assurance that SSCs important to safety have been designed to withstand the effects of natural flooding phenomena likely to occur at the site

- RG1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," as it relates to the contents of a COL application

2.4.8.4 *Technical Evaluation*

In this section of the report, the staff reviewed the applicant's analyses related to cooling-water canals and reservoirs. The staff's independent analysis, where needed for the review, is also described.

Information Submitted by the Applicant

The applicant stated that there are no safety-related cooling-water canals or reservoirs proposed for the WLS site because the UHS is provided by the atmosphere for the proposed AP1000 units.

NRC Staff's Technical Evaluation

The staff reviewed COL Information Item 2.4-3 related to the provision of site-specific information about the cooling-water canals and reservoirs at the plant site included under WLS COL FSAR Section 2.4. Additional aspects of this information item are addressed in Sections 2.4.9. 2.4.11, and 2.4.12 of this report.

The staff reviewed the functioning of the AP1000 UHS. The passive cooling system of the proposed units is assisted by a water spray on the containment vessel provided by a passive containment cooling-water storage tank located on top of the containment building, which holds a 3-day supply of water following a design basis accident. Additional water is stored in a passive containment cooling ancillary water storage tank for an additional 4 days. Technical specifications would ensure that this 7-day supply of water to assist in cooling is always available. The proposed reactors at the WLS site would not depend on any external source of water supply for safe shutdown.

The staff concluded that there are no safety-related cooling-water canals or reservoirs are required for the safe operation of the WLS and, therefore, no further evaluation of safety-related water canals and reservoirs are necessary.

Based on a review of the applicant's information in WLS COL FSAR Revision 11, the staff concluded that the applicant has appropriately analyzed the potential for floods caused by cooling-water canals and reservoirs near the WLS site. Therefore, based on the reasons given above, the staff concluded that the applicant adequately identified and evaluated extreme flood events at the site. Therefore, based on the reasons given above, the staff concluded that the applicant adequately determined hydrologic characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

2.4.8.5 *Post Combined License Activities*

There are no post COL activities related to this section.

2.4.8.6 *Conclusion*

The staff reviewed the application and confirmed that the applicant has addressed the information relevant to design basis for canal and reservoirs used to transport and impound water supplied to the SSCs, and that there is no outstanding information required to be addressed in the WLS COL FSAR related to this section.

As set forth above, the applicant has presented and substantiated information relative to the design bases of canals and reservoirs important to the design and siting of the plant. The staff reviewed the information provided and, for the reasons given above, concluded that the applicant has provided sufficient details about the site description to allow the staff to evaluate, as documented in Section 2.4.8 of this report, whether the applicant has met the relevant requirements of 10 CFR 52.79, and 10 CFR Part 100, with respect to determining the acceptability of the site. The information provided on the cooling-water canals and reservoirs addressed COL Information Item 2.4-3. The staff concludes that the applicant has provided sufficient information to satisfy the applicable requirements of 10 CFR Part 52 and 10 CFR Part 100.

2.4.9 Channel Diversions

2.4.9.1 *Introduction*

WLS COL FSAR Section 2.4.9 describes channel diversions. It includes hydrogeologic and geomorphologic descriptions of the Broad River Basin and an evaluation of the likelihood of diversion of Broad River away from its present course.

Section 2.4.9 of this report presents the staff's evaluation of the applicant's submittal related to channel diversions. As stated in Section 2.4 above, hydrologic characteristics associated with conditions that would result in a loss of external water supply and seismic design considerations of water-supply structures are not relevant for the AP1000 design. Therefore, low-water conditions from channel diversions were not part of the staff's review.

2.4.9.2 *Summary of Application*

This section of the WLS COL FSAR addresses information about site-specific channel diversions. The applicant addressed the information item identified in AP1000 DCD Tier 2, Section 2.4.1.3, Revision 19 related to channel diversions as follows:

AP1000 COL Information Item

- WLS COL 2.4-3

In addition, this section addresses the following COL-specific information identified in AP1000 DCD Tier 2, Section 2.4.1.3, Revision 19.

> Combined License applicants will address the water supply sources to provide makeup water to the service water system cooling tower.

2.4.9.3 *Regulatory Basis*

The relevant requirements of NRC regulations for the identification and evaluation of channel diversions, and the associated acceptance criteria, are described in NUREG-0800, Section 2.4.9.

The applicable regulatory requirements for identifying and evaluating channel diversions are as follows:

- 10 CFR 52.79(a)(1)(iii), "Contents of applications; technical information in final safety analysis report," as it relates to identifying hydrologic site characteristics with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

- 10 CFR Part 100, "Reactor Site Criteria," as it relates to identifying and evaluating hydrological features of the site. The requirement to consider physical site characteristics in site evaluations is specified in 10 CFR 100.20(c).

- 10 CFR 100.23(d), "Geologic and seismic siting factors," sets forth the criteria to determine the siting factors for plant design bases with respect to water levels at the site.

The staff also used the following regulatory guides for the acceptance criteria identified in NUREG-0800, Section 2.4.9:

- RG 1.59, "Design Basis Floods for Nuclear Power Plants," supplemented by best current practices, as it relates to providing assurance that natural flooding phenomena that could potentially affect the site have been appropriately identified and characterized

- RG 1.102, "Flood Protection for Nuclear Power Plants," as it relates to providing assurance that SSCs important to safety have been designed to withstand the effects of natural flooding phenomena likely to occur at the site

- RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," as it relates to the contents of a COL application

2.4.9.4 *Technical Evaluation*

In this section of the report, the staff reviewed the applicant's analyses related to channel diversions. The staff's independent analysis, where needed for the review, is also described.

Information Submitted by the Applicant

The applicant stated that there is no evidence to suggest diversions or realignments of the Broad River in the past, no unstable steep side slopes are present and there is no record of ice-induced channel diversions. The applicant noted that several shoals are located in the basin but are confined within the natural banks of the river. The applicant concluded that channel diversion due to geothermal activity is not expected in the region. The applicant also concluded that any potential channel diversion would not affect safety-related structures or systems at the WLS site because the passive cooling system for the AP1000 design uses the atmosphere as the UHS and the cooling system does not rely directly on the Broad River for water.

NRC Staff's Technical Evaluation

The staff reviewed COL Information Item 2.4-3 related to the provision of site-specific information about the channel diversions at the plant site included under WLS COL FSAR Section 2.4. Additional aspects of this information item are addressed in Sections 2.4.8, 2.4.11 and 2.4.12 of this report.

Based on the applicant's description of the cooling system of the proposed units at the WLS site, the staff concluded that the Broad River would serve as the source for makeup water for normal plant operation of the proposed units. The staff reviewed the functioning of the AP1000 UHS. The passive cooling system of the proposed units is assisted by a water spray on the containment vessel provided by a passive containment cooling-water storage tank located on top of the containment building, which holds a 3-day supply of water following a design basis accident. Additional water is stored in a passive containment cooling ancillary water storage tank for an additional 4 days. Technical specifications would ensure that this 7-day supply of water to assist in cooling is always available. The proposed reactors at the WLS site would not depend on any external source of water supply for safe shutdown.

The staff concluded that no safety-related cooling-water sources required for the safe operation of the WLS. Therefore, the staff determined that a diversion of the Broad River away from the site for any reason would not affect the safety of the plant. Based on the applicant's description of the geology and geomorphology of the Broad River Basin and the available freeboard between normal water-surface elevation in the river and the grade elevation of the WLS site, the staff also determined that a diversion of the Broad River toward the WLS site is unlikely. Therefore, the staff concluded that flooding of the WLS site from a diversion of the Broad River toward the WLS site is not a plausible hazard.

Based on a review of the applicant's information in WLS COL FSAR Revision 11, the staff concluded that the applicant has appropriately analyzed the potential for channel diversions near the WLS site. Therefore, based on the reasons given above, the staff concluded that the applicant adequately identified and evaluated extreme flood events at the site. Therefore, based on the reasons given above, the staff concluded that the applicant adequately determined hydrologic characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

2.4.9.5 *Post Combined License Activities*

There are no post COL activities related to this section

2.4.9.6 *Conclusion*

The staff reviewed the application and confirmed that the applicant has addressed the information to demonstrate that the characteristics of the site fall within the site parameters specified in the AP1000 DC rule, and that there is no outstanding information required to be addressed in the WLS COL FSAR related to this section.

As set forth above, the applicant has presented and substantiated information to establish the site description ensuring that the plant and essential water supplies will not be adversely affected. The staff reviewed the information provided and, for the reasons given above, concluded that the applicant has provided sufficient details about the site description to allow the staff to evaluate, as documented in Section 2.4.9 of this report, whether the applicant has met the relevant requirements of 10 CFR 52.79(a)(1) and 10 CFR Part 100 with respect to determining the acceptability of the site. This section addressed channel diversions in COL Information Item 2.4-3. The staff concludes that the applicant has provided sufficient information to satisfy the applicable requirements of 10 CFR Part 52 and 10 CFR Part 100.

2.4.10 Flooding Protection Requirements

2.4.10.1 *Introduction*

WLS COL FSAR Section 2.4.10 describes locations and elevations of safety-related facilities and those of structures and components required for protection of safety-related facilities. These requirements are then compared with design-basis flood conditions to determine whether flood effects need to be considered in the plant's design or in emergency procedures.

Section 2.4.10 of this report presents an evaluation of the flooding protection for the proposed plant site.

2.4.10.2 *Summary of Application*

This section of the WLS COL FSAR addresses the needs for site-specific information on flooding protection requirements. The applicant addressed the information item identified in AP1000 DCD Tier 2, Section 2.4.1.2, Revision 19 related to flooding protection requirements as follows:

AP1000 COL Information Item

- WLS COL 2.4-2

In addition, this section addresses the following COL-specific information identified in AP1000 DCD Tier 2, Section 2.4.1.2, Revision 19.

> Combined License applicants referencing the AP1000 certified design will address the following site-specific information about historical flooding and potential flooding factors, including the effects of local intense precipitation.
>
> - Probable Maximum Flood (PMF) on Streams and Rivers – Site-specific information that will be used to determine the design basis flooding at the site. This information will include the PMF on streams and rivers.
>
> - Dam Failures – Site-specific information about potential dam failures.
>
> - Probable Maximum Surge and Seiche Flooding – Site-specific information about probable maximum surge and seiche flooding.

- Probable Maximum Tsunami Loading – Site-specific information about probable maximum tsunami loading.

- Flood Protection Requirements – Site-specific information about flood protection requirements or verification that flood protection is not required to meet the site parameter for flood level.

In WLS COL FSAR Section 2.4.10, the applicant addressed the flood protection requirements at the site. The causes of floods and their effects were discussed in other WLS COL FSAR sections. No further action is required for sites within the bounds of the site parameter for flood level.

2.4.10.3 *Regulatory Basis*

The relevant requirements of NRC regulations for the identification and evaluation of flooding protection requirements, and the associated acceptance criteria, are described in NUREG-0800. Section 2.4.10

- 10 CFR 52.79(a)(1)(iii), "Contents of applications; technical information in final safety analysis report," as it relates to identifying hydrologic site characteristics with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

- 10 CFR Part 100, "Reactor Site Criteria," as it relates to identifying and evaluating hydrological features of the site. The requirement to consider physical site characteristics in site evaluations is specified in 10 CFR 100.20(c).

- 10 CFR 100.23(d), "Geologic and seismic siting factors," sets forth the criteria to determine the siting factors for plant design bases with respect to water levels at the site.

The staff also used the following regulatory guides for the acceptance criteria identified in NUREG-0800, Section 2.4.10:

- RG 1.59, "Design Basis Floods for Nuclear Power Plants," supplemented by best current practices, as it relates to providing assurance that natural flooding phenomena that could potentially affect the site have been appropriately identified and characterized

- RG 1.102, "Flood Protection for Nuclear Power Plants," as it relates to providing assurance that SSCs important to safety have been designed to withstand the effects of natural flooding phenomena likely to occur at the site

- RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," as it relates to the contents of a COL application

2.4.10.4 *Technical Evaluation*

In this section of the report, the staff reviewed the applicant's analyses related to flooding protection requirements. The staff's independent analysis, where needed for the review, is also described.

Information Submitted by the Applicant

The applicant stated that all safety-related facilities at the WLS site are located above the maximum flood level based on the design-basis flood evaluate in earlier sections of the WLS COL FSAR. Therefore, the applicant concluded that flood protection measures and emergency procedures are not required.

NRC Staff's Technical Evaluation

The staff reviewed COL Information Item 2.4-2 related to the provision of site-specific information about the PMF at the plant site included under WLS COL FSAR Section 2.4. Additional aspects of this information item are addressed in Sections 2.4.2. 2.4.3, 2.4.4, 2.4.5, 2.4.6, and 2.4.7 of this report.

The staff determined that the site characteristic maximum flood water-surface elevation near the WLS site from several flooding mechanisms described in earlier sections of this report remains below the site grade and meets the AP1000 DCD site parameter. Therefore, the staff concluded that flooding protection of safety-related SSCs at the WLS site is not needed.

Based on a review of the applicant's information in WLS COL FSAR Revisions 0 through 11, the staff concluded that the applicant has appropriately analyzed the need for flooding protection at the WLS site. Therefore, based on the reasons given above, the staff concluded that the applicant adequately identified and evaluated extreme flood events at the site. Therefore, based on the reasons given above, the staff concluded that the applicant adequately determined hydrologic characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

2.4.10.5 *Post Combined License Activities*

There are no post COL activities related to this section.

2.4.10.6 *Conclusion*

The staff reviewed the application and confirmed that the applicant has addressed the information to demonstrate that the characteristics of the site fall within the site parameters specified in the AP1000 DC rule, and that no outstanding information is required to be addressed in the WLS COL FSAR related to this section.

As set forth above, the applicant has presented and substantiated information relative to the flood protection measures important to the design and siting of this plant. The staff concluded that the applicant considered the appropriate site phenomena in establishing the flood protection measures for SSCs. The staff reviewed the information provided and, for the reasons given above, concluded that the applicant provided sufficient details about the site description to allow the staff to evaluate, as documented in Section 2.4.10 of this report, whether the applicant met the relevant requirements of 10 CFR 52.79(a)(1) and 10 CFR Part 100 with respect to determining the acceptability of the site. This addressed flooding protection requirements in COL Information Item 2.4-2. The staff concludes that the applicant provided sufficient information to satisfy the applicable requirements of 10 CFR Part 52 and 10 CFR Part 100.

2.4.11 Low Water Considerations

2.4.11.1 *Introduction*

WLS COL FSAR Section 2.4.11 describes natural events that may reduce or limit the available safety-related cooling-water supply. The applicant ensures that an adequate water supply will exist to shut down the plant under conditions requiring safety-related cooling.

Section 2.4.11 of this report presents an evaluation of the effects of low water-surface elevations caused by various hydrometeorological events.

As stated in Section 2.4 above, hydrologic characteristics associated with conditions that would result in a loss of external water supply and seismic design considerations of water-supply structures are not relevant for the AP1000 design. Therefore, low-water conditions were not part of the staff's review.

2.4.11.2 *Summary of Application*

This section of the WLS COL FSAR addresses the impacts of low water on water supply. The applicant addressed the information item identified in AP1000 DCD Tier 2, Section 2.4.1.2, Revision 19 related to low-water considerations as follows:

<u>*AP1000 COL Information Item*</u>

- WLS COL 2.4-3

In addition, this section addresses the following COL-specific information identified in AP1000 DCD Section 2.4.1.3, Revision 19.

> Combined License applicants will address the water supply sources to provide makeup water to the service water system cooling tower.

2.4.11.3 *Regulatory Basis*

The relevant requirements of NRC regulations for the low-water considerations, and the associated acceptance criteria, are described in NUREG-0800, Section 2.4.11.

The applicable regulatory requirements for identifying the effects of low water are as follows:

- 10 CFR 52.79(a)(1)(iii), “Contents of applications; technical information in final safety analysis report,” as it relates to identifying hydrologic site characteristics with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

- 10 CFR Part 100, “Reactor Site Criteria,” as it relates to identifying and evaluating hydrological features of the site. The requirement to consider physical site characteristics in site evaluations is specified in 10 CFR 100.20(c).

- 10 CFR 100.23(d), “Geologic and seismic siting factors,” sets forth the criteria to determine the siting factors for plant design bases with respect to seismically induced floods and water waves at the site.

2.4.11.4 *Technical Evaluation*

The following material in this section describes the staff’s review of information provided and analyses carried out by the applicant in its WLS COL FSAR. The staff’s independent analysis, where needed for the review, is also described.

The staff reviewed COL Information Item 2.4-3 related to the provision of site-specific information about low-water considerations at the plant site included under WLS COL FSAR Section 2.4. Additional aspects of this information item are addressed in Sections 2.4.8, 2.4.9, and 2.4.12 of this report.

To ensure that the site characteristics related to low-water events are based on the most conservative of plausible conceptual models, in RAI 825, Question 02.04.11-1, the staff requested that the applicant describe the process followed to determine the conceptual models for low water from drought and from other phenomena and the effects of low water on safety-related water supplies under possible water-use limits. In an October 27, 2008, response to RAI 825, Question 02.04.11-1, the applicant stated that it characterized low-flow conditions in the Broad River using the 10-year return period 7-day flow (7Q10) at the Gaffney, SC, USGS streamflow gauge. The applicant noted that it supplemented the available streamflow record at the Gaffney gauge by additional periods of streamflow data from two upstream gauges at Blacksburg, SC and Boiling Springs, NC. The applicant estimated the 7Q10 flow from a Log-Pearson Type III distribution fitted to the annual 7-day low flow for each year in the record. The applicant stated that the estimated 7Q10 flow for the Broad River at the Gaffney gauge is 13.6 m^3/s (479 cfs), which is approximately the same as the Federal Energy Regulatory Commission (FERC) minimum flow requirement of 13.7 m^3/s (483 cfs) for the Ninety-Nine Islands Hydroelectric Station.

The applicant defined a minimum discharge of 15.2 m^3/s (538 cfs), the sum of the minimum FERC requirement of 13.7 m^3/s (483 cfs) and the expected consumptive water use of 1.6 m^3/s (55 cfs) for the proposed units, which are needed to support current water use and quality downstream of the site. When the Broad River discharge falls below 15.2 m^3/s (538 cfs), onsite water storage would supplement makeup water from the Broad River for the proposed units. When the Broad River discharge falls below 13.7 m^3/s (483 cfs), only Make-Up Ponds B and C would supply makeup water to the proposed units. The staff reviewed the applicant's October 27, 2008, response to RAI 825, Question 02.04.11-1, and subsequent updates to the WLS COL FSAR up to and including Revision 11 to conclude that the applicant's process to determine the conceptual models for low flow are adequately described. The staff also determined that there are no safety-related systems that can be affected by low water at the WLS site. Accordingly, the staff considers RAI 825, Question 02.04.11-1, resolved.

Low Flow in Rivers and Streams

Information Submitted by the Applicant

The applicant stated that the passive cooling system of the AP1000 design does not rely on the Broad River as a source of water and therefore, no safety-related facilities of the WLS would be affected by low-water conditions in the river.

NRC Staff's Technical Evaluation

There are no safety-related systems that can be affected by low water at the WLS site.

Low Water Resulting from Surges, Seiches, or Tsunami

Information Submitted by the Applicant

The applicant stated that no safety-related systems can be affected by low water at the WLS site.

NRC Staff's Technical Evaluation

The staff agreed with the applicant that no safety-related systems can be affected by low water at the WLS site.

Historical Low Water

Information Submitted by the Applicant

The applicant stated that no safety-related systems can be affected by low water at the WLS site.

NRC Staff's Technical Evaluation

The staff agreed with the applicant that no safety-related systems can be affected by low water at the WLS site.

Future Controls

Information Submitted by the Applicant

The applicant stated that no safety-related systems can be affected by low water at the WLS site.

NRC Staff's Technical Evaluation

The staff agreed with the applicant that no safety-related systems can be affected by low water at the WLS site.

Plant Requirements

Information Submitted by the Applicant

The applicant stated that no safety-related systems can be affected by low water at the WLS site.

NRC Staff's Technical Evaluation

The staff agreed with the applicant that no safety-related systems can be affected by low water at the WLS site.

Heat Sink Dependability Requirements

Information Submitted by the Applicant

The applicant reported that the atmosphere provides the UHS for the AP1000 design and the passive containment cooling system does not rely on water from the Broad River. The applicant also stated that no water from the Broad River or from other outside sources is required for safe emergency shutdown because the passive containment cooling-water storage tank stores water required for 72 hours of containment wetting and the passive containment cooling ancillary water storage tank has the capacity to provide containment wetting for an additional 4 days.

NRC Staff's Technical Evaluation

In RAI 825, Question 02.04.11-2, the staff requested that the applicant describe the term "normal plant shutdown" and to clarify whether any safety-related water would be needed during normal plant shutdown.

In an October 27, 2008, response to RAI 825, Question 02.04.11-2, the applicant stated that a normal plant shutdown is a non-emergency procedure and does not require any safety-related water. The normal shutdown would require approximately 130,749 m^3 (106 ac-ft) of water for the two proposed units and an additional 176,388 m^3 (143 ac-ft) to maintain shutdown conditions for 90 days after a normal shutdown. The applicant stated that Make-Up Pond A, with its usable storage capacity of approximately 1,480,178 m^3 (1,200 ac-ft), would have sufficient water to support a normal plant shutdown and to maintain shutdown conditions for durations that are significantly longer than those of any recorded period of low flow. The applicant also stated that it has no plans to draw down Make-Up Pond A to support plant water needs during power production. The staff notes that the applicant has updated the WLS COL FSAR text.

The staff reviewed the functioning of the AP1000 UHS. The passive cooling system of the proposed units is assisted by a water spray on the containment vessel provided by a passive containment cooling-water storage tank located on top of the containment building, which holds a 3-day supply of water following a design-basis accident. Additional water is stored in a passive containment cooling ancillary water storage tank for an additional 4 days. Technical specifications would ensure that this 7-day supply of water to assist in cooling is always available. The proposed reactors at the WLS site would not depend on any external source of water supply for safe shutdown. The staff evaluated the applicant's October 27, 2008, response to RAI 825, Question 02.04.11-2, and concluded that the process used by the applicant to determine heat sink dependability requirements is adequate. The staff concluded that there are no safety-related systems that can be affected by low water at the WLS site. Accordingly, the staff considers RAI 825, Question 02.04.11-2, resolved.

2.4.11.5 *Post Combined License Activities*

There are no post COL activities related to this section.

2.4.11.6 *Conclusion*

The staff reviewed the application and confirmed that the applicant has addressed the information required and that no site characteristics related to low-water conditions apply to the AP1000 design.

As set forth above, the applicant has presented and substantiated information relative to the low-water effects important to the design and siting of this plant. The staff found that the applicant has considered the appropriate site phenomena in establishing the design bases for SSCs. The staff reviewed the information provided and, for the reasons given above, concluded that the applicant has provided sufficient details about the site description to allow the staff to evaluate, as documented in Section 2.4.11 of this report, whether the applicant has met the relevant requirements of 10 CFR 52.79(a)(1) and 10 CFR Part 100 with respect to determining the acceptability of the site. This addressed low-water considerations in COL Information Item 2.4-3. The staff concludes that the applicant has provided sufficient information to satisfy the applicable requirements of 10 CFR Part 52 and 10 CFR Part 100.

2.4.12 Groundwater

2.4.12.1 *Introduction*

This section of the WLS COL FSAR describes the hydrogeological characteristics of the site. One of the key objectives of groundwater investigations and monitoring at this site is to evaluate the maximum groundwater-surface elevation at the site, which is used in Section 2.5 of this report to determine the effects of groundwater on the stability of the plant foundations and slopes. The evaluation is performed to ensure that the maximum groundwater-surface elevation remains less than the 29.9 m (98.0 ft) plant elevation. Other significant objectives are to examine whether groundwater provides any safety-related water supply, to determine whether dewatering systems are required to maintain groundwater-surface elevations below the required elevation, and to describe subsurface pathways for potential groundwater contaminants.

The specific areas of review are as follows: (1) identification of the aquifers, types of onsite groundwater use, sources of recharge, present withdrawals and known and likely future withdrawals, flow rates, travel time, gradients, and other properties that affect movement of accidental contaminants in groundwater, groundwater-surface elevations beneath the site, seasonal and climatic fluctuations, monitoring and protection requirements, and man-made changes that have the potential to cause long-term changes in local groundwater regime; (2) effects of groundwater-surface elevations and other hydrodynamic effects of groundwater on the design bases of plant foundations and those of other SSCs important to safety; (3) reliability of groundwater resources and related systems used to supply safety-related water to the plant; (4) reliability of dewatering systems to maintain groundwater conditions within the plant's design bases; (5) potential effects of seismic and non-seismic information about the postulated worst-case groundwater conditions for the proposed plant site; and (6) any additional information requirements prescribed within the "Contents of Application" sections of the applicable subparts of 10 CFR Part 52.

2.4.12.2 *Summary of Application*

This section of the WLS COL FSAR addresses groundwater conditions in terms of impacts on structures and water supply. The applicant addressed information related to groundwater as follows:

AP1000 COL Information Item

- WLS COL 2.4-4

In addition, this section addresses the following COL-specific information identified in AP1000 DCD Section 2.4.1.4, Revision 19.

> Combined License applicants referencing the AP1000 certified design will address site-specific information on groundwater. No further action is required for the sites within the bounds of the site parameter for groundwater.

License Condition

- Part 10, License Condition 15, "Removal of Legacy Stormwater Drain Line"

The applicant proposed a license condition requiring that a single legacy Cherokee project stormwater drain line (designed to transfer stormwater from the Cherokee power block area to Hold-Up Pond A) and any associated bedding material representing a potential preferential groundwater pathway be removed and the excavation backfilled with compacted native soils.

2.4.12.3 *Regulatory Basis*

The relevant requirements of NRC regulations for groundwater, and the associated acceptance criteria, are specified in NUREG-0800, Section 2.4.12.

The applicable regulatory requirements for groundwater are set forth in the following:

- 10 CFR 52.79(a)(1)(iii), as it relates to identifying hydrologic site characteristics with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

- 10 CFR Part 100, as it relates to identifying and evaluating hydrological features of the site. The requirement to consider physical site characteristics in site evaluations is specified in 10 CFR 100.20(c).

The staff also used the acceptance criteria identified in NUREG-0800, Section 2.4.12:

- Local and Regional Groundwater Characteristics and Use: The applicant should supply a complete description of regional and local groundwater characteristics and groundwater use, groundwater monitoring and protection requirements, and any man-made changes with a potential to affect regional groundwater characteristics over a long period of time.

- Effects on Plant Foundations and other Safety-Related Structures, Systems, and Components: The applicant should supply a complete description of the effects of groundwater-surface elevations and other hydrodynamic effects on the design bases of plant foundations and other SSCs important to safety.

- Reliability of Groundwater Resources and Systems Used for Safety-Related Purposes: The applicant should supply a complete description of all SSCs important to safety that depends on groundwater, as well as data and analysis regarding the reliability of the groundwater source.

- Reliability of Dewatering Systems: The applicant should supply a complete description of the site dewatering system, including its reliability to maintain groundwater conditions within the groundwater design bases of SSCs important to safety.

- Consideration of Other Site Related Evaluation Criteria: The applicant should supply an assessment of the potential effects of seismic and non-seismic information about the postulated worst-case scenario related to groundwater effects for the proposed plant site.

The regulatory basis of the information incorporated by reference will be addressed in the staff's final safety evaluation report (FSER) related to the AP1000 certified design.

2.4.12.4 *Technical Evaluation*

The staff reviewed WLS COL FSAR Section 2.4.12 Revisions 0 to 7; corrections and additions to the WLS COL FSAR submitted by the applicant as letters; and associated applicant responses to RAIs issued by the staff. The conclusions of the review are current with, and apply to, WLS COL FSAR Revision 11. However, frequent references are made to earlier WLS COL FSAR revisions where necessary to explain the reasons for RAIs. The staff also checked the referenced AP1000 DCD and departures and supplements specified in the WLS COL FSAR.

The staff reviewed COL Information Item 2.4-4 related to the provision of site-specific information about groundwater at the plant site included under WLS COL FSAR Section 2.4.

For clarity, the technical evaluation is organized into six subsections, each addressing a specific issue: conceptual model; offsite wells; aquifer properties; alternative pathways; maximum water table level; and monitoring. Each subsection describes (1) the staff's review of information and analyses that the applicant provided in the WLS COL FSAR, (2) RAIs issued by the staff and the applicant's responses, and (3) the staff's independent analysis, where needed for the review.

Conceptual Model

Information Submitted by the Applicant

The applicant described its conceptual model of the hydrogeology of the site in WLS COL FSAR Sections 2.4.12.1, 2.4.12.2, and 2.4.12.3. The site is located in the Piedmont physiographic province and is underlain by metamorphic rocks of volcanic, intrusive, and sedimentary origin. Groundwater may be obtained from fractures within the bedrock, but near the surface it occurs under unconfined conditions in artificial fill materials, soil and saprolite that overlie bedrock, and (partially weathered rock (PWR)). The PWR tends to have the highest hydraulic conductivity. Groundwater originates from precipitation, which infiltrates in upland areas, then flows mostly within the near-surface materials toward lower areas where it discharges to the Broad River, the makeup ponds, and other small bodies of surface water. Groundwater supplies mostly domestic wells in the area near the site. The applicant does not currently use or plan to use onsite groundwater.

In response to four staff RAIs, the applicant provided additional details and clarifications related to the rationale for the conceptual model, the impact of variable precipitation on groundwater flow and direction, the role of seeps and springs, and the definition of the word "preferential."

NRC Staff's Technical Evaluation

The staff reviewed the applicant's description of the site conceptual model. To better understand the applicant's rationale and support for the conceptual model, the staff issued RAI 826, Question 02.04.12-1. In a December 11, 2008, response, the applicant provided an overview of the methodology it used to develop the conceptual model of the site. The applicant utilized regional data, Cherokee-era site data, and data collected as part of the WLS characterization effort to characterize the key materials (i.e., fill; saprolite; residual soil; PWR). When site-specific data did not exist, the applicant used literature values. Using the conceptual model and groundwater well data, the applicant identified five potential flow paths between a postulated leak at the nuclear island and the site boundary. In the initial WLS COL FSAR, the applicant had identified only one path; in the current WLS COL FSAR, because of site grading changes, the applicant reduced the number of plausible pathways from five to four. The applicant provided additional details related to the flow paths in response to other WLS COL FSAR RAIs. The applicant proposed to update the WLS COL FSAR with this new information about conceptual model identification. The staff notes that applicant included this information in WLS COL FSAR Section 2.4.12.2, “Sources.” The staff concluded that the applicant’s December 11, 2008, response to RAI 826, Question 02.04.12-1, was acceptable and considers the question resolved.

To help understand how groundwater elevation and flow direction responded spatially and temporally to precipitation, the staff issued RAI 826, Question 02.04.12-3. In a December 11, 2008, response, the applicant supplemented the onsite precipitation data for December 2005 to November 2006 with data from 1950 to 2008 from the Greenville-Spartanburg Airport, located 72 km (45 mi) to the west. The applicant noted that precipitation was relatively evenly distributed throughout the year, yet groundwater levels increased during the winter, reaching a maximum in April and May, and decreased in summer, reaching a minimum in October and November. The applicant attributed the seasonal effect to evapotranspiration, which is lowest in winter and highest in summer. The staff notes that the applicant included this information in WLS COL FSAR Section 2.4.12.2, “Sources.” The staff concluded that the applicant’s December 11, 2008, response to RAI 826, Question 02.04.12-3, was acceptable and considers the question resolved.

Using precipitation data from the Greenville-Spartanburg Airport, the staff concluded that precipitation in 2005 was 8 percent above average (for the period 1950-2007), 15 percent below average in 2006, and 37 percent below average in 2007. The applicant monitored groundwater levels from April 2006 to April 2007. The staff concluded that those groundwater levels are reflective of drier than average conditions. This issue is discussed further under the topic “Maximum Water Table Level.”

To help understand the role of springs and seeps in groundwater discharge, the staff issued RAI 826, Question 02.04.12-9. In a December 11, 2008, response, the applicant provided a figure showing the locations of springs and seeps in relation to the WLS nuclear island. The applicant stated that, in 1973, springs and seeps were observed in various locations across the site, but predominantly in drainage channels. Cut-and-fill activities associated with the Cherokee Nuclear Plant buried most of those locations. In 2006, springs and seeps were much less prevalent and none was near the proposed site for the nuclear island. The applicant also stated that the number of springs and seeps observed in 2006 may have been affected by the excavation dewatering, which began in December 2005. The figure provided by the applicant helped the staff to recognize the association of seeps with drainages and the proximity of those drainages to the WLS nuclear island. This information was sufficient for the staff to perform and complete its review. The staff concluded that the applicant’s December 11, 2008, response to RAI 826, Question 02.04.12-9, was acceptable and considers the question resolved.

To clarify the applicant’s usage of the technical term “preferential” with regard to groundwater flow, the staff issued RAI 70, Question 02.04.12-17. In a July 31, 2009, response, the applicant agreed to use the term “limiting” in place of “preferential” when referring to the groundwater flow path that represents the shortest travel time, and to revise the WLS COL FSAR accordingly. The applicant also acknowledged the potential for buried Cherokee pipes to act as preferential flow paths, in the sense that they may provide high-permeability paths for groundwater flow. The applicant evaluated the expected post-construction groundwater surface and determined that the pipe that runs from the nuclear power block area north to Hold-Up Pond A could be below the future groundwater surface and therefore could act as a preferential pathway. The applicant proposed to update the WLS COL FSAR to explain the issue and stated that the buried pipe and bedding material would be removed. The staff notes that the applicant included this information in WLS COL FSAR Section 2.4.12.2, “Sources.” The staff concluded that the applicant’s July 31, 2009, response to RAI 70, Question 02.04.12-17, was acceptable and considers the question resolved.

Offsite Wells

Information Submitted by the Applicant

The applicant provided information about offsite wells in WLS COL FSAR 2.4.12.2. This information was provided for the staff to identify potential groundwater pathways and determine whether construction and operation of the WLS nuclear plants could affect offsite wells. In response to two staff RAIs, the applicant provided additional information about the locations of groundwater users near the site.

NRC Staff’s Technical Evaluation

The staff reviewed the applicant’s information about offsite wells. The staff concluded that additional information was required to complete its evaluation of risks to offsite groundwater users.

To help understand the locations of groundwater users near the site who might be at risk, the staff issued RAI 826, Question 02.04.12-2. In a December 11, 2008, response, the applicant provided information about well depth, well abandonment, and conversion to municipal water. The applicant reported that only 3 of the 50 wells identified in the Cherokee Nuclear Station Environmental Report were drilled deeper than 45.7 m (150 ft). Since 1985, the State of South Carolina reported 22 wells drilled deeper than 45.7 m (150 ft) within 1.6 km (1 mi) of the WLS site property boundary. The applicant provided information from the Draytonville Water District showing that 55 percent of residents within 3.2 km (2 mi) of the reactor buildings had connected to the public water supply and that, based on planned expansion, service would be available to 83 percent of residents in 2009. The applicant stated that it did not detect a trend to abandon existing wells.

The staff noted that all of the public wells are more than 0.8 km (0.5 mi) from the proposed nuclear island. In addition, all of the 22 wells drilled to deeper than 45.7 m (150 ft) since 1985 are more than 1.6 km (1 mi) from the proposed nuclear island and all are separated from the WLS nuclear island by a water body (i.e., the Broad River; Make-Up Pond A; Make-Up Pond B). Therefore, the staff concluded that it is unlikely that groundwater uptake would be directly involved in a groundwater radionuclide pathway. The staff concluded that the applicant’s December 11, 2008, response to RAI 826, Question 02.04.12-2, was acceptable and considers the question resolved.

To help understand potential risks to groundwater users near the site, the staff issued RAI 826, Question 02.04.12-7. In a December 11, 2008, response, the applicant stated that WLS COL FSAR Figure 2.4.12-202 was developed using data from the period from 1976 to 1985 during which the Cherokee Nuclear Station was dewatering the excavation. The staff notes that annual precipitation during this period ranged from 72 to 134 percent of normal, suggesting that the groundwater observations span a reasonable range of precipitation conditions. The applicant stated that the [name withheld] well is located 1,524 m (5,000 ft) south of the nuclear island. At that distance, the well is outside the zone of influence of the construction dewatering. The applicant stated that the Piedmont aquifer consists of porous material above continuous bedrock and because there are no confining layers, the [name withheld] well likely produces water from the same unconfined aquifer as that which exists at the WLS site. The applicant proposed adding the offsite well information to the WLS COL FSAR to support the assumptions that offsite wells are not a potential contaminant transport pathway and that the wells would not be affected by construction and operation of the WLS reactors. The staff notes that the applicant included this information in WLS COL FSAR Section 2.4.12.2, “Sources.” The staff concluded that the applicant’s December 11, 2008, response to RAI 826, Question 02.04.12-7, was acceptable and considers the question resolved.

Aquifer Properties

Information Submitted by the Applicant

The applicant provided information about the properties of aquifers at the site in WLS COL FSAR Section 2.4.12.2.4, and additional information in WLS COL FSAR 2.5.4. Aquifer materials described were the artificial fill, soil and saprolite, PWR, and fractured bedrock. Properties of particular interest to the staff were hydraulic conductivity, total porosity, and effective porosity. These properties are primary influences on the direction and velocity of groundwater movement in the subsurface, and are among the major influences on movement of radionuclides that move with groundwater. Retardation of radionuclide movement by interaction with aquifer materials is discussed below in Section 2.4.13. In response to four staff RAIs, the applicant provided additional information about the methods used to determine porosity and effective porosity, the decrease of hydraulic conductivity with increasing depth, and the selection of a conservative hydraulic conductivity value for calculations of groundwater velocity.

NRC Staff’s Technical Evaluation

The staff reviewed information provided by the applicant about the hydraulic properties of aquifer materials. The staff concluded that additional information about aquifer properties was required to support its evaluation of maximum groundwater levels and the direction and velocity of radionuclide movement with groundwater.

To clarify the methods used by the applicant to determine porosity, and to allow staff to evaluate the reasonableness of measured and estimated porosity values, the staff issued RAI 826, Question 02.04.12-4. In a December 11, 2008, response and subsequent May 12, 2009, response, the applicant stated that it had estimated the effective porosity for fill and a mix of residual soil and saprolite to be 0.09 and 0.20, respectively, using a USGS method that required particle size data. The applicant estimated the effective porosity for the PWR to be 0.08 by measuring the liquid drained from a single PWR sample. The staff reviewed and confirmed the calculations. The applicant compared the average hydraulic conductivity and effective porosity values of soil/saprolite and PWR at the WLS site to those at the Catawba Nuclear site, which is located in a similar piedmont region about 32 km (20 mi) to the east. Average hydraulic conductivities at the WLS site were 17 and 55 percent less for soil/saprolite and PWR, respectively, than at the Catawba site. The average effective porosity of the soil/saprolite at the WLS site was 77 percent of the value at Catawba. The average effective porosity of the PWR at WLS was 45 percent greater than that at Catawba. The applicant proposed to update WLS COL FSAR Tables 2.4.12-203 and 2.4.12-204. Subsequently, the applicant deleted WLS COL FSAR Table 2.4.12-203 because the information was contained in the text and in WLS COL FSAR Table 2.5.4-211. The staff notes that the applicant included this information in WLS COL FSAR Section 2.4.12.2, “Sources,” and WLS COL FSAR Section 2.5.4, “Stability of Subsurface Materials and Foundations.” The staff concluded that the applicant’s December 11, 2008, and May 12, 2009, responses to RAI 826, Question 02.04.12-4, were acceptable and considers these questions resolved.

The applicant estimated the effective porosity of the PWR by saturating the sample, then allowing it to drain; the volume of the liquid that drains provides a measure of the effective porosity. This method is typically unable to drain all liquid that would normally drain in a field setting, thus the method underestimates effective porosity. The staff notes that the effective porosity of the PWR material is based on the measurement results for a single sample. However, comparison with the effective porosity at other sites shows it to be consistent. The staff concluded that using the effective porosity value, determined using the method described above, would be conservative for use in calculating travel time.

To clarify the source and validity of certain parameter values presented by the applicant, the staff issued RAI 826, Question 02.04.12-5. In the December 11, 2008, response, the applicant identified grain size distribution, specific gravity, unit weight of soil and hydraulic conductivity as parameters that were measured using American Society for Testing and Materials procedures. The applicant identified total porosity and effective porosity as parameters that were calculated or estimated based on the values of other parameters. Total porosity was estimated using the dry unit weight and specific gravity of the soil and unit weight of water. Effective porosities of fill, soil, and saprolite were estimated using grain size distribution. The effective porosity of PWR was estimated using the saturated and drained unit weights. The staff reviewed the procedures and confirmed the calculations. The applicant proposed to update the WLS COL FSAR with this information about porosity estimation. The staff notes that the applicant included this information in WLS COL FSAR Section 2.4.12.2, “Sources,” and WLS COL FSAR Section 2.5.4, “Stability of Subsurface Materials and Foundations.” The staff concluded that the applicant’s December 11, 2008, response to RAI 826, Question 02.04.12-5, was acceptable and considers the question resolved.

To clarify the basis for an observed decrease in hydraulic conductivity with increasing depth, the staff issued RAI 826, Question 02.04.12-10. In a December 11, 2008, response, the applicant provided two figures, one showing the 1970s hydraulic conductivity data and the other the 2006 data. Both sets of data show that hydraulic conductivity of the PWR decreases with increasing depth. The applicant did not adjust or rectify the depths for different ground-surface elevation during these two periods. The applicant identified conservative values of hydraulic conductivity to be used in calculating travel time. For each material, the applicant defined the conservative hydraulic conductivity to be the geometric mean of all values above the median value for that material. For the PWR, the applicant used a slightly higher value (1.4×10^{-3} cm/s versus 1.0×10^{-3} cm/s (3.97 ft/day to 2.83 ft/day)) that was obtained in 2006 from an aquifer test conducted along the flow path expected to have the shortest travel time (i.e., the one going north from the proposed Unit 2 to the Broad River). The staff examined the figures provided by the applicant and confirmed that conductivity in the PWR decreases with depth. The staff considered the conductivity data and agreed that the value used for the PWR is conservative for travel-time calculations. The applicant proposed to update the discussion of hydraulic conductivity in the WLS COL FSAR. The staff notes that the applicant included this information in WLS COL FSAR Section 2.4.12.2, "Sources." The staff concluded that the applicant's December 11, 2008, response to RAI 826, Question 02.04.12-10, was acceptable and considers the question resolved.

To clarify what value of hydraulic conductivity should be used for conservative calculations of groundwater velocity, the staff issued RAI 70, Question 02.04.12-16. In a July 31, 2009, response, the applicant explained that tests conducted in the 1970s for the Cherokee site investigation provided results for unconsolidated material, which was a label that referred to the aquifer material without discriminating between soil, alluvium, saprolite, and PWR. The applicant described the current site conceptual model in terms of specific material (e.g., saprolite, PWR) rather than the bulk "unconsolidated material." The applicant further explained that the PWR was the most transmissive material and that the hydraulic conductivity value of 1.4×10^{-3} cm/s (3.97 ft/day) was nine times greater than the median value. The applicant proposed to update the discussion of hydraulic conductivity in the WLS COL FSAR and clarify the data support and property estimates. The staff notes that the applicant included this information in WLS COL FSAR Section 2.4.12.2, "Sources." The staff concluded that the applicant's July 31, 2009, response to RAI 70, Question 02.04.12-16, was acceptable and considers the question resolved.

Alternative Flow Paths

Information Submitted by the Applicant

The applicant identified a single bounding groundwater pathway that could carry radionuclides from a postulated release to locations where groundwater could discharge to surface water by which members of the public could be exposed. Groundwater originates from precipitation that infiltrates into soil in upland areas. It then moves generally down slopes toward the Broad River and other surface-water bodies. The applicant did not describe how the bounding pathway was identified and did not evaluate alternative pathways.

In response to four staff RAIs, the applicant provided additional details regarding alternative groundwater pathways, the geologic materials along those pathways, and the impact of temperature and dissolved solids on flow along groundwater pathways, and hydraulic gradients along those pathways.

NRC Staff's Technical Evaluation

To ensure that all possible groundwater flow paths are being considered, the staff issued RAI 826, Question 02.04.12-8. In a December 11, 2008, response, the applicant stated that WLS COL FSAR Figure 2.4.12-204, Sheet 8, which shows post-construction water table conditions, was produced from knowledge of 1973 groundwater conditions and the current water table. Using the same figure, the applicant identified five alternative conceptual flow paths from the nuclear island to the accessible environment. Subsequently, after revising the site layout and drainage plan, the applicant reduced the number of pathways to the four shown in WLS COL FSAR Table 2.4.12-1. The applicant identified Pathway 1 as the limiting flow path, meaning the most conservative pathway with respect to predicting the fastest contaminant movement. The staff notes that the applicant included this information in WLS COL FSAR Section 2.4.12.3, "Groundwater Movement." The staff concluded that the applicant's December 11, 2008, response to RAI 826, Question 02.04.12-8, was acceptable and considers the question resolved.

The staff determined that there was uncertainty in the geologic materials that are present along each plausible groundwater pathway. The staff also determined that the major materials, the soil, saprolite, and PWR, were all exposed in the existing excavation and that a postulated leak could enter any of these materials. To address these issues, the staff issued RAI 70, Question 02.04.12-15. In a December 18, 2009, response, the applicant assumed the presence of PWR (the most highly conductive of the three materials) for all pathways. Table 2.4.12-1 of this report shows the corresponding travel-time estimates are shorter and that Pathway 1, from Unit 2 to Hold-Up Pond A, has the shortest travel time and would therefore represent the most conservative pathway to use for the transport analysis in WLS COL FSAR Section 2.4.13. The applicant stated that a storm drain system (DRS) would be designed to route runoff from the nuclear power block area to reduce the potential for flooding. The applicant does not expect the DRS to cause any other groundwater pathway to have a shorter travel time than Pathway 1. The staff notes that the applicant included this information in WLS COL FSAR Section 2.4.12.2, "Sources."

Table 2.4.12-1 Pathway Descriptions and Travel-Time Estimates (for WLS COL FSAR Revision 1, actual geology was used; for WLS COL FSAR Revision 7, the geology for all pathways was assumed to be the PWR material)

Pathway Number	Groundwater Pathway Description	Estimated Travel Time (yr)	
		WLS COL FSAR Revision 1	WLS COL FSAR Revision 7
1	Unit 2 to Hold-Up Pond A	7.2	1.6
2	Unit 2 to the Broad River	2.8	2.7
3	Unit 2 to Make-Up Pond A	23	4.0
4	Unit 1 to Make-Up Pond B	9.8	5.5

The staff evaluated the pathways identified by the applicant and determined that they adequately represent the plausible pathways to each of the major water bodies. The travel paths are conservatively evaluated as straight lines rather than the curved flow paths indicated by the groundwater contour map. Post-construction water table elevations might cause slight differences in groundwater flow and direction, but any change from a straight line would only elongate the travel path and lengthen the travel time. Groundwater gradients would also be affected by post-construction water table elevations. For the travel-time estimates in Table 2.4.12-1 of this report, the applicant used its estimate of the maximum water table elevation of 178 m (584 ft) above MSL to calculate the gradient. The staff considers this choice conservative with respect to calculation of the groundwater gradient. The staff notes that the applicant included this information in WLS COL FSAR Section 2.4.12.3, "Groundwater Movement." The staff concluded that the applicant's December 18, 2009, response to RAI 70, Question 02.04.12-15, was acceptable and considers the question resolved.

Liquid released from the liquid waste management system could have a higher temperature and different content of dissolved solids than ambient groundwater. To address the possibility that this might affect groundwater flow paths, the staff issued RAI 826, Question 02.04.12-12. In a December 11, 2008, response, the applicant responded that the leaked fluid would have a total dissolved solids (TDS) content of less than 1 ppm. The applicant stated that any physical properties of the leaked fluid that differed from groundwater conditions would quickly dissipate relative to the travel time. That is, the leaked fluid would quickly take on the physical characteristics of groundwater and, therefore, would neither preferentially rise nor sink as it moved away from the nuclear island.

The staff examined AP1000 DCD Table 5.2-2, which lists the reactor coolant water chemistry specifications. AP1000 DCD Table 5.2-2 suggests the TDS could be as high as 2 parts per million (ppm) if all constituents were at their maximum values. Even if the TDS was 2 ppm, the concentration would still be much lower than the average groundwater TDS of 107 ppm. At the average groundwater temperature of 17ºC (63ºF), the density difference between liquids with 2 and 107 ppm would be less than 0.01 percent. However, the temperature in the effluent holding tank could be much higher than the ambient groundwater temperature. AP1000 DCD Table 11.2-2 lists the design temperature of the effluent hold-up tanks as 65.6ºC (150ºF). The density of a liquid at that temperature and with a TDS of 2 ppm would be almost 2 percent less than the density of groundwater at the WLS site. Such a density difference could lead to buoyancy and affect hydraulic conductivity until the temperature difference dissipated. During that time, the buoyant leaked fluid could rise into shallower aquifer material and potentially travel via an alternate pathway. The staff considered the finite volume of leaked fluid and believes the temperature difference would dissipate quickly such that the fluid properties would resemble those of the ambient groundwater. Furthermore, if the initial properties of the leaked liquid caused it to flow upward or downward into an alternate flow path, the aquifer materials above and below are less transmissive than the PWR, which is the only material considered in the flow paths analyzed by the applicant. Since the leaked fluid properties exist for a limited time and only in the vicinity of the reactor buildings, and because alternate pathways would have less transmissive properties, the staff concludes that including a separate alternative pathway is not warranted. The staff concluded that the applicant's December 11, 2008, response to RAI 826, Question 02.04.12-12, was acceptable and considers the question resolved.

To clarify what groundwater elevations and travel distances should be used in calculating groundwater gradients, the staff issued RAI 826, Question 02.04.12-13. In a December 11, 2008, response, the applicant revised its estimate of the maximum groundwater elevation upward from 176.5 to 178.0 m (579 to 584 ft) above MSL based on the maximum height of water in the excavation, 176.5 m (579 ft) above MSL, and its expectation that seasonal groundwater variation would be approximately 1.5 m (approximately 5 ft). In subsequent updates to the WLS COL FSAR, the applicant estimated the distance from Unit 2 to the nearest edge of Hold-Up Pond A to be 405 m (1,330 ft). The applicant estimated the groundwater gradient to be 0.0368 between Unit 2 and Hold-Up Pond A and 0.036 between Unit 2 and the Broad River. The staff notes that the applicant updated WLS COL FSAR Figure 2.4.12-208 with the information necessary to calculate groundwater gradients. The staff concluded that the applicant's December 11, 2008, response to RAI 826, Question 02.04.12-13, was acceptable and considers the question resolved.

Maximum Groundwater Elevation

Information Submitted by the Applicant

The maximum allowable groundwater elevation is specified as a site parameter in the AP1000 DCD and is defined in the WLS COL FSAR to be less than plant elevation 29.87 m (98 ft). The initial WLS COL FSAR established the plant elevation to be equivalent to the local elevation of 179.83 m (590 ft) above MSL. Therefore, the maximum allowable groundwater elevation was limited by the AP1000 DCD requirement to be less than 179.22 m (588 ft) above MSL. Throughout most of the staff review period, the maximum allowable groundwater elevation remained unchanged, thus, most of the following discussion refers to maximum groundwater elevation of 179.22 m (588 ft) above MSL.

Due to of changes to the site layout and grading, the current WLS COL FSAR establishes plant elevation to be 180.7 m (593 ft) above MSL. Therefore, the maximum allowable groundwater elevation is limited by the AP1000 DCD requirement to be less than 180.14 m (591 ft) above MSL.

In the earlier versions of the WLS COL FSAR, the applicant stated that the maximum actual groundwater elevation was expected to be 178.00 m (584 ft) above MSL. The applicant based this maximum actual groundwater elevation on the pre-development groundwater elevations, anticipated land-surface elevations, elevations of surface-water bodies such as the make-up ponds, water elevation in the excavation left after Cherokee plant construction, and seasonal groundwater levels observed in monitoring wells.

In response to five staff RAIs, the applicant provided additional information about establishing the maximum groundwater elevation, characterizing the post-construction groundwater elevations, site grading and drainage. Based on all the information made available, the applicant conducted groundwater modeling to estimate the maximum groundwater elevation. The highest modeled groundwater elevations near the plant were 179.0 and 179.1 m (587.2 and 587.5 AP1000 DCD ft) above MSL.

NRC Staff's Technical Evaluation

The staff reviewed the applicant's information and analysis related to groundwater levels. The staff paid particular attention to the approach that the applicant used to estimate maximum post-construction groundwater levels. The staff concluded that this approach, while valid in a broad sense, relied too much on generalizations and individual judgment to predict maximum groundwater levels accurately enough to ensure that the AP1000 DCD requirement would be met. To obtain more exact information, the staff issued a series of RAIs to examine the bases of the applicant's groundwater level estimates and to examine the applicability of alternative calculation techniques. Discussion of these RAIs is organized into the four following unnumbered sections corresponding to the principal topics addressed by the RAIs.

Initial Non-Modeling Estimates of Groundwater Elevations

To examine the value of using water marks left by water in the excavation as indicators of groundwater level, and related topics, the staff issued RAI 826, Question 02.04.12-6. In a December 11, 2008, response, the applicant stated it did not have a precise record of the dewatering activities previously carried out during the construction of the Cherokee Nuclear units, and did not know whether there were any dewatering activities during the period after Cherokee construction ceased and before December 2005, when the applicant initiated dewatering of the Cherokee excavation pit. The applicant provided aerial photographs for two dates, February 1994 and February 2005, which show water levels in the excavation pit relative to the Cherokee structures. The applicant used a topographic survey conducted in 2006 to estimate that water-surface elevation in the excavation pit varied between 175.0 and 176.5 m (574 and 579 ft) above MSL from 1994 through 2005. The applicant noted that the average groundwater level fluctuation during the period from April 2006 to April 2007 was 1.4 m (4.5 ft). The applicant estimated, given that the high-water mark in the excavation was determined to be 176.48 m (579 ft) above MSL, the design groundwater elevation to be 176.5±1.5 m (579±5 ft) above MSL, which allows for a 1.5 m (5 ft) seasonal variation over the high-water mark. The current WLS COL FSAR contains this information.

The applicant presented evidence of water levels in the excavation to justify the selection of 178.0 m (584 ft) above MSL as the maximum water table elevation. The staff concluded that the water levels were controlled by the net lateral flux of groundwater and by precipitation and open-water evaporation. Groundwater flow away from the excavation has been facilitated, in part, by the presence of preferential flow paths created by the stormwater drains emplaced during Cherokee-era construction. The applicant acknowledged such preferential paths exist and committed to removing those drains.

- WLS Proposed License Condition 15: Prior to fuel load, the licensee shall confirm that a single legacy Cherokee project stormwater drain line (designed to transfer stormwater from the Cherokee power block area to Hold-Up Pond A) and any associated bedding material representing a potential preferential groundwater pathway have been removed and the excavation has been backfilled with compacted native soils.

Due to of uncertainty in the water balance of the excavation, the staff does not consider information regarding water levels sufficient to estimate the maximum water table level.

To understand how the applicant had estimated the post-construction configuration of the groundwater surface, and the factors that would control the post-construction groundwater surface at the site, the staff issued RAI 826, Question 02.04.12-14. In a December 11, 2008, response, the applicant reiterated that its estimated post-construction groundwater levels are based on the current water table and pre-construction water table. Water table elevations are expected to conform generally to the surface topography, as modified by WLS construction. The applicant did not recommend modeling of groundwater at the time of the response. The applicant expected to implement a groundwater monitoring program after construction.

To obtain additional information necessary for understanding the data and methods that the applicant used to estimate the post-construction configuration of the groundwater surface, the staff issued RAI 17, Question 02.04.12-19. In a September 30, 2010, response, the applicant provided information about its conceptual model of the post-construction conditions, backfill materials, site grading and drainage, ground cover and stormwater management, and post-construction groundwater flow conditions and maximum groundwater level.

The applicant reiterated that the site conceptual model would be consistent with the Piedmont Master Conceptual Model of LeGrand. In that conceptual model, groundwater is controlled by surface drainages and a two-layer slope-aquifer system in which the aquifer consists of residual soil and saprolite overlying weathered and unweathered bedrock.

The applicant stated that the excavation would be filled with engineered backfill around each of the two nuclear islands and extending outward to form the foundation support of the adjacent buildings. Although the exact properties of the engineered fill are yet to be determined, the applicant stated that the hydraulic conductivity would be 10 to 100 times greater than that of in situ residual soil and saprolite. In WLS COL FSAR Section 2.5.4, the applicant stated that the hydraulic conductivity of the engineered fill would fall between 0.00501 cm/s (14.2 ft/day) and 0.0750 cm/s (212.6 ft/day). This range of values is greater than the conservative value of 0.00140 cm/s (3.97 ft/day) for the hydraulic conductivity of the PWR. The applicant provided WLS COL FSAR Figure 2 to show the area receiving the engineered fill. Backfill in the areas beyond the engineered fill would be compacted residual soil and saprolite material with conductivity slightly lower than undisturbed residual soil and saprolite, so somewhat less than 1.1×10^{-4} cm/s (0.32 ft/day). The staff considers these values sufficiently similar to values for the surrounding material such that groundwater conditions would not be appreciably altered.

The applicant stated that the surface around the WLS site is relatively flat and gently slopes away from the plant. The applicant stated that surface topography would be graded to facilitate stormwater runoff away from safety-related structures.

The applicant stated that the 179.2 m (588 ft) above MSL contour that surrounds the nuclear power block areas would enclose an area of 26.5 ha (65.4 ac). Of that, 15.4 percent (4.1 ha (10.1 ac)) would be impervious surfaces such as roads and parking lots and 20 percent (5.2 ha (12.9 ac)) would be semi-impervious compacted gravel (or similar hardscaped material) that reduces infiltration and promotes runoff. Stormwater runoff from the impervious and semi-impervious areas would be collected in the DRS and routed away from the nuclear power block area to reduce the potential for flooding. About 11.8 percent (3.1 ha (7.7 ac)) of the area would be buildings. Precipitation that falls on those buildings would be collected by a roof drain collection system and routed through downspouts into the DRS piping network. The remaining area, about 53 percent (14 ha (34.7 ac)), would be a grass surface cover. The applicant provided a figure that shows that the grass area would surround, but not occur within, the nuclear power block area.

The applicant considered post-construction groundwater conditions and concluded that the structures would not significantly affect groundwater flow. The applicant stated that the high conductivity of the engineered backfill would help equilibrate any local groundwater perturbation. The applicant reiterated that Cherokee-era drains that could affect groundwater would be removed. The applicant concluded that the post-construction groundwater flow would return to conditions consistent with the Piedmont Master Conceptual Model. The applicant proposed to update the WLS COL FSAR with this information about the data and methods used to estimate the post-construction configuration of the groundwater surface. The staff confirmed that the changes were included in WLS COL FSAR.

The staff reviewed the material provided by the applicant. The key reference, LeGrand, provides a synthesis of knowledge related to natural conditions in the Piedmont and can be used to understand the hydrology of the WLS site prior to Cherokee construction. LeGrand does not provide guidance when a site has been significantly altered. The cut-and-fill operations that occurred during Cherokee construction reworked the topography and hydrogeology so extensively that it no longer resembles a typical Piedmont setting. The buildings and extensive impervious surfaces of the WLS site will further diminish the resemblance to a typical Piedmont setting. Thus, generalizations based on LeGrand are not entirely applicable to the WLS site.

The staff considered the post-construction surface conditions provided by the applicant. Runoff and roof drainage are routed to the DRS network, but the terminus of the network is unknown. The termination of the DRS network in the drainages that surround the plant could lead to increased recharge in these areas. The applicant stated that 20 percent of the nuclear power block area would be semi-impervious and 53 percent of the area around the nuclear power block would be grass, but no estimate of recharge was provided for either area. Given the properties of residual soil and saprolite, an average recharge rate of 5.1 cm/yr (2 in/yr) could be sufficient for groundwater to rise to the surface.

The applicant proposed to update the WLS COL FSAR with respect to water table elevation calculations, backfill properties, qualitative descriptions of surface conditions, and a description of a roof drainage system. The staff reviewed the WLS COL FSAR and confirmed that the applicant made the proposed changes. However, staff considered the issue of the maximum water table elevation unresolved; with the resolution of this issue is discussed below.

Recharge Rates and Initial Modeling of Groundwater Elevations

The staff requested estimates of the maximum post-construction groundwater level that is based on anticipated post-construction recharge rates associated with the main surface features and potential groundwater mounds beneath the cooling towers and drainage ditches to better understand how the post-construction groundwater surface would respond.

To obtain additional information needed for the staff's review of the maximum groundwater level, the staff issued RAI 94, Question 02.04.12-20. In a May 18, 2011, response, the applicant provided information to support the view that the post-construction groundwater elevation would not exceed 179.2 m (588 ft) above MSL. The new information consisted of (1) estimates of annual average and maximum recharge rates, (2) a one-dimensional (1-D) analytical method to estimate water table fluctuations in response to recharge, and (3) a quasi two-dimensional (2-D) semi-analytical method to estimate water table fluctuations in response to recharge.

The applicant used annual values of precipitation, evaporation, and runoff to estimate an average annual recharge rate of 13 cm/yr (5.1 in/yr) and a maximum post-construction recharge rate of 21 cm/yr (8.2 in/yr). Recharge is a very site-specific process and the rate of recharge must be estimated for specific soil and vegetation conditions. For example, recharge into an unvegetated surface material (e.g., each of the semi-impervious materials described in the WLS COL FSAR) is expected to differ from recharge into grass-covered fill material. Recharge is also dependent on event-specific weather conditions and could be underestimated if processes such as precipitation, evaporation, runoff, and transpiration are represented as annual average values. Given the information presented by the applicant, the staff could not determine whether the proposed values are appropriate to represent the post-construction conditions at the proposed WLS site.

The applicant used a 1-D analytical method based on specific yield to estimate groundwater fluctuation. The applicant noted that the specific yield estimate for well MW-1215 was 0.0041, but claimed that it would underestimate recharge and substituted effective porosity for specific yield in the analysis. For fill, soil/saprolite, and PWR, the effective porosities are 0.09, 0.2, and 0.08, respectively. Using the average effective porosity of 0.145 for fill and soil/saprolite ((0.09+0.20)/2) = 0.145), the COL applicant estimated a 1.43 m (4.7 ft) groundwater rise for the maximum recharge event of 21 cm (8.2 in.) (i.e., the entire annual recharge amount occurred in a single event). Assuming the event occurred when the water table started at 175.7 m (576.5 ft) above MSL (pre-construction elevation in excavation), the final groundwater elevation was estimated by the applicant to be 177.2 m (581.2 ft) above MSL. Had the applicant used the actual specific yield of 0.0041 (from well MW-1215), the estimated rise in groundwater level would have been more than 30 m (100 ft) in excess of the AP1000 DCD level of 179.2 m (588 ft) above MSL. The applicant did not address possible variations in specific yield and did not justify the conservativeness in its estimate of specific yield. The staff identified additional questions relating to specific yield or effective porosity is the appropriate parameter by which to characterize water table fluctuation when groundwater as near the soil surface.

The applicant also used a 2-D semi-analytical method described by Park and Parker to estimate groundwater fluctuations in response to precipitation that varies over time. The applicant applied the method by representing the maximum recharge event (21 cm/yr (8.2 in/yr)) with four 5.21 cm (2.05 in.) events on four separate days distributed somewhat evenly throughout the year. The applicant estimated the model parameters but did not calibrate the model to the actual WLS site. The applicant estimated the maximum water table rise to be 0.37 m (1.2 ft). The applicant’s analysis appeared to assume that temporary mounding above the AP1000 DCD value for maximum groundwater level was not a problem, whereas the regulatory requirement is that the maximum groundwater elevation must be less than AP1000 DCD value at all times.

<u>Groundwater Modeling Using MODFLOW: Model Conditions</u>

After the initial estimation of maximum groundwater elevations as described above, the applicant estimated post-construction groundwater elevations by conducting computer modeling using the MODFLOW program.

Additional information relevant to groundwater modeling was provided in the applicant’s November 22, 2011, response to RAI 484, Question 10.04.05-2. Although this RAI requested information about Standard Review Plan (SRP) Section 10.04.05, Circulating Water System, the response also contained information about changes to site surface grading and drainage. This information is relevant because the configuration of the land surface is one of the inputs for groundwater modeling. In this response, the applicant stated that the ridge to the northwest of Unit 1 and the two cooling-tower berms would be removed. Site grading in the area of the nuclear power block would be reshaped to promote drainage away from the nuclear island. The surface would grade down to the 178.8 m (586.5 ft) elevation of the vehicle barrier system that would surround the two units. Beyond the vehicle barrier system, the surface would grade down 0.3 to 0.5 m (1 to 1.5 ft) before it engages steeper slopes to the adjacent water bodies.

The applicant provided a second response to RAI 94, Question 02.04.12-20, on November 22, 2011, in which the applicant used groundwater simulations to demonstrate that the AP1000 DCD requirement regarding the maximum groundwater level would be met. The applicant conducted seven groundwater simulations using MODFLOW 2000, Version 1.19.01, embedded in the pre- and post-processing package called Groundwater Vistas.

For the simulations, the applicant established an initial potentiometric surface, an extreme precipitation event, model domain, material properties, boundary conditions, a base case and set of sensitivity cases, and six observation points around the perimeter of both units.

Based on the new site grading and drainage plan, the placement of impervious surfaces, and knowledge of groundwater hydrology in the Piedmont and at the WLS site, the applicant updated the post-construction potentiometric surface map. The new map shows groundwater levels grading from 176.8 m (580 ft) above MSL along the south end of the reactor area to 173.7 m (570 ft) above MSL along the north end.

For the extreme event, the applicant chose Tropical Storm Jerry, which occurred in August 1995. The total precipitation received at the Greenville-Spartanburg Airport during the 47-hour storm was 36.75 cm (14.47 in.). During a single day of that storm, 31.29 cm (12.32 in.) of rain was received, which is the highest 24-hour total in the 45-year record and far exceeds the next highest 24-hour amount of 15.77 cm (6.21 in.) received in September 1972.

The applicant set up the model domain as a square, 914 m (3,000 ft) on each side, centered on the two units and encompassing the vehicle barrier system. Table 2.4.12-2 of this report shows the hydraulic parameters were assigned according to the geologic descriptions in the WLS COL FSAR. For each material, the applicant used the median values for hydraulic conductivity and specific yield. For building foundations and vehicle barrier system, the applicant assumed values that were much lower than those for the geologic materials.

Table 2.4.12-2 Hydraulic Parameters Used in the MODFLOW Simulations

Material	Hydraulic Conductivity (cm/s [ft/day])	Specific Yield (-)
Building Foundations	3.53×10^{-8} (0.0001)	0.001
Granular Backfill	0.011 (31.18)	0.20
Soil Backfill	5.39×10^{-5} (0.1528)	0.09
Soil/Saprolite	1.14×10^{-4} (0.3232)	0.20
Partially Weathered Rock	1.53×10^{-4} (0.4337)	0.08

The boundary conditions were defined for the base of the domain, the sides (groundwater inflow and outflow), and the top in the form of recharge. The base of the model domain was assumed to be no flow. The sides of the model domains were represented with a constant head boundary on the south side of the domain and general head boundaries along the remainder of the domain.

Water input at the top of the model domain was defined by the precipitation event and the runoff-recharge relationship for each surface condition. The precipitation event was modeled as three events. First, a 47-hour storm was applied at rates equivalent to 40 percent of those for Tropical Storm Jerry. Second, a period of 72 hours with no precipitation was imposed to allow the groundwater to equilibrate to the precipitation just received. Finally, the hourly precipitation record of Tropical Storm Jerry was applied.

Precipitation received at the surface was partitioned into runoff according to methodology described by the USDA. The USDA method relies on knowledge of the surface condition, soil type, and vegetation. For each of the 22,576 model cells, the applicant assigned one of the following surface types: buildings; roads; vehicle barrier system; hardscape; grass; and brush. For each surface type, the applicant selected a runoff curve number from the tables provided by USDA and used it to represent the percentage precipitation that would run off. In the case of the grass cover, the applicant added the runoff from buildings, roads, and vehicle barrier system to the amount of precipitation before calculating runoff from grass.

The applicant assumed no interim water storage on the surface such that water that did not run off infiltrated the surface. The applicant assumed no evapotranspiration and no time delay as water moved through the vadose zone to the groundwater. Thus, the applicant used the infiltration rate to define the upper recharge boundary condition.

Groundwater Modeling Using MODFLOW: Results of Simulations

The applicant conducted seven simulations. For the base case (Run 1), it used the projected post-construction potentiometric surface, median hydraulic parameters, and Type B soils (per USDA). The six sensitivity cases examined the impact of a higher initial potentiometric surface (Run 2), lower conductivity values (Run 3), minimum specific yield values (Run 4), maximum specific yield values (Run 5), Type A (instead of Type B) soils (Run 6), and a combination that included lower conductivities and specific yields and Type A soils (Run 7).

The base case results showed that the highest groundwater elevations occurred at the two observation points on the south side of the units. The highest elevation, 177.4 m (582.2 ft) above MSL, occurred just to the southwest of Unit 1. Just to the north of the units, the groundwater elevations are around 175.3 m (575 ft) above MSL. For the sensitivity cases, the highest groundwater elevations always occurred at the two southern observation points. Runs 2 and 7 yielded the highest overall elevations, 179.0 and 179.1 m (587.2 and 587.5 ft) above MSL, respectively. The applicant stated that its analysis demonstrated compliance with the AP1000 DCD site parameter criteria for maximum groundwater level of 149 m (588 ft) above MSL. The applicant provided a copy of all MODFLOW input files to the staff.

The staff examined the proposed site grading and drainage plans and believes it would significantly enhance the ability of the site to increase runoff, reduce recharge, and reduce the potential for groundwater to rise above the AP1000 DCD limit of 179.2 m (588 ft) above MSL. High elevations to the northwest, west, and east that could have increased groundwater levels have been removed. Surface topography slopes down from the units in all four directions, thus facilitating runoff removal in all directions. The maximum elevation of the surface drainage to the southwest is 178.8 m (586.5 ft) above MSL; this ensures that higher groundwater levels further south would not propagate northward to the nuclear islands.

The staff examined the proposed post-construction potentiometric surface. In contrast to the previous map, the new map does not show two groundwater divides on either side of the reactor area. Consistent with that change, the new groundwater levels are about 1.5 m (5 ft) lower than previous estimates. The staff notes that the proposed map appears to be reasonable and consistent with knowledge of the area.

The staff examined the MODFLOW analysis. The use of the precipitation record from Tropical Storm Jerry to evaluate maximum water table rise is appropriate. The event far exceeds the 100-year storm of 18 cm (7.2 in.) predicted for the Cherokee County. Including a pre-storm that adds 40 percent of tropical storm Jerry precipitation adds conservatism. In fact, the total precipitation added (51.46 cm (20.26 in.)) in the 7-day simulation far exceeds all monthly precipitation records.

The staff examined the model domain. Cell sizes are smaller in the reactor areas and larger away from the reactors. The extent of the domain does not extend to natural boundaries such as Make-Up Pond B and the Broad River. Instead, the applicant chose to focus the analysis on a smaller domain. Doing so required establishing head boundary conditions at interim locations. The analysis only addresses a short period of 7 days. In that time, any effects from lateral boundary conditions would not be discernible. As a result, the staff performed confirmatory analyses as described below.

The staff examined the material properties and confirmed they are reasonable to somewhat conservative with respect to hydraulic conductivity. The applicant's groundwater modeling was based on the median hydraulic conductivity for the well-graded gravel granular material (1.1×10^{2} cm/sec (11,381 ft/year)). In WLS COL FSAR Table 2.5.4-211, the minimum value of hydraulic conductivity reported for the poorly graded gravel granular fill materials is stated as "<~5.0E-03" cm/sec ("<~5,173" ft/year). The MODFLOW sensitivity case Run 3 examined the impact of using minimum conductivity values. The result was an increase in the maximum groundwater elevation to 177.2 m (581.47 ft), which is only 0.02 m (0.08 ft) above the base case (Run 1) and far below the assumed maximum groundwater elevation of 178.0 m (584 ft).

The staff examined the boundary conditions. Using a no-flow bottom boundary is consistent with data from the site. Even if there was a small amount of flow, up or down, the fluxes would be too small to affect a 7-day simulation. Therefore, the lateral boundary conditions are not ideal, so the staff ran the base case with different lateral boundary conditions and confirmed that these boundaries do not measurably affect the results of the short 7-day simulations.

The staff examined the methodology used to calculate runoff and establish recharge rates. It appears to the staff that the applicant treated the runoff curve numbers as runoff coefficients (i.e., percentages) rather than as parameters in the USDA's Equation 2.3. The staff used Equation 2.3 to calculate the true runoff percentage, which yielded a runoff fraction that was slightly higher than the value determined by the applicant. The applicant's calculation resulted in slightly higher recharge rates, which its analysis revealed would not cause the site to exceed the AP1000 DCD requirement. Repeating the applicant's analysis with the slightly lower but correct recharge rates would yield the same result.

Table 2.4.12-3 of this report lists the runoff coefficients used to establish recharges rates for the MODFLOW simulations. The staff examined the applicant's choices of runoff curve numbers to represent surface conditions at the site and identified some differences from how the staff would parameterize the site. The practice is to route the runoff from buildings and roads to hardscape, and runoff from hardscape to grass.

Using the runoff curve numbers and runoff routing method described above, the staff repeated the base-case simulation. The results at the observation points showed very little difference from the applicant's results. This lack of sensitivity is due in part to the relatively low recharge rates in the hardscape and to the high conductivity of the engineered fill, which quickly dissipates local groundwater mounds caused by variations in recharge rates.

The staff's review confirmed that groundwater would not exceed an elevation of 179.2 m (588 ft) MSL. As noted above, because recent changes to the site grading yielded a higher plant elevation, the current WLS COL FSAR requires the maximum groundwater elevation to be less than 180.1 m (591 ft). Since all other conditions remain the same, the staff affirms that the site groundwater would not exceed an elevation of 180.1 m (591 ft) MSL.

In summary, the applicant described changes to the site grading and drainage plan and provided groundwater modeling results to support its assessment that the site meets the AP1000 DCD requirement relative to the maximum groundwater elevation. The staff evaluated the site grading changes and the modeling analysis and conducted confirmatory modeling analyses. Since the applicant used a very extreme storm event, did not account for evaporation or transpiration, and did not account for runoff that would be routed well away from the site before it could infiltrate, the staff concluded that the site would be able to meet the AP1000 DCD requirement. The staff notes that the applicant included a description of the latest site grading and drainage plan, the vehicle barrier system, and the groundwater modeling analysis in WLS COL FSAR Section 2.4.12.2, "Sources." The staff concluded that the applicant's responses to RAI 826, Questions 02.04.12-6, and 02.04.12-14; RAI 17, Question 02.04.12-19; RAI 94, Question 02.04.12-20; and RAI 484, Question 10.04.05-2, were acceptable. Accordingly, the staff considers these questions resolved.

Table 2.4.12-3 Runoff Curve Numbers Used to Estimate Recharge

Surface Condition	Runoff Curve Numbers				Comments
	Applicant		NRC Staff		
	Type B Soils	Type A Soils	Type B Soils	Type A Soils	

Buildings	100	100	100	100	A value of 100 is not provided in USDA (1986), but staff judged it to be sufficient given the stated intent of the applicant to manage all precipitation that falls on buildings. The applicant routed runoff to grass, the staff routed runoff to hardscape
Roads	100	98	89	83	The applicant used USDA (1986) values associated with curbed roads; the staff used values associated with uncurbed roads. The applicant routed runoff to grass; the staff routed runoff to hardscape
Hardscape	85	76	85	76	Both the applicant and the staff routed runoff to grass
VBS	100	98	98	98	The applicant used value of 100 for Type B soil; the staff used value of 98. Both the applicant and the staff routed runoff to grass
Brush	82	72	48	30	The applicant used values for dirt (Table 2-2a, USDA 1986); the staff used values for brush (Table 2-2c, USDA 1986)
Grass	61	61	61	39	Applicant used value for Type A soil that is not consistent with USDA (1986)

Monitoring

<u>*Information Submitted by the Applicant*</u>

The applicant stated that a groundwater monitoring program would be developed. The applicant provided a list of three areas within the site to be considered for monitoring and a list of generalized considerations for implementation of the program. In response to a staff RAI requesting details, the applicant provided additional information about post-construction monitoring plans to reduce uncertainties in groundwater flow paths used for WLS COL FSAR Section 2.4.13.

<u>*NRC Staff's Technical Evaluation*</u>

In the December 11, 2008, response to RAI 826, Question 02.04.12-11, the applicant described post-operational monitoring activities that address traditional monitoring goals as well as goals related to reducing uncertainties about the plausible groundwater pathways. The activities include placement of near-field and far-field wells to detect early releases and to verify that there is no offsite migration, and placement of both shallow and deep wells to monitor plausible flow paths close to the facilities. The applicant stated that the post-construction groundwater monitoring program would be consistent with the guidance provided by the Nuclear Energy Institute (NEI 2009), which references NEI and EPRI. The staff reviewed those references and confirmed they would form the basis for an adequate monitoring program. The staff notes that the applicant included this monitoring information in WLS COL FSAR Section 2.4.12.4, "Monitoring or Safeguard Requirements." The staff concluded that the applicant's December 11, 2008, response to RAI 826, Question 02.04.12-11, was acceptable and considers the question resolved.

2.4.12.5 *Post Combined License Activities*

For the reasons discussed in the technical evaluation section above, the staff proposes to include the following license condition:

- License Condition (2-1) – Prior to fuel load, the licensee shall confirm that a single legacy Cherokee project stormwater drain line (designed to transfer stormwater from the Cherokee power block area to Hold-Up Pond A) and any associated bedding material representing a potential preferential groundwater pathway have been removed and the excavation has been backfilled with compacted native soils.

2.4.12.6 *Conclusion*

The staff reviewed the application and confirmed that the applicant addressed the information relevant to groundwater, and that there is no outstanding information required to be addressed in the WLS COL FSAR related to this section. As set forth above, the applicant presented and substantiated information to establish the site description. The staff reviewed the information provided and, for the reasons given above, concludes that the applicant has provided sufficient details about the site description to allow the staff to evaluate, as documented in Section 2.4.12, herein, whether the applicant has met the relevant requirements of 10 CFR 52.79(a)(1) and 10 CFR Part 100 with respect to determining the acceptability of the site. This applicant addressed groundwater in COL Information Item 2.4-4. The staff concludes that the applicant provided sufficient information on groundwater characteristics to satisfy the applicable requirements of 10 CFR Part 52 and 10 CFR Part 100.

2.4.13 Accidental Release of Radioactive Liquid Effluent in Ground and Surface Waters

2.4.13.1 *Introduction*

This section of the WLS COL FSAR provides a characterization of the attenuation, retardation, dilution, and concentrating properties governing transport processes in the surface-water and groundwater environment at the site. The goal of this section is not to provide an assessment of the effects of a specific release scenario but to provide a suitable conceptual model of the hydrological environment for other assessments. Since it would be impractical to characterize all the physical and chemical properties (e.g., hydraulic conductivities, porosity, and mineralogy) of a time-varying and heterogeneous environment, the section characterizes the environment in terms of the projected transport of a postulated release of radioactive waste. The accidental release of radioactive liquid effluents in ground and surface waters is evaluated using information about existing uses of groundwater and surface water and their known and likely future uses as the basis for selecting a location to summarize the results of the transport calculation. The source term from a postulated accidental release is reviewed under NUREG-0800, Section 11.2, following the guidance in Branch Technical Position (BTP) 11-6, "Postulated Radioactive Releases Due to Liquid-containing Tank Failures."

The source term is determined from a postulated release from a single tank outside of the containment. The results of a radionuclide transport analysis are evaluated against SRP Section 11.2 and BTP 11-6 guidance and effluent concentration limits (ECLs) as acceptance criteria to meet the requirements of 10 CFR Part 20, Appendix B, as SRP acceptance criteria. Under SRP guidance, the effluent concentration limits of 10 CFR Part 20, Appendix B, are applied as acceptance criteria only for the purpose of assessing the acceptability of the results of the consequence analysis and are not intended for demonstrating compliance with ECLs.

The following specific areas are reviewed by the staff: (1) alternative conceptual models of the hydrology at the site that reasonably bound the site's hydrogeological conditions to the degree that these conditions affect the transport of radioactive liquid effluent in the groundwater and surface-water environment; (2) a bounding set of plausible surface and subsurface pathways from potential points of an accidental release to determine the critical pathways that may result in the most severe effect on existing uses and known and likely future uses of groundwater and surface-water resources in the vicinity of the site; (3) the ability of the groundwater and surface-water environments to delay, disperse, dilute, or concentrate accidentally released radioactive liquid effluents during transport; and (4) the assessment of scenarios wherein an accidental release of radioactive effluents is combined with potential effects of seismic and non-seismic events.

2.4.13.2 *Summary of Application*

This section of the WLS COL FSAR addresses the accidental release of radioactive liquid effluents in groundwater and surface waters. The applicant addressed information related to accidental release of radioactive liquid effluents as follows:

AP1000 COL Information Item

- WLS COL 2.4-5

In addition, this section addresses the following COL-specific information identified in AP1000 DCD Section 2.4.1.5, Revision 19.

> Combined License applicants referencing the AP1000 certified design will address site-specific information on the ability of the ground and surface water to disperse, dilute, or concentrate accidental releases of liquid effluents. Effects of these releases on existing and known future use of surface water resources will also be addressed.

2.4.13.3 *Regulatory Basis*

The relevant requirements of NRC regulations for the pathways of liquid effluents in ground and surface waters, and the associated acceptance criteria, are specified in NUREG-0800, Section 2.4.13.

The applicable regulatory requirements for evaluating accidental release of radioactive liquid effluents in ground and surface waters are set forth in the following:

- 10 CFR 52.79(a)(1)(iii), as it relates to identifying hydrologic site characteristics with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

- 10 CFR Part 100, as it relates to identifying and evaluating hydrological features of the site. The requirement to consider physical site characteristics in site evaluations is specified in 10 CFR 100.20(c).

The staff also used the acceptance criteria identified in NUREG-0800, Section 2.4.13:

- Alternate Conceptual Models: Alternate conceptual models of hydrology in the vicinity of the site are reviewed.

- Pathways: The bounding set of plausible surface and subsurface pathways from the points of release are reviewed.

- Characteristics that Affect Transport: Radionuclide transport characteristics of the groundwater environment with respect to existing and known and likely future users should be described.

- Consideration of Other Site-Related Evaluation Criteria: The COL applicant’s assessment of the potential effects of site-proximity hazards, seismic, and non-seismic events on the radioactive concentration from the postulated tank failure related to accidental release of radioactive liquid effluents to ground and surface waters for the proposed plant site is needed.

- BTP 11-6 provides guidance in assessing a potential release of radioactive liquids after the postulated failure of a tank and its components, located outside of containment, and effects of the release of radioactive materials at the nearest potable water supply, located in an unrestricted area, for direct human consumption or indirectly through animals, crops, and food processing.

In addition, the hydrologic characteristics should conform to appropriate sections from RG 1.113, “Estimating Aquatic Dispersions of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I.”

The staff used best current practices to analyze groundwater transport of radioactive liquid effluents. The regulatory basis of the information incorporated by reference is addressed in the staff's FSER for the AP1000 certified design, NUREG-1793.

2.4.13.4 *Technical Evaluation*

The staff reviewed WLS COL FSAR Section 2.4.13 Revisions 0 to 11; corrections and additions to the WLS COL FSAR submitted by the applicant as letters; and associated applicant responses to RAIs issued by the staff. The staff also checked the referenced AP1000 DCD and departures and supplements specified in the WLS COL FSAR.

The staff reviewed COL Information Item 2.4-5 related to the provision of site-specific information about accidental release of liquid effluents into ground and surface water at the plant site included under WLS COL FSAR Section 2.4. Additional aspects of this information item are addressed in Section 2.4.12 of this report.

The staff reviewed the specific items related to the assessment of an accidental release of radioactive liquid effluents in groundwater and surface water included in the WLS COL FSAR and associated RAI responses. To improve readability, the staff's discussion of these items is organized into the following sections, which correspond to the sections of WLS COL FSAR Section 2.4.13, Revision 11.

Groundwater

<u>*Information Submitted by the Applicant*</u>

In WLS COL FSAR Section 2.4.13, the applicant provided information that described movement of accidentally released effluents from the nuclear island area to the nearest potable water supply in an unrestricted area. The applicant identified the groundwater pathway from Unit 2 to Hold-Up Pond A (Pathway 1) as the bounding pathway because of its higher groundwater velocity and shorter travel time. The failure of a Unit 2 effluent hold-up tank, located below ground level in the auxiliary building, was identified as the appropriate scenario for analysis of a postulated accidental release of radioactive liquid effluent. The radioactive source term for this release was described based on AP1000 DCD, Tables 11.1-2 and 11.1-8. The conceptual model of the release is that one of the effluent hold-up tanks, having a capacity of 106,000 L (28,000 gal), ruptures and releases 80 percent of the volume, which equates to 84,800 L (22,400 gal) of liquid effluent. Radionuclides in the effluent travel with groundwater to Hold-Up Pond A where they enter Hold-Up Pond A and flow directly into the Broad River. The applicant used RESRAD-OFFSITE Version 2.0 to calculate the transport rates and determine the radionuclide concentrations in the Broad River.

The applicant stated that three soil samples were collected from a depth range of 13.7 to 22.3 m (45 to 73 ft) below ground in two wells. The samples were analyzed for soil distribution coefficients (i.e., K_d values) for colbalt (Co)-60, cesium (Cs)-137, iron (Fe)-55, iodine (I)-129, nickel (Ni)-63, plutonium (Pu)-242, strontium (Sr)-90, technetium (Tc)-99, and uranium (U)-235.

In response to three staff RAIs, the applicant provided additional information and clarification related to the K_d values for the three main geologic materials.

NRC Staff's Technical Evaluation

The staff reviewed the applicant's information about accidental releases. Because of the subsurface location of the release, transport by surface water away from the release location was not considered feasible, and was not considered further. The staff reviewed the applicant's information regarding the groundwater pathway, and concluded that additional information was required to complete its evaluation of the risks from radionuclide transport by this pathway.

To clarify the suitability of the onsite K_d measurements for the evaluation of accidental releases, the staff issued RAI 828, Question 02.04.13-10. In the November 25, 2008, response, the applicant described the soil K_d values with respect to the effluent chemistry, site geochemistry, and the relationship to radionuclide migration. The applicant stated that the water in the effluent hold-up tanks is slightly acidic, has a TDS content of less than 1 ppm, and would not vary significantly from the ambient groundwater conditions in pH, salts, metals, or organics. Given those conditions, the applicant stated that the effluent would not alter the groundwater chemistry outside the range under which the K_d values were determined. The applicant stated that the K_d values determined for three samples at the WLS site are sufficient to represent the range of values within each of the alternative flow pathways. The applicant bounded its calculations by using the lowest measured K_d value regardless of media. The applicant used a K_d value of 0 for tritium and stated that tritium contributes 99 percent of the dose. The applicant also analyzed the sensitivity of the RESRAD-OFFSITE results to variations in K_d values. The staff evaluated the RESRAD-OFFSITE analyses and confirmed the results. The applicant updated the WLS COL FSAR to explain in more detail the nature of the K_d values and to describe the results of the sensitivity tests. The staff concluded that the applicant's November 25, 2008, response to RAI 828, Question 02.04.13-10, was acceptable and considers the question resolved.

The staff noted that the three samples tested included one from fill material and two from soil/saprolite. There were no measurements of K_d in the PWR, which is the material considered in the most conservative pathway. To evaluate the significance of this data gap, the staff issued RAI 828, Question 02.04.13-12. In a November 25, 2008, response, the applicant stated that PWR samples were not analyzed because of the difficulty of working with a representative sample. Therefore, the applicant assumed that K_d values for fill and soil/saprolite were reasonable alternatives for PWR. Partly because of this assumption, the applicant evaluated the sensitivity of its analysis to variations in the distribution coefficients (see Section 2.4.13.4.5 herein). The staff considered this to be a reasonable approach. The staff notes that the applicant included this information in WLS COL FSAR Section 2.4.13.1, "Groundwater." The staff concluded that the applicant's November 12, 2009, response to RAI 828, Question 02.04.13-12, was acceptable and considers the question resolved.

The staff noted that, in the proposed revision to Section 2.4.13.4, the applicant referred to the "lowest uncertainty corrected K_d values." In a broad sense, the term "uncertainty" was intended to capture all aspects that contribute to uncertainty, many of which would never be known. The applicant defined the conservative K_d value as the lowest of three (3) measured K_d values minus one standard deviation. Calling any value the "lowest uncertainty corrected value" implies that all uncertainty has been removed and that the value is known with certainty. That is not the case for K_d values. To clarify the description of K_d, the staff issued RAI 73, Question 02.04.13-23. In a November 12, 2009, response, the applicant stated that the K_d values are indeed the mean values minus one standard deviation and concurred that the phrase "uncertainty corrected" could be misinterpreted. The staff notes that the applicant included this information in WLS COL FSAR Section 2.4.13.1, "Groundwater." The staff concluded that the applicant's November 12, 2009, response to RAI 73, Question 02.04.13-23, was acceptable and considers the question resolved.

Accident Scenario

Information Submitted by the Applicant

The applicant described the limiting accident scenario as the failure of a Unit 2 effluent hold-up tank, located in the Unit 2 auxiliary building, and resulting transport of radioactive contaminants to the Broad River. The applicant chose this scenario because the hold-up tank of Unit 2 was closest to the closest point of exposure: Hold-Up Pond A and the Broad River. The applicant ruled out other tanks because they had lower volumes and lower isotope inventories relative to the effluent hold-up tanks. Following BTP 11-6, March 2007, the applicant assumed that 80 percent of the effluent tank capacity (a release volume of 84,800 L (22,400 gal)) was immediately released through cracks in the auxiliary building walls and floor into the surrounding subsurface soil.

In response to three staff RAIs, the applicant provided additional information and clarification related to the identification of plausible alternative conceptual models.

NRC Staff's Technical Evaluation

The staff reviewed the applicant's information about the accidental release scenario. This information identified the assumed source of the release and the groundwater pathway that the applicant believed would be most probable. The staff identified certain missing or incomplete information needed for evaluation of the applicant's analysis of the accident scenario.

To understand the process used to identify subsurface pathways that affect the transport of radioactive liquid effluents so as to ensure that the most conservative of plausible conceptual models has been identified, the staff issued RAI 828, Question 02.04.13-3. In a November 25, 2008, response, the applicant increased the number of plausible flow paths from two to five (in subsequent WLS COL FSAR updates, the applicant reduced the number of flow paths from five to four). Despite the greater number of plausible pathways, the applicant stated that Pathway 1, the pathway that had been chosen previously for the RESRAD-OFFSITE modeling analysis, remained the most conservative pathway. Therefore, the applicant determined that a RESRAD-OFFSITE analysis of another pathway was not necessary. The staff concluded that this explanation was reasonable. The staff notes that the applicant included this information in WLS COL FSAR Section 2.4.13.3, "Groundwater Movement." The staff concluded that the applicant's November 25, 2008, response to RAI 828, Question 02.04.13-3, was acceptable and considers this question resolved.

To understand the impact of the post-construction water table (which may differ from the pre-construction water table) on the selection of alternative pathways, the staff issued RAI 828, Question 02.04.13-18. The staff examined the November 25, 2008, response and concluded that the process and the methods used by the applicant to determine the bounding set of plausible surface and subsurface pathways were reasonable. As discussed in WLS COL FSAR

Section 2.4.12, the applicant has generated a post-construction water table that is consistent with the recharge-affecting surface conditions. The new estimate of the post-construction water table does not alter the potential pathways and thus does not change the selection of the primary pathway for the transport analysis, which is from Unit 2 to the Broad River. The staff concluded that the applicant's November 25, 2008, response to RAI 828, Question 02.04.13-18, was acceptable and considers the question resolved.

The staff examined the responses to these RAIs and determined that the process and the methods used by the applicant to determine alternate conceptual models of the site hydrology are reasonable but potentially incomplete. The applicant did not evaluate the impact of a failure of the dams associated with Make-Up Ponds A and B. Such an event could increase the hydraulic gradient substantially and shorten travel times. The applicant did not evaluate alternative geohydrologic features such as continuous PWR along all pathways. The applicant did not evaluate the potential for preferential flow paths (e.g., buried pipes, or coarse bedding material beneath them) created by Cherokee construction activities. To address these issues, the staff issued RAI 73, Question 02.04.13-19.

In a November 12, 2009, response, the applicant stated that the transport calculation was changed to assume PWR occurred along all potential transport pathways. This change eliminated any concern that the occurrence of PWR would be underestimated. The changed calculation also led the applicant to confirm the Pathway 1 (from Unit 2 to Hold-Up Pond A) as the limiting pathway. The applicant considered the impact of dam failure and demonstrated that, because of distance, Pathway 1 would remain the limiting pathway. The applicant acknowledged the potential for Cherokee-era drainage piping to be a preferential flow pathway in the event of an effluent tank release. The applicant reviewed the Cherokee piping system and identified the piping corridor from the nuclear power block area north to Hold-Up Pond A as the only one having the potential to affect flow and transport. The applicant committed to removing this piping system and associated bedding materials. The staff notes that the applicant included this information in WLS COL FSAR Section 2.4.12.2, "Sources." The staff concluded that the applicant's November 12, 2009, response to RAI 73, Question 02.04.13-19, was acceptable and considers the question resolved.

Source Term

Information Supplied by the Applicant

The applicant identified the source-term concentrations per the information in the AP1000 DCD Tables 11.1-8 (H-3), 11.1-2 (corrosion products chromium (Cr)-51, manganese (Mn)-54, manganese (Mn)-56, Fe-55, Fe-59, Co-58, and Co-60), and 11.1-2 for the other isotope concentrations after adjusting them by the factor 0.12/0.25 in accordance with BTP 11-6, March 2007. The applicant described the two pathways that were considered in WLS COL FSAR Revision 0. Travel time to the Broad River was shorter (because of hydrogeology) than to Hold-Up Pond A. Unit 2 was closer to the Broad River than Unit 1. Thus, transport of the Unit 2 effluent tank contents directly to the Broad River was identified as the limiting scenario because it minimized the transport distance and time. The applicant assumed the contents of the failed effluent tank entered the subsurface environment at a depth of 10.2 m (33.5 ft) and completely filled the soil pore space in an area large enough to contain 84,791 L (22,400 gal) of effluent.

In response to four staff RAIs, the applicant provided additional information and clarification related to the use of RESRAD-OFFSITE to calculate contaminant transport.

NRC Staff's Technical Evaluation

In RAI 828, Question 02.04.13-14, the staff requested that the applicant provide the calculation package used to convert source concentration values in AP1000 DCD Tables 11.1-2 and 11-1-8 to RESRAD-OFFSITE input values (with units of pCi/g water). In a November 25, 2008, response, the applicant summarized the steps used to convert source concentrations to RESRAD-OFFSITE input values. The activities listed in AP1000 DCD Table 11.1-2 for non-corrosion products were stated to be corrected by 0.12/0.25 as recommended by AP1000 DCD Section 2.1 (the staff noted that the WLS COL FSAR incorrectly cites NRC BTP 11-6 as the source for this correction). The activities for the corrosion products were taken directly from AP1000 DCD Table 11.1-2. The resulting activities for all constituents were adjusted by a factor of 1.01 and then converted from microcuries to picocuries by multiplying by 1×10^6. The applicant proposed no changes to the WLS COL FSAR.

The staff's review of the calculation package showed that the applicant used the uncorrected values, i.e., the applicant did not modify the values in DCD Table 11.1-2 by the factor 0.12/0.25. The applicant did not explain the reason for adjusting concentrations by a factor of 1.01, but such an adjustment increases concentrations slightly and therefore could be viewed as being conservative. The staff noted that the table of concentrations in the RAI response from the applicant contained a xenon (Xe)-133 concentration of 1.2×10^{-2} µCi/g. The AP1000 DCD value is 1.2×10^2 µCi/g. The applicant did not include Xe-133 in its RESRAD-OFFSITE groundwater analysis. The staff issued two more RAIs related to the source term, as discussed below.

In RAI 828, Question 02.04.13-16, the staff requested that the applicant explain more fully the conceptual model of the accidental release with respect to soil volume occupied, duration of leak, and impact on surrounding groundwater. In a November 25, 2008, response, the applicant stated that the release to groundwater was assumed to be instantaneous and that the leaked fluid immediately occupied a rectangular plume volume that was 2.0 m (6.6 ft) high and 6.5 m (21.4 ft) wide on each side. Although the leak was considered to be instantaneous, the applicant assumed that the release mechanism was sufficiently gradual that there was no perturbation to the groundwater surface, flow rate, and flow direction. The applicant assumed that aspects of a real leak event, such as potential groundwater flow into the auxiliary building and the time delay for contaminants to exit the building, were inconsequential; this assumption makes the release analysis more conservative. The applicant proposed no changes to the WLS COL FSAR. The staff considered the assumptions and agreed with the applicant's assessment. The staff concluded that the applicant's November 25, 2008, response to RAI 828, Question 02.04.13-16, was acceptable and considers the question resolved.

To address questions raised by the staff's review of the applicant's response to RAI 828, Question 02.04.13-14, as described above, the staff issued RAI 73, Question 02.04.13-26. In the November 12, 2009, response to RAI 73, Question 02.04.13-26, the applicant stated that it would apply the 0.12/0.25 factor correctly and repeat the RESRAD-OFFSITE analysis. The applicant stated that the modification by a factor of 1.01 is just a conservative adjustment. The staff noted that the table of concentrations in the RAI response from the applicant contained a Xe-133 concentration of 1.2×10^{-2} µCi/g, whereas the correct value from the AP1000 DCD is

1.2 x 10^2 µCi/g. The applicant indicated that the 1.2x10^{-2} value was only a typographical error. The applicant did not include Xe-133 in its RESRAD-OFFSITE groundwater analysis because it is a gas and because it has a short half-life of 0.014 years. The staff noted that the other source-term gases listed in WLS COL FSAR Section 2.4.13.3 also have short half-lives (much less than 0.1 years) except for krypton (Kr)-85, which has a half-life of 10.8 years. However, Kr-85 is a noble gas and would partition rapidly into the atmosphere, both at the time of the release and if it reaches the Broad River. Therefore, significant exposure by way of groundwater and surface water appears unlikely. The staff notes that the applicant updated WLS COL FSAR Section 2.4.13.3, "Source Term." The staff concluded that the applicant's November 12, 2009, response to RAI 73, Question 02.04.13-26, was acceptable and considers the question resolved.

To clarify the handling of tritium in the applicant's evaluation of accidental releases, the staff issued RAI 73, Question 02.04.13-27. In a November 12, 2009, response the applicant stated that the AP1000 vendor (Westinghouse) recognized that tritium was not in AP1000 DCD Table 11.1-2 and indicated that the best available value for tritium was in AP1000 DCD Table 11.1-8, Realistic Source Terms. The staff notes that the applicant updated WLS COL FSAR Section 2.4.13.3, "Source Term." The staff concluded that the applicant's November 12, 2009, response to RAI 73, Question 02.04.13-27, was acceptable and considers the question resolved. The staff also concluded that that the applicant's November 12, 2009, response to RAI 73, Questions 02.04.13-26 and 02.04.13-27, provides information to find the applicant response to RAI 828, Question 02.04.13-14, acceptable and considers the question resolved.

Conceptual Model

<u>*Information Submitted by the Applicant*</u>

The applicant stated in WLS COL FSAR Revision 0 that the conceptual model was conservative because it used the shortest travel time, did not credit dilution in the Broad River, assumed a straight travel path, and assumed the entire domain had the properties of PWR (which has a high permeability). The applicant used RESRAD-OFFSITE to determine radionuclide concentrations in the Broad River for an evaluation period of 50 years. The applicant provided a list of parameters used in its analysis. The applicant calculated the maximum radionuclide concentration for each isotope in the Broad River during the evaluation period and compared it to the limiting concentrations defined in 10 CFR Part 20, Appendix B, Table 2, Column 2. The applicant reported that all concentrations were below the limits. In addition, the applicant calculated the sum of fractions of effluent concentrations using all maximum radionuclide concentrations and showed it to be well below a value of 1.0. The applicant stated that this result was conservative because the maximum concentration of each radionuclide occurred at a different time due to variations in transport time to the Broad River.

In response to 12 staff RAIs, the applicant provided additional information and clarification related to the conceptual model and parameter choices affecting the RESRAD-OFFSITE calculation of contaminant transport.

NRC Staff's Technical Evaluation

The staff reviewed the applicant's conceptual model for groundwater transport of radionuclide contaminants from a release. The staff determined the applicant's conceptual model generally reasonable, but required much additional information and analysis from the applicant to support the staff's review. Therefore, in RAI 828, Question 02.04.13-5, the staff requested that the applicant clarify why different values of porosity were used in each section. If this was done for conservatism, the staff requested that the applicant explain why each is conservative. In a November 25, 2008, response, the applicant set the total porosity of the PWR equal to its estimate of 0.08 for the effective porosity. The applicant stated that the lowest porosity values are the most conservative values because they result in the shortest travel time and, thus, the highest concentration of radionuclides in the receptor water body. The staff notes that the applicant updated WLS COL FSAR Table 2.4.13-203. The staff concluded that the applicant's November 25, 2008, response to RAI 828, Question 02.04.13-5, was acceptable and considers the question resolved.

To clarify the conservatism of the hydraulic conductivity value used for the PWR, the staff issued RAI 828, Question 02.04.13-6. In a November 25, 2008, response, the applicant changed the statement from "highest measured" to "conservative" hydraulic conductivity. The staff notes that the applicant updated WLS COL FSAR Section 2.4.12.2, "Sources." The staff concluded that the applicant's November 25, 2008, response to RAI 828, Question 02.04.13-6, was acceptable and considers the question resolved.

In RAI 828, Question 02.04.13-7, the staff requested that the applicant define the nature of the contaminated zone; namely, describe the materials that make up the contaminated zone. In a November 25, 2008, response the applicant stated that the contaminated zone consists of PWR with some portions of residual soils and fill. The applicant considers use of PWR to represent the most conservative approach for estimating transport. The applicant set both total and effective porosity of PWR to the value of 0.08. This value is much lower than the WLS COL FSAR Revision 0 value of 0.44 and the applicant considers it much more conservative. The staff notes that the applicant updated WLS COL FSAR Table 2.4.13-203 to the revised porosity values for the partially weathered zone. The staff concluded that the applicant's November 25, 2008, response to RAI 828, Question 02.04.13-7, was acceptable and considers the question resolved.

To gain a clearer understanding of the significance of precipitation, runoff, evapotranspiration, and recharge with respect to radionuclide transport, the staff issued RAI 828, Question 02.04.13-11. In a November 25, 2008, response, the applicant revised its estimate of average annual precipitation from 1.01 to 1.23 m/yr (39.8 to 48.4 in/yr) based on precipitation data from Gaffney, SC, for the years from 1944 to 2007. The applicant described the calculation process in which it used the SCS curve number method to estimate the runoff coefficient. For the watershed containing Make-Up Pond A and Hold-Up Pond A, the applicant calculated annual runoff to be 53 percent of precipitation. For the watershed containing Make-Up Pond B, it calculated annual runoff to be 39 percent of precipitation. For the RESRAD-OFFSITE analysis, the applicant used the value of 39 percent for both watersheds, which is realistic for the Make-Up Pond B watershed and conservative for the watershed containing Make-Up Pond A and Hold-Up Pond A. Using this analysis, the applicant revised its runoff coefficient from 0.36 to 0.39.

Given that the average annual precipitation is 123.0 cm (48.41 in.), runoff amounts to 48.0 cm (18.88 in.). The applicant calculated evaporation using regional pan evaporation data, which showed an annual pan evaporation rate of 132 cm (51.8 in.). Pan evaporation data tend to overestimate actual evaporation, so the applicant used a correction factor of 0.7 to adjust the pan data, yielding an estimate of annual evaporation of 91 cm (36 in.). In the RESRAD-OFFSITE analysis, the applicant specified an evaporation factor of 0.74. RESRAD-OFFSITE uses the evaporation factor to estimate how much of the infiltration water (i.e., precipitation - runoff, or P-R) is lost to evaporation (which includes transpiration from vegetation). In this case, annual evapotranspiration (E) is estimated to be 55.5 cm (21.85 in.). Annual net infiltration, also called groundwater recharge, is thus 19.5 cm (7.68 in.) (i.e., (P-R)-E). The staff notes that the applicant updated WLS COL FSAR Table 2.4.13-203 to include the above runoff coefficient and evaporation factor. The staff concluded that the COL applicant's November 25, 2008, response to RAI 828, Question 02.04.13-11, was acceptable and considers the question resolved.

To review the appropriateness of contaminant transport modeling methods used by the applicant, the staff issued RAI 828, Question 02.04.13-13. In a November 25, 2008, response, the applicant sent the input and output files. The staff reviewed the RESRAD-OFFSITE input file and noted that the dry bulk density of the saturated zone material is 1.51 g/cm^3 (94.3 pcf). The applicant stated that this value should be 1.59 g/cm^3 (99.3 pcf). The RESRAD-OFFSITE model requires the user to specify the initial radionuclide concentration on a picocurie per gram of soil basis. Instead of providing the appropriate soil-based concentrations, the applicant used the liquid-based concentrations that are identified in AP1000 DCD, Table 11.1-2. Since RESRAD-OFFSITE assumes the input concentrations are soil based, RESRAD-OFFSITE multiplies the input concentrations by a factor equal to the bulk density divided by the porosity to yield the initial groundwater concentration. The net effect is that the RESRAD-OFFSITE groundwater concentrations are much higher (by the factor bulk density divided by porosity) than the liquid concentrations specified in the AP1000 DCD. The staff notes that the applicant updated the RESRAD-OFFSITE analysis to use the correct porosity and appropriate method for initializing the contaminant concentration. The staff concluded that the applicant's November 25, 2008, response to RAI 828, Question 02.04.13-13, was acceptable and considers the question resolved.

In RAI 828, Question 02.04.13-15, the staff requested that the applicant explain how contaminant concentrations were affected by mixing in the Broad River as part of the RESRAD-OFFSITE analyses. In a November 25, 2008, response, the applicant stated that it used a mixing volume of 150,000 m^3 (5,297,200 ft^3) in the Broad River. That volume is the default volume used by RESRAD-OFFSITE to represent a lake. The volume represents 17.5 percent of the static pool that would reside behind the Ninety-Nine Islands Dam if there were no flow. The volume does not take into account the daily flow of water past the site. That flow is on the order of 6.2×10^6 m^3/day (2.2×10^8 ft^3/day), which, on a daily basis, is equivalent to 41 mixing volumes. The applicant stated that a residence time of 1 year was chosen to standardize the exposure duration to the accumulation duration. Actual residence time would be less than 1 day. The applicant stated that groundwater concentrations were not evaluated at the river edge because no regulatory basis exists for an intermediate release point. The applicant described the rationale for the mixing volume in the Broad River in WLS COL FSAR Section 2.4.13.4, "Conceptual Model." The staff concluded that the applicant's November 25, 2008, response to RAI 828, Question 02.04.13-15, was acceptable and considers the question resolved.

To explain the period used for evaluating radionuclide concentrations, the staff issued RAI 828, Question 02.04.13-17. In a November 25, 2008, response, the applicant expanded the evaluation period from 50 to 1,000 years. The applicant stated that this time period was sufficient for all radionuclides to either appear in the receptor body or disappear through radioactive decay. The staff notes that the applicant updated WLS COL FSAR Section 2.4.13.4, "Conceptual Model," to reflect the change in the length of the evaluation period. The staff concluded that the applicant's November 25, 2008, response to RAI 828, Question 02.04.13-17, was acceptable and considers the question resolved.

To clarify the initial dimensions assumed for the contaminant plume, the staff issued RAI 73, Question 02.04.13-20. In a November 12, 2009, response, the applicant stated that it would use the effective porosity to define the plume size and would update the WLS COL FSAR accordingly. The applicant approximated the plume as a cube with dimensions of 10.2 m (33.5 ft) on each side. The staff confirmed this dimension was consistent with the effective porosity. The staff notes that the applicant updated WLS COL FSAR Section 2.4.13.4, "Conceptual Model," and WLS COL FSAR Table 2.4.13-203 to reflect the change in the plume size. The staff concluded that the applicant's November 12, 2009, response to RAI 73, Question 02.04.13-20, was acceptable and considers the question resolved.

The staff agreed that travel times decrease as the effective porosity is reduced. However, the staff noted that as the total porosity is decreased, the retardation factor is increased, which means that travel times are increased. Bulk density (p_b), water density (p_w), total porosity (n), and soil distribution coefficient (K_d) affect the retardation of contaminants in the following manner.

$$\text{Retardation (R)} = 1+((p_b/p_w)/n)K_d$$

Therefore, it is imperative that appropriate values of total and effective porosity be used for any contaminant that has the potential to be absorbed. Therefore, in RAI 73, Question 02.04.13-21, the staff requested that the applicant justify why a total porosity value of 0.08 is conservative for contaminants that sorb to the sediments. In a November 12, 2009, response, the applicant identified appropriate site-specific values for total porosity and effective porosity for PWR of 27 percent and 8 percent, respectively. The applicant used these values in revised RESRAD-OFFSITE transport analyses. Additional discussion is included below in the applicant's response to RAI 73, Question 02.04.13-29. The staff concluded that the applicant's November 12, 2009, response to RAI 73, Question 02.04.13-21, was acceptable and considers the question resolved.

The staff noted that the applicant's proposed revision to WLS COL FSAR Section 2.4.13 described the annual precipitation used in the RESRAD-OFFSITE analysis as being the highest annual precipitation rate. Since average annual precipitation is used, the description is incorrect in both the proposed text and in WLS COL FSAR Table 2.4.13-203 and needs to be corrected in both locations. The staff also noted that the recharge rate of 19.5 cm/yr (7.68 in/yr) is lower than expected. In WLS COL FSAR Section 2.4.1.2.4, the applicant stated that, because surface materials in many areas are relatively impermeable, recharge is only 25 to 38 cm/yr (10 to 15 in/yr). In Environmental Report Section 2.3.1.1.5, the applicant stated that the recharge rate ranges from 22 to 33 percent of annual precipitation, which translates to 27.2 to 40.6 cm/yr (10.7 to 16.0 in./yr). Since recharge is used in RESRAD-OFFSITE to leach contaminants from the contaminated zone into the groundwater, the staff issued RAI 73, Question 02.04.13-24.

In a November 12, 2009, response to RAI 73, Question 02.04.13-24, the applicant stated that the value of 19.5 cm/yr (7.68 in/yr) is conservative. However, the applicant provided a more conservative value for the RESRAD-OFFSITE analyses. The input value for annual precipitation was updated to 1.27 m (i.e., approximately 50 in. as stated in WLS COL FSAR Section 2.3.1.1) and the evapotranspiration coefficient was revised to a value of 0.64, based on regional information. The runoff coefficient was assigned a value of zero to maximize the recharge rate. These three changes increased the effective recharge rate for the RESRAD-OFFSITE analyses to approximately 46 cm/yr (18 in./yr). The staff notes that the applicant updated WLS COL FSAR, Table 2.4.13-203, to reflect the changes. The staff concluded that the applicant's November 12, 2009, response to RAI 73, Question 02.04.13-24, was acceptable and considers the question resolved.

To understand the rationale for using the default mixing volume in RESRAD-OFFSITE, the staff issued RAI 73, Question 02.04.13-28. In a November 12, 2009, response, the applicant defined the mixing volume as 856,036 m^3 (30,230,600 ft^3), which it determined to be the volume of the Broad River reservoir from the postulated release point downstream to the Ninety-Nine Islands Dam. The applicant determined the residence time of radionuclides in the reservoir to be 0.00397 years (1.5 days) by assuming only 50 percent of the FERC license requirement for minimum flow (13.7 m^3/s [483 cfs]) passing the Ninety-Nine Islands Dam. The staff notes that the applicant updated WLS COL FSAR Table 2.4.13-203 to reflect the changes. The staff concluded that the applicant's November 12, 2009, response to RAI 73, Question 02.04.13-28, was acceptable and considers the question resolved.

In RAI 73, Question 02.04.13-29, the staff requested that the COL applicant clarify the choice of parameters used to initialize the RESRAD-OFFSITE analyses and explain how leaked fluid concentrations were transformed into initial radionuclide concentration on a picocurie per gram of soil basis. In a November 12, 2009, response, the applicant provided a revised RESRAD-OFFSITE analysis. The contaminated zone was assumed by the applicant to be a cube composed of PWR having an effective porosity of 0.08 and a dry bulk density of 1.98 g/cm^3 (124 pcf). The initial source-term concentrations were converted to a soil mass basis by using the dry bulk density; this method yielded the final concentration input to RESRAD-OFFSITE in picocuries per gram. Although the WLS site geology is variable, all five pathways (subsequently reduced to the current four pathways) were conservatively evaluated by the applicant by assuming the pathway was composed entirely of PWR, which has the highest conductivity and yields the shortest travel time.

The limiting pathway was from Unit 2 to Hold-Up Pond A. Contaminant transport to Hold-Up Pond A was via groundwater. Once at the Hold-Up Pond A, the applicant assumed all groundwater entered the surface water in the pond, travel over the spillway, and enter the Broad River. Dilution in the Broad River was described previously in this report in the response to RAI 73, Question 02.04.13-28. The applicant reported that the radionuclide concentrations were below the limits in 10 CFR Part 20, Appendix B, Table 2, Column 2. In addition, the total sum of fractions was also less than 1.0 for both the hypothetical well (approximately 0.8) and the surface-water analyses (approximately 4 x 10^{-5}). The nearest potable water supply is located approximately 34 km (21 mi) downstream of the outfall from Hold-Up Pond A. The staff expects that the Broad River concentrations and sum of fractions would be even lower at that location. The staff notes that the applicant updated WLS COL FSAR Section 2.4.13, Table 2.4.13-203, and WLS COL FSAR Table 2.4.13-204 to reflect the changes. The staff concluded that the applicant's November 12, 2009, response to RAI 73, Question 02.04.13-29, was acceptable and considers the question resolved.

Sensitive Parameters

Information Submitted by the Applicant

The applicant did not provide any sensitivity analyses in WLS COL FSAR Revision 0. Partly because of the RAI process, the applicant included sensitivity analyses in WLS COL FSAR Revision 1 and all subsequent revisions. The applicant examined the RESRAD-OFFSITE sensitivities for selected parameters, including total porosity, effective porosity, hydraulic conductivity, hydraulic gradient, and K_d values for those radionuclides that were evaluated for site soils. The applicant stated that no variation in any single parameter had sufficient impact to cause the concentrations to exceed 10 CFR Part 20, Appendix B, Table 2, Column 2 limits or for the sum of fractions calculation to exceed 1.0.

In response to one staff RAI, the applicant provided additional clarification related to the nature of the K_d values used for the PWR.

NRC Staff's Technical Evaluation

The staff examined the sensitivity cases and determined they covered a sufficient range of parameter variation to conclude that reducing parameter uncertainty would not alter the results.

With respect to the sensitivity analysis of the K_d values, the lower values affected the results for two radionuclides, namely, I-129 and Tc-99. Predicted concentrations were 10 and 5.8 percent higher, respectively. The applicant stated that, in both cases, the higher concentrations were still well below limits listed in 10 CFR Part 20, Appendix B, Table 2, Column 2.

The staff determined that the K_d value for PWR had the potential to be less than that for the soil/saprolite zone, which was the only material tested. Therefore, the staff issued RAI 73, Question 02.04.13-25. In a November 12, 2009, response, the applicant stated that the PWR material is a transition material between the saprolite above and the underlying bedrock and that the measurements on samples from the soil/saprolite are representative and appropriate for the accidental release analysis. The staff recognizes the difficulty of measuring K_d values directly on PWR and agrees that, as an alternative, using the lowest K_d values of all three samples and the results of the sensitivity tests (i.e., no impact) provides sufficient information to conclude that the WLS site meets the 10 CFR Part 20 concentration and sum of fractions requirements. The staff concluded that the applicant's November 23, 2009, response to RAI 70, Question 02.04.13-25, was acceptable and considers the question resolved.

Regulatory Compliance

Information Submitted by the Applicant

The applicant stated that meeting the concentration limits of 10 CFR Part 20 Appendix B, Table 2, Column 2 results in a dose of less than 0.05 Roentgen equivalent man (REM) and, therefore, demonstrates that the requirements of 10 CFR 20.1301 and 10 CFR 20.1302 are met.

NRC Staff's Technical Evaluation

The staff evaluated and confirmed the applicant's demonstration that radionuclide concentrations and sum of fractions in the Broad River would be well below the requirements of 10 CFR Part 20.

2.4.13.5 *Post Combined License Activities*

There are no post COL activities related to this section.

2.4.13.6 *Conclusion*

The staff reviewed the application and confirmed that the applicant addressed the relevant information and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. As set forth above, the applicant presented and substantiated information to establish the potential effects of accidental releases from the liquid waste management system. The staff reviewed the information provided and, for the reasons given above, concludes that the applicant has provided sufficient details about the site description, and about the design of the liquid waste management system, to allow the staff to evaluate, as documented in this section, whether the applicant has met the relevant requirements of 10 CFR 52.79(a)(1) and 10 CFR Part 100 with respect to determining the acceptability of the site, and with respect to 10 CFR 20 as it relates to ECLs. This addresses, and closes, COL information item 2.4-5. The staff concluded that the applicant provided sufficient information to satisfy the applicable requirements of 10 CFR Part 20, 10 CFR Part 52, and 10 CFR Part 100.

2.4.14 Technical Specifications and Emergency Operation Requirements

2.4.14.1 *Introduction*

WLS COL FSAR Section 2.4.14 describes technical specifications and emergency operation requirements as necessary. The requirements described implement protection against floods for safety-related facilities to ensure that an adequate supply of water for shutdown and cool-down purposes is available.

Section 2.4.14 of this report presents the staff's evaluation of required technical specifications and emergency operations for the proposed plant site.

2.4.14.2 *Summary of Application*

This section of the WLS COL FSAR addresses technical specifications and emergency operation requirements. The applicant addressed information related to technical specifications and emergency operation requirements as follows:

AP1000 COL Information Item

- WLS COL 2.4-6

In addition, this section addresses the following COL-specific information identified in AP1000 Tier 2, Section 2.4.1.6, Revision 19.

> Combined License applicants referencing the AP1000 certified design will address any flood protection emergency procedures required to meet the site parameter for flood level.

2.4.14.3 *Regulatory Basis*

The relevant requirements of NRC regulations for consideration of emergency protective measures, and the associated acceptance criteria, are described in NREG-0800, Section 2.4.14.

The applicable regulatory requirements for technical specifications and emergency operation requirements are set forth in the following:

- 10 CFR 50.36, "Technical Specifications," as it relates to identifying technical specifications related to all emergency procedures required to ensure adequate plant safety from controlling hydrological events by the organization responsible for the review of issues related to technical specifications.

- 10 CFR 52.79(a)(1)(iii), as it relates to identifying hydrologic site characteristics with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

- 10 CFR Part 100, as it relates to identifying and evaluating hydrological features of the site. The requirement to consider physical site characteristics in site evaluations is specified in 10 CFR 100.20(c).

- 10 CFR 100.23(d) sets forth the criteria to determine the siting factors for plant design bases with respect to seismically induced floods and water waves at the site.

2.4.14.4 *Technical Evaluation*

Information Submitted by the Applicant

The applicant stated that the maximum flood water-surface elevation at the WLS site is 178.3 m (584.8 ft) above MSL, resulting from a PMF event in the watershed of Make-Up Pond B. The applicant noted that the safety-related plant grade at the WLS site is 179.8 m (590 ft) above MSL. Therefore, the applicant concluded that a freeboard of over 1.5 m (5 ft) is available under the worst flooding scenario. The applicant stated that based on the description provided in WLS COL FSAR Section 2.4.12.5, the maximum expected groundwater elevation would not exceed the design criteria for the chosen reactor design. The applicant stated that no safety-related facilities would be affected by low flow or drought conditions in the Broad River. The applicant concluded that, based on site-specific conditions at the WLS site, no emergency protective measures designed to minimize the impact of hydrologic events on safety-related facilities are needed.

As described in Section 2.4.2.4 of this report, the applicant revised the analysis for site flooding under local intense precipitation in response to the staff's RAI 484, Question 10.04.05-2. The applicant's revised analysis established a slightly higher maximum water-surface elevation of 179.72 m (589.62 ft) above MSL caused by the effects of the local intense precipitation.

NRC Staff's Technical Evaluation

The staff reviewed COL Information Item 2.4-6 related to the provision of site-specific information about flood protection emergency operation procedures at the plant site included under WLS COL FSAR Section 2.4.

As stated in Section 2.4.10 of this report, the staff concluded that the site characteristic maximum flood water-surface elevation near the WLS site from several flooding mechanisms described in earlier sections herein remains below the site grade and meets the AP1000 DCD site parameter. As stated in Section 2.4.11 of this report, the staff concluded that there are no safety-related systems that can be affected by low water at the WLS site. Therefore, the staff concluded that no emergency operation requirements for flooding and water availability are needed for the WLS site.

2.4.14.5 *Post Combined License Activities*

There are no post COL activities related to this section

2.4.14.6 *Conclusion*

The staff reviewed the application and confirmed that the applicant has addressed the information relevant to technical specification and emergency operations requirements, and there is no outstanding information required to be addressed in the WLS COL FSAR related to this section.

As set forth above, the applicant has presented and substantiated information to establish the site description. The staff reviewed the information provided and, for the reasons given above, concludes that the applicant has provided sufficient details about the site description to allow the staff to evaluate, as documented in Section 2.4.14 of this report, whether the applicant has met the relevant requirements of 10 CFR 52.79(a)(1) and 10 CFR Part 100 with respect to determining the acceptability of the site. The WLS COL FSAR addressed flood protection emergency operation procedures in COL Information Item 2.4-6. The staff concluded that the applicant provided sufficient information to satisfy the applicable requirements of 10 CFR Part 52 and 10 CFR Part 100.

2.4.15 Combined License Information

The applicant also identified the following AP1000 DCD Tier 2 departure:

> STD DEP 1.1-1 within FSAR Section 2.4.1, “Hydrologic Description,” identifies instances where the FSAR sections are renumbered to include content consistent with Regulatory Guide 1.206, as well as NUREG-0800. Here, Subsections in Section 2.4.15 of the DCD are renumbered as Section 2.4.1, 2.4.2, 2.4.3, 2.4,4, 2.4.5, 2.4.6, 2.4.7, 2.4.8, 2.4.9, 2.4.10, 2.4.11, 2.4.12, 2.4.13, and 2.4.14.

The applicant’s evaluation, in accordance with 10 CFR Part 52, Appendix D, Section VIII.B.5, determined that this departure did not require prior NRC approval. The staff concluded that it is reasonable that the departure does not require prior NRC approval. The applicant’s process for evaluating departures and other changes to the AP1000 DCD are subject to NRC inspections. The NRC evaluated the AP1000 DCD Tier 2 departure in the aforementioned renumbered sections.

2.5 Geology, Seismology, and Geotechnical Engineering

In William States Lee III Nuclear Station (WLS) Combined License (COL) Final Safety Analysis Report (FSAR) Section 2.5, “Geology, Seismology, and Geotechnical Engineering,” Revision 9, the applicant described the geologic, seismic, and geotechnical engineering characteristics of the proposed COL site. Following NRC guidance in RG 1.206, “Combined License Applications for Nuclear Power Plants - LWR Edition,” and Regulatory Guide (RG) 1.208, “A Performance-Based Approach to Define Site-Specific Earthquake Ground Motion,” the applicant defined the following four zones around WLS and conducted technical investigations in the zones that became progressively more detailed passing from site region to site location.

- Site region – Area within a 320-km (200-mi) radius of the site location
- Site vicinity – Area within a 40-km (25-mi) radius of the site location
- Site area – Area within an 8-km (5-mi) radius of the site location
- Site location – Area within a 1-km (0.6-mi) radius of proposed Units 1 and 2

The applicant referred to the 1974 Preliminary Safety Analysis Report (PSAR) for the former Cherokee Nuclear Station (CNS) to provide information deemed pertinent for understanding geologic setting and characteristics of the WLS site. The information about the CNS site is pertinent for the WLS site because site locations coincide. However, most material in Final Safety Analysis Report (FSAR) Section 2.5 draws on information developed from sources published since the 1974 CNS PSAR, as well as data derived from investigations specifically performed for geologic, seismic, and geotechnical characterization of WLS.

The staff reviewed WLS COL FSAR Section 2.5, interacted with the applicant during public meetings, and issued requests for additional information (RAIs) to confirm the assertions made by the applicant in the WLS COL FSAR. In early versions of the WLS COL FSAR, the applicant used the Electric Power Research Institute (EPRI) base seismic source models for seismic hazard analysis. The applicant replaced those models with the new seismic source characterization (SSC) model for the central and eastern United States (CEUS) published in NUREG-2115, "Central and Eastern United States Seismic Source Characterization for Nuclear Facilities."

Following the 2011 Fukushima Dai-ichi nuclear power plant accident in Japan, which occurred as a result of the Great Tohoku earthquake and the subsequent tsunami, the NRC Near-Term Task Force (NTTF) issued a series of recommendations for reevaluating and improving nuclear power plant safety in the United States (U.S.). Consequently, on March 12, 2012, the NRC issued an information letter requesting that licensees of all operating nuclear power plants in the U.S. reevaluate the seismic hazard at their respective plants using the most recent data and evaluation methodologies available. That information letter also requested that licensees of operating plants in the CEUS use the seismic source model provided in NUREG-2115 to characterize seismic hazard at their respective plants. Consistent with existing guidance in Regulatory Guide (RG) 1.208, "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion" pertaining to the need to consider the latest information in the evaluation of seismic hazard, the NRC also requested that all COL and Early Site Permit (ESP) applicants in the CEUS address seismic hazard for their respective proposed plant sites using information in NUREG-2115 and modify the ground motion response spectra (GMRS), if needed. The staff issued this request to WLS in RAI 105, Question 01.05-1.

In a January 30, 2014, response to RAI 105, Question 01.05-1, the applicant stated that it replaced the previous EPRI seismic source models with the CEUS SSC model presented in NUREG-2115 as the starting point for developing GMRS for the WLS site. With this change in the base seismic source model, some RAIs the staff had previously issued became moot as specified in Section 2.5.2.4 of this report. Accordingly, the following sections of this report describe only the most recent version of the WLS COL FSAR, as well as additional WLS COL FSAR markups provided by the applicant in the January 30, 2014, response to RAI 105, Question 01.05-1. The technical evaluations undertaken by the staff do not discuss the obsolete portions of the WLS COL FSAR that were replaced by the CEUS SSC model. The RAIs discussed below in detail are the RAIs that remained applicable to the staff's review following the change in the base seismic source model, along with the new RAIs developed by the staff related to the most recent version of the WLS COL FSAR.

This section of the report is divided into five main parts, Sections 2.5.1 through 2.5.5, which parallel the five main WLS COL FSAR sections prepared by the applicant for the WLS COL application. The five sections are Section 2.5.1, "Basic Geologic and Seismic Information"; Section 2.5.2, "Vibratory Ground Motion"; Section 2.5.3, "Surface Faulting"; Section 2.5.4, "Stability of Subsurface Materials and Foundations"; and Section 2.5.5, "Stability of Slopes." These sections summarize the content of the WLS COL FSAR, as well as present the staff's evaluations, conclusions, and findings with regard to the geologic, seismic, and geotechnical engineering characteristics of WLS Units 1 and 2.

2.5.1 Basic Geologic and Seismic Information

2.5.1.1 *Introduction*

In WLS COL FSAR Section 2.5.1, the application describes basic geologic and seismic information collected by the applicant during site characterization investigations. This information addresses both regional and site-specific geology and seismicity. The investigations included surface and subsurface field studies, performed at progressively greater levels of detail closer to the site, within each of four circumscribed areas corresponding to site region, site vicinity, site area, and site location, as previously defined. The applicant conducted these investigations to assess the geologic and seismic suitability of WLS to determine whether new geologic or seismic data exist that could significantly affect seismic design based on the results of the probabilistic seismic hazard analysis (PSHA) for the site, and to provide the geologic and seismic data appropriate for plant design.

2.5.1.2 *Summary of Application*

WLS COL FSAR Section 2.5 incorporates by reference AP1000 DCD, Revision 19. Section 2.5.1. In addition, in WLS COL FSAR Section 2.5.1, the applicant provided site-specific supplemental information to address the following:

AP1000 COL Information Item

- WLS COL 2.5-1

The applicant provided additional information in WLS COL 2.5-1 to address COL Information Item 2.5-1 (COL Action Item 2.5.1-1). WLS COL 2.5-1 addresses the provision of regional and site-specific geologic, seismic, and geophysical information, as well as conditions caused by human activity. This information specifically includes the following topics: structural geology; seismicity; geologic history; evidence of paleoseismicity; site stratigraphy and lithology; engineering significance of geologic features; site groundwater conditions; dynamic behavior during prior earthquakes; zones of alteration, irregular weathering, or structural weakness; unrelieved residual stresses in bedrock; materials that could be unstable because of mineralogy or physical properties; and the effects of human activities in the site area.

Two main sections comprise WLS COL FSAR Section 2.5.1. WLS COL FSAR Section 2.5.1.1, “Regional Geology,” discusses physiography, geomorphology, and stratigraphy; tectonic setting, including tectonic structures that are possibly Quaternary in age (i.e., 2.6 million years ago, or 2.6 Ma, to present); and seismicity and paleoseismicity within the site region. WLS COL FSAR Section 2.5.1.1 also describes specific seismic sources inside the site region (i.e., the Charleston, SC area, the Giles County Seismic Zone of southwestern Virginia (GCVSZ), and the Eastern Tennessee Seismic Zone (ETSZ)), as well as significant seismic sources outside the site region, including the New Madrid and Central Virginia Seismic Zones. WLS COL FSAR Section 2.5.1.2, “Site Geology,” addresses physiography and geomorphology, geologic setting and history, stratigraphy and lithology, and structural geology within the site vicinity and site area. WLS COL FSAR Section 2.5.1.2 also discusses physiography and geomorphology, geologic setting and history, stratigraphy and lithology, structural geology, and geologic mapping at the site location, as well as engineering geology (including the effects of human activities), seismicity and paleoseismicity, and groundwater conditions in the site area.

The applicant developed WLS COL FSAR Section 2.5.1 for WLS based on information derived from geologic maps and reports published by State and Federal agencies and research scientists; aerial photographs and digital elevation models; communications with experts in geology, seismology, and tectonics of the site region, site vicinity, site area, and site location; and geologic field investigations performed specifically for characterization of the WLS vicinity, site area, and site location. These field investigations included geologic field reconnaissance; lineament studies; analysis of test pits, trenches, and boreholes located at WLS; and geologic mapping of bedrock exposures in the excavations for former CNS Units 2 and 3. The applicant also used information presented in the PSAR for the former CNS site to supplement data derived from geologic and seismic investigations conducted specifically for the WLS application. That action was possible because the former CNS site and the proposed WLS site are coincident. As noted by the applicant in WLS COL FSAR Sections 2.5.4.5.3.2 and 2.5.4.7.4.1, sound concrete placed over previously mapped continuous bedrock during construction of CNS Unit 1 entirely underlies the coincident WLS Unit 1 nuclear island. The existing excavation for former CNS Unit 2, which lies between WLS Units 1 and 2, exposes foundation grade level bedrock. CNS Unit 3 coincides with WLS Unit 2, the excavation for which currently reaches top of sound rock and not foundation grade level bedrock. It should be noted that, to avoid an area of deeply weathered saprolitic (i.e., soft, typically clay-rich, thoroughly decomposed rock formed in place by chemical weathering that characteristically preserves geologic structures present in the unweathered parent rock) bedrock at the northwest corner of former CNS Unit 1, the applicant moved the WLS Unit 1 nuclear island footprint 20.1 meters (m) (66 feet (ft)) south and 15.2 m (50 ft) east of CNS Unit 1. The applicant also moved the WLS Unit 2 nuclear island footprint 20.1 m (66 ft) south to maintain the spacing between WLS Units 1 and 2. Due to the short distances involved, proposed WLS Units 1 and 2 remain coincident with former CNS Units 1 and 3, respectively. The applicant included boring logs for seven boreholes placed to characterize subsurface geology in the relocated nuclear island footprints in Attachment 6 of WLS COL FSAR Appendix 2AA (“Lee Nuclear Station Field Exploration Data”).

To confirm previous geologic mapping of foundation bedrock exposures in excavations at the CNS site, the applicant performed geologic mapping of foundation grade level bedrock surfaces exposed in a small portion of the CNS Unit 2 excavation and compared that geologic map to archived scanned records for the CNS site. In addition, to supplement that confirmatory mapping effort and document lithologies and geologic features (including faults, shear and breccia zones, and fractures) that occur in concrete-covered foundation grade level bedrock underlying CNS Unit 1, the applicant compiled a detailed geologic map of the foundation grade level bedrock surface in the CNS Unit 1 excavation from original field notes and geologic maps initially prepared in the 1970s to characterize site location geology in that excavation. As explained in WLS COL FSAR Section 2.5.1.2.5.5, the applicant digitized the original geologic maps and prepared a detailed map compilation report containing copies of the original geologic maps and the digitized map data, detailed descriptions of the quality assurance controls exercised to produce the digitized geologic map compilation, and discussions of lithologies and geologic structures that occur in the CNS and WLS Unit 1 excavation. The applicant documented the similarity of rock types and orientations and ages of tectonic structures in the excavations for former CNS Units 1, 2, and 3 and proposed WLS Units 1 and 2 in the map compilation report.

Based on the geologic and seismic investigations performed for WLS Units 1 and 2, the applicant concluded in WLS COL FSAR Section 2.5.1 that no geologic or seismic conditions exist at the site that would negatively affect construction or operation of safety-related structures. With regard to the assessment of evidence for capable tectonic sources in WLS COL FSAR Section 2.5.1, based on the definition in RG 1.208, in WLS COL FSAR Section 2.5.3 the applicant stated that a capable tectonic source is a tectonic structure that can generate both tectonic surface deformation (e.g., faulting or folding) and vibratory ground motion in the present seismotectonic setting. The following sections of this report summarize the basic geologic and seismic information provided by the applicant in WLS COL FSAR Section 2.5.1: Sections 2.5.1.2.1, "Regional Geology," and 2.5.1.2.2, "Site Geology."

2.5.1.2.1 Regional Geology

WLS COL FSAR Section 2.5.1.1 discusses the physiography, geomorphology, stratigraphy, tectonic setting, and seismicity and paleoseismicity of the site region, defined as the area that lies within a 320 km (200 mi) radius of WLS. Under the discussion of regional tectonic setting, the applicant specifically addressed potential regional tectonic structures of Quaternary (2.6 Ma to present) age. Under regional seismicity and paleoseismicity, the applicant also presented information on seismic sources located both inside and outside the site region. The following subsections of this report summarize the information provided by the applicant in WLS COL FSAR Section 2.5.1.1.

2.5.1.2.1.1 *Regional Physiography, Geomorphology, and Stratigraphy*

WLS COL FSAR Section 2.5.1.1.1 describes the physiography, geomorphology, and stratigraphy of the site region. The applicant stated that, from northwest to southeast, the site region encompasses parts of five physiographic provinces, including the Appalachian Plateau, Valley and Ridge, Blue Ridge, Piedmont, and Atlantic Coastal Plain provinces. Figure 2.5.1-1 of this report shows the location of WLS in relation to these five provinces. WLS lies in the Piedmont physiographic province.

2.5.1.2.1.1.1 Appalachian Plateau, Valley and Ridge, Blue Ridge, and Atlantic Coastal Plain Physiographic Provinces

The applicant reported that unmetamorphosed, slightly deformed sedimentary rocks of Permian (299 to 251 Ma) to Cambrian (542 to 488 Ma) age underlie the Appalachian Plateau province, which extends from New York State to Alabama; that folded and faulted sedimentary rocks of Paleozoic (542 to 251 Ma) age underlie the Valley and Ridge province, which extends from New York State through Pennsylvania, Maryland, and Virginia; and that the Blue Ridge province extends from Pennsylvania into northern Georgia and consists of intensely folded and faulted metamorphosed basement and cover rocks intruded by igneous bodies. The applicant stated that the Atlantic Coastal Plain province extends southwest from Massachusetts to south-central Georgia, where it merges with the Gulf section of the Coastal Plain province. This province exhibits a low, gently rolling topography and contains semi-consolidated sedimentary rocks of Cretaceous (145.5 to 65.5 Ma) age and younger.

2.5.1.2.1.1.2 Piedmont Physiographic Province

The applicant reported that the Piedmont physiographic province, which contains WLS, lies between the Blue Ridge and Atlantic Coastal Plain provinces. The Piedmont province extends from New York State into Alabama and contains two distinct lithotectonic elements, defined based on the occurrence of different rock types and field evidence for a different tectonic history in each element (i.e., the Piedmont Zone to the northwest and the Carolina Zone to the southeast). A series of faults referred to as the Central Piedmont Shear Zone (CPSZ), a Paleozoic (> 251 Ma) structure located about 8 km (5 mi) northwest of WLS (Figure 2.5.1-2 of this report), separate the Piedmont Zone and the Carolina Zone. WLS lies in the Carolina Zone of the Piedmont physiographic province, specifically in the Charlotte terrane of that zone, as shown in Figure 2.5.1-2 of this report. The applicant reported that late Proterozoic (1000 to 54 Ma) to early Paleozoic (> 488 Ma) plutonic rocks intrude a suite of metamorphosed igneous rocks and make up the dominant lithology in the Charlotte terrane, and that sediments of the Atlantic Coastal Plain unconformably overlie rocks of the Carolina Zone southeast of WLS.

2.5.1.2.1.1.3 Mesozoic Extensional Basin

The applicant also discussed early to middle Mesozoic (251 to 145.5 Ma) extensional basins in WLS COL FSAR Section 2.5.1.1.1. The applicant indicated that these basins, which occur in the site region in both the Piedmont and Atlantic Coastal Plain physiographic provinces but do not underlie the site (Figure 2.5.1-1 of this report), formed in response to continental rifting associated with formation of the existing Atlantic Ocean basin.

2.5.1.2.1.2 *Regional Tectonic Setting*

WLS COL FSAR Section 2.5.1.1.2 describes the regional tectonic setting of WLS, including regional geologic history, tectonic stress in the mid-continent region, site region and site vicinity gravity and magnetic data, and principal regional tectonic structures. The following subsections of this report provide a summary of the information presented by the applicant in WLS COL FSAR Section 2.5.1.1.2, including a discussion of principal regional tectonic structures that are possibly Quaternary (2.6 Ma to present) in age.

2.5.1.2.1.2.1 Regional Geologic History

In the description of geologic history in WLS COL FSAR Section 2.5.1.1.2.1, the applicant stated that WLS lies in the southern part of the Appalachian Mountains orogenic belt, which developed during Paleozoic time (542 to 251 Ma) as a result of four compressional orogenic episodes related to continental collisions and the associated opening and closing of the proto-Atlantic Ocean. The applicant reported that the middle Ordovician (472 to 461 Ma) Taconic orogeny, the second regional tectonic deformation event to affect the Appalachian orogenic belt during the Paleozoic, is the earliest tectonic event that deformed rocks in the WLS region. The applicant noted that the fourth and final orogenic event, the Alleghanian orogeny, which occurred during late Paleozoic time between 359 and 251 Ma, is the most significant collisional tectonic event of the Appalachian orogenic belt. The applicant stated that this collisional event closed the Paleozoic proto-Atlantic Ocean basin; created the fold and thrust fault belt of the Valley and Ridge physiographic province at the latitude of WLS; and transported the ancestral North American basement westward to form the western part of the Blue Ridge physiographic province. The applicant indicated that, at the latitude of WLS, ancestral North American basement rocks underlie the Valley and Ridge, Blue Ridge, and Inner Piedmont physiographic provinces at depths of less than 10 to 14 km (6 to 9 mi). The applicant also reported that a basal décollement (i.e., a large-displacement, shallow-dipping to subhorizontal, regional shear zone that truncates all rock units above it) developed along the top of the ancestral North American basement and formed the source of the Paleozoic thrust sheets in these three physiographic provinces.

The applicant noted further that the early to middle Mesozoic (251 to 145.5 Ma) rift basins of the modern continental margin record the start of extension and continental rifting leading to the formation of the current Atlantic Ocean basin. The applicant pointed out that Wheeler suggested many earthquakes that occur in the eastern part of the Piedmont physiographic province and beneath the Coastal Plain may be spatially associated with reactivated buried normal faults initially developed during Mesozoic rifting. However, the applicant indicated that no definitive correlation of seismicity with Mesozoic normal faults has been demonstrated to date. The applicant stated that, after continental extension and rifting during early to middle Mesozoic time, deposition of Cretaceous (145.5 to 65.5 Ma) and Cenozoic (65.5 Ma to present) sediments of the Coastal Plain occurred along a passive continental margin.

2.5.1.2.1.2.2 Tectonic Stress in the Mid-Continent

WLS COL FSAR Section 2.5.1.1.2.2 discusses tectonic stress in the mid-continent region. The applicant stated that data compiled for the CEUS SSC model (NUREG-2115) confirmed previous work indicating a prevailing east-northeast to northeast orientation of maximum horizontal compressive stress in the mid-continent. This orientation is consistent with the theoretical trend of compressive forces exerted on the North America plate by seafloor spreading at the mid-Atlantic ridge as proposed by Zoback.

2.5.1.2.1.2.3 Gravity and Magnetic Data

WLS COL FSAR Section 2.5.1.1.2.3 presents gravity and magnetic data for the site region and vicinity. In WLS COL FSAR Section 2.5.1.1.2.3.1, the applicant reviewed the regional gravity data, including reprocessed data in the CEUS SSC database (NUREG-2115), and concluded that long wavelength anomalies observed in the vicinity of WLS are also characteristic of large parts of the Appalachian orogenic belt. The applicant also concluded that short wavelength characteristics noted in the WLS vicinity are gravity highs and lows related to mafic and granitic igneous intrusive rock bodies, respectively. The applicant further concluded that the gravity data do not show any evidence of Cenozoic (65.5 Ma to present) tectonic activity or specific Cenozoic tectonic structures in the site region or site vicinity.

In WLS COL FSAR Section 2.5.1.1.2.3.2, the applicant reviewed the regional magnetic data, also including reprocessed data in the CEUS SSC database (NUREG-2115), and concluded that first-order magnetic anomalies mainly reflect regional terranes of the Paleozoic Appalachian orogen, which trend northeast-southwest. The applicant associated superimposed anomalies having wavelengths of 5 to 19 km (3 to 12 mi) with igneous intrusive rock bodies or ore deposits. The applicant further concluded that the magnetic data generally are not of sufficient resolution to identify discrete faults.

2.5.1.2.1.2.4 Principal Regional Tectonic Structures

WLS COL FSAR Section 2.5.1.1.2.4 discusses principal regional tectonic structures within 320 km (200 mi) of WLS. The discussion included geophysical anomalies and lineaments and Paleozoic, Mesozoic, Cenozoic, and Quaternary tectonic structures.

Regional Geophysical Anomalies and Lineaments

In WLS COL FSAR Section 2.5.1.1.2.4.1, the applicant discussed geophysical anomalies and lineaments found within the site region. From southeast to northwest, these features are the East Coast Magnetic Anomaly (ECMA); the southeastern limit of Iapetan (i.e., late Proterozoic, 1,000 to 542 Ma, to early Paleozoic, > 488 Ma, in age) normal faulting; the Clingman, Ocoee, and New York-Alabama (NYAL) lineaments; the Appalachian gravity gradient; the northwestern boundary of Iapetan normal faulting; the Appalachian thrust front; and the Grenville Front. The applicant documented an age of > 65.5 Ma for these anomalies and lineaments.

Regional Paleozoic Tectonic Structures

In WLS COL FSAR Section 2.5.1.1.2.4.2, the applicant discussed the following 13 Paleozoic (542 to 251 Ma) tectonic structures that occur in the WLS region, associating them with thrust faulting that occurred during compressional Appalachian orogenic events:

- Kings Mountain Shear Zone – 8 km (5 mi) north of the site
- Tinsley Bridge Fault – 8 km (5 mi) southwest of the site
- Cross Anchor Fault – 16 km (10 mi) west of the site
- Southwest extension of the Boogertown Shear Zone – 13 km (8 mi) east of the site
- Reedy River Thrust Fault – 29 km (18 mi) west-northwest of the site
- Gold Hill-Silver Hill Shear Zone – 48 km (30 mi) south of the site
- Middleton-Lowdensville Shear Zone – 64 km (40 mi) south of the site
- Modoc Shear Zone – 121 km (75 mi) south of the site
- Brevard Fault Zone – 89 km (55 mi) northwest of the site
- Chappells Shear Zone – 92 km (57 mi) south of the site
- Hyco Shear Zone – 225 km (139 mi) northeast of the site
- Brindle Creek Thrust Fault – 64 km (40 mi) north of the site

- Beaver Creek Shear Zone – 64 km (40 mi) south of the site

The applicant indicated that the majority of these regional tectonic structures dip eastward, shallowing in dip as they merge into the basal décollement. Based on interpretations by researchers who determined that most of the seismicity in eastern North America occurs in basement rocks below the décollement surface, the applicant stated that seismicity in the Southern Appalachian Mountains is likely unrelated to shallow thrust faults mapped at the surface. The applicant did not attribute any seismicity to Paleozoic faults in the site region, and indicated that published literature does not report any evidence for associated late Cenozoic (23 Ma to present), including Quaternary (2.6 Ma to present), deformation related to these faults. Therefore, the applicant concluded that none of the Paleozoic structures found in the site region are capable tectonic features.

Regional Mesozoic Tectonic Structures

In WLS COL FSAR Section 2.5.1.1.2.4.3, the applicant described the following six Mesozoic (251 to 65.5 Ma) tectonic features that occur in the WLS region:

- Wateree Creek Fault – 89 km (55 mi) south of the site
- Summers Branch Fault – 89 km (55 mi) south of the site
- Ridgeway Fault – 96.6 km (60 mi) southeast of the site
- Longtown Fault – 96.6 km (60 mi) southeast of the site
- Mulberry Creek Fault – 89 km (55 mi) southwest of the site
- Mesozoic Rift Basins – located in the site region

The applicant stated that no known correlation exists between seismicity and these Mesozoic structures, and that published literature does not indicate any late Cenozoic, including Quaternary (2.6 Ma to present), deformation associated with the structures. Therefore, the applicant concluded that none of the Mesozoic structures found in the site region are capable tectonic features. However, the applicant noted that the Extended Continental Crust-Atlantic Margin (ECC-AM) seismotectonic zone of the CEUS SSC model (NUREG-2115), one of the multiple zones included in the seismic hazard calculation for WLS, contains Mesozoic tectonic structures of the site region. The applicant discussed the ECC-AM seismotectonic zone in WLS COL FSAR Section 2.5.2.2.3.2.

Regional Quaternary Tectonic Structure

In WLS COL FSAR Section 2.5.1.1.2.4.5, based on the catalogue of known or suggested Quaternary (2.6 Ma to present) tectonic structures in the CEUS developed by Crone and Wheeler and Wheeler, the applicant identified 15 potential Quaternary tectonic features in the WLS region, including faults, liquefaction features, and seismic zones. Crone and Wheeler and Wheeler classified potential tectonic features in regard to strength of evidence for Quaternary faulting and related deformation features based on evaluation of published information, not on a direct field examination of the actual geologic features. Their four classification categories for tectonic features are as follows:

- Class A – Geologic evidence demonstrates the existence of a Quaternary fault of tectonic origin, whether inferred from liquefaction or other deformation features or exposed at the ground surface.
- Class B – Geologic evidence demonstrates the existence of a fault or suggests Quaternary deformation, but the fault may not cut deep enough into the crust to be a potential source of significant earthquakes or the available evidence is not strong enough to assign the feature to Class A, but is too strong to assign it to C.
- Class C – Geologic evidence is insufficient to demonstrate the existence of a tectonic fault or Quaternary deformation associated with the feature.
- Class D – Geologic evidence demonstrates that the feature is not a tectonic fault.

The 15 potential Quaternary tectonic structures identified and described by the applicant within the WLS region include the Class A Charleston area, Bluffton, and Georgetown liquefaction features; the Class B Pembroke faults; and the Class C Fall Lines of Weems, Belair fault zone, Pen Branch fault, Cape Fear arch, Hares Crossroads fault, Lindside fault zone, Stanleytown-Villa Heights faults, Cooke fault, East Coast fault system, and Giles County and Eastern Tennessee seismic zones. Figure 2.5.1-3 of this report shows the locations of these potential Quaternary features relative to WLS.

The applicant discussed Charleston, SC area tectonic features (i.e., proposed source faults, seismic zones, and seismically-induced liquefaction and paleoliqefaction features) in FSAR Section 2.5.1.1.3.2.1; the Cape Fear arch in WLS COL FSAR Section 2.5.1.1.2.4.4; the ETSZ in WLS COL FSAR Section 2.5.1.1.3.2.2; and the GCVSZ in WLS COL FSAR Section 2.5.1.1.3.2.3. Other sections of this report summarize the information provided by the applicant in these four WLS COL FSAR sections related to the specific regional Quaternary tectonic structures defined above. The applicant described the remaining seven potential Quaternary tectonic structures (i.e., the Fall Lines of Weems, the Belair and Lindside fault zones, and the Pen Branch, Hares Crossroads, Stanleytown-Villa Heights, and Pembroke faults) in WLS COL FSAR Section 2.5.1.1.2.4.5. The following paragraphs summarize the information provided by the applicant in WLS COL FSAR Section 2.5.1.1.2.4.5 related to these seven potential Quaternary faults.

<u>Fall Lines of Weems (Class C)</u>

The applicant described the Fall Lines of Weems as alignments of rapids or anomalously steep sections of rivers draining the Piedmont and Blue Ridge physiographic provinces of North Carolina and Virginia (Figure 2.5.1-3 of this report), and stated that the features lie as close as 8 km (5 mi) to WLS. Based on published literature, field reconnaissance, and a review of the staff's evaluation of these features for the North Anna ESP application as presented in NUREG-1835, "Safety Evaluation Report for an Early Site Permit (ESP) at the North Anna ESP Site," issued September 2005, the applicant concluded that the Fall Lines of Weems developed due to a contrast in resistance to erosion of adjacent rock types and are not tectonic in origin.

Belair Fault Zone (Class C)

The applicant stated that a series of northeast-striking, southeast-dipping, oblique-slip faults comprise the Belair fault zone, which lies about 201 km (125 mi) south of WLS (Figure 2.5.1-3 of this report), and indicated that information provided by Prowell and O'Connor constrains last movement on the Belair fault zone to between late Eocene (< 33.9 Ma) and 26,000 years before present. The applicant concluded that, although the available data do not preclude Quaternary deformation along the fault zone, the data do not unequivocally demonstrate Quaternary movement in the zone.

Pen Branch Fault (Class C)

The applicant reported that the Pen Branch fault, located about 241 km (150 mi) south of WLS (Figure 2.5.1-3 of this report), is a northeast-striking structure that bounds the northwestern side of the buried Mesozoic (251 to 65.5 Ma) Dunbarton Basin, but has no surface expression. The applicant stated that seismic reflection and borehole data collected at the Savannah River Site, as well as investigations performed for the Vogtle ESP application to assess the age of the youngest strata deformed by the Pen Branch fault, show no evidence of fault movement younger than Eocene (55.8 to 33.9 Ma). The applicant concluded that the Pen Branch fault is not a capable tectonic structure.

Hares Crossroads Fault (Class C)

The applicant reported that Prowell postulated an offset of Coastal Plain strata by the Hares Crossroads fault based on field observations in a single roadcut at a location approximately 322 km (200 mi) east-northeast of WLS (Figure 2.5.1-3 of this report). Because the postulated fault occurs at a single location, the applicant concluded that it is most likely a local feature produced by landsliding and not a tectonic structure.

Lindside Fault Zone (Class C)

The applicant stated that the Lindside fault zone, located about 274 km (170 mi) north of WLS (Figure 2.5.1-3 of this report), is a northeast-striking normal fault, which does not exhibit any evidence for Quaternary movement.

Stanleytown-Villa Heights Faults (Class C)

The applicant indicated that the postulated Stanleytown-Villa Heights faults, located approximately 241 km (150 mi) northeast of WLS (Figure 2.5.1-3 of this report), mark the juxtaposition of Quaternary (2.6 Ma to present) alluvium and rocks of Cambrian (542 to 488 Ma) age. The applicant stated that field evidence summarized by Crone and Wheeler indicates these features are short, down-dropped in a downhill slope direction, and not associated with any other faults. The applicant concluded that field evidence suggests the faults most likely resulted from landsliding and are not tectonic structures.

Pembroke Faults (Class B)

The applicant reported that the postulated Pembroke faults lie about 241 km (150 mi) north of WLS (Figure 2.5.1-3 of this report) in alluvial deposits of probable Quaternary age, but exhibit no geomorphic expression and are possibly the result of dissolution collapse rather than tectonism.

2.5.1.2.1.3 *Regional Seismicity and Paleoseismology*

WLS COL FSAR Section 2.5.1.1.3 discusses the seismicity and paleoseismicity of the CEUS related to seismic sources that lie within the WLS region, as well as selected seismogenic tectonic sources that occur outside the site region. The applicant emphasized the discussion of tectonic features in the area around Charleston, SC, including postulated faults and possible seismic zones, because a currently unknown tectonic structure in that area produced the 1886 Charleston earthquake, one of the largest historical earthquakes ever to occur in the CEUS. The following subsections of this report summarize the information presented by the applicant in WLS COL FSAR Section 2.5.1.1.3 related to seismicity and paleoseismicity in and beyond the WLS region.

2.5.1.2.1.3.1 Seismic Source Zones and Potential Source Faults

In WLS COL FSAR Sections 2.5.1.1.3.1 and 2.5.1.1.3.2, the applicant addressed the seismicity of the CEUS and seismic sources defined by regional seismicity, respectively. In WLS COL FSAR Section 2.5.1.1.3.2, the applicant identified and discussed five principal areas of concentrated seismicity within 320 km (200 mi) of WLS and ten postulated buried source faults for the 1886 Charleston, SC earthquake. The applicant described the ten potential source faults for the 1886 Charleston, SC earthquake and three of the five areas of concentrated seismicity in the WLS region, specifically the three areas within 80 km (50 mi) of Charleston (i.e., the Middleton Place-Summerville, Bowman, and Adams Run seismic zones), in WLS COL FSAR Section 2.5.1.1.3.2.1. WLS COL FSAR Sections 2.5.1.1.3.2.2 and 2.5.1.1.3.2.3 address the remaining two areas of concentrated seismicity in the site region, the ETSZ and the GCVSZ, respectively. In WLS COL FSAR Section 2.5.1.1.3.2.4, the applicant discussed two areas of concentrated seismicity outside the site region, specifically the New Madrid and Central Virginia seismic zones. Figure 2.5.1-4 of this report shows the locations of all five principal areas of concentrated seismicity within WLS region and the two seismic zones located outside the site region. Based on data presented in the CEUS SSC model (NUREG-2115), the applicant considered only Charleston and the New Madrid seismic zone (NMSZ) as sources of repeated (i.e., two or more) large magnitude (i.e., M > 6.5) earthquakes, or RLMEs, in the PSHA for WLS as described in WLS COL FSAR Section 2.5.2.2.4.

Charleston Tectonic Features

In WLS COL FSAR Section 2.5.1.1.3.2.1, the applicant reported that the 1886 Charleston earthquake generated a Modified Mercalli Intensity (MMI) X shaking in the epicentral area of the earthquake. This earthquake, with an expected moment magnitude (E[M]) of 6.9 (i.e., estimated given the uncertainty in earthquake magnitude), was the largest historical seismic event to occur in the eastern United States. The applicant stated that liquefaction and paleoliquefaction features discovered in coastal South Carolina provide evidence that the Charleston seismic source exhibits characteristics of an RLME source as discussed in the CEUS SSC model (NUREG-2115). The applicant acknowledged that there is no identifiable primary tectonic surface deformation feature associated with the 1886 earthquake, such that researchers infer an earthquake source based on the geology, geomorphology, and instrumental seismicity of the site region. The applicant noted that the CEUS SSC model (NUREG-2115) includes three alternative geometries for the Charleston seismic source, as discussed in WLS COL FSAR Section 2.5.2.2.4.1, since there is no known causative tectonic feature associated with the 1886 earthquake.

Charleston Area Seismic Zones and Potential Source Faults

In WLS COL FSAR Section 2.5.1.1.3.2.1, the applicant discussed three zones of increased seismicity identified in the Charleston area, specifically the Middleton Place-Summerville, Bowman, and Adams Run seismic zones, as well as seismically-induced liquefaction features that occur in the Charleston area. Figure 2.5.1-4 of this report shows the locations of these three seismic zones. Since a specific source fault has not been defined for the 1886 Charleston earthquake, the applicant discussed ten postulated buried source faults for this earthquake, with all the faults located in or near the meizoseismal area (i.e., the area of maximum observed earthquake-induced damage due to shaking) of the earthquake. These ten postulated source faults include the East Coast Fault System (ECFS); the Helena Banks fault zone, located offshore southeast of Charleston and outside the meizoseismal area of the 1886 earthquake; and the Adams Run, Ashley River, Charleston, Cooke, Sawmill Branch, Dorchester, Summerville, and Woodstock faults. Figure 2.5.1-5 of this report shows the locations of these postulated source faults and the seismically-induced liquefaction features found in the Charleston area.

Charleston Area Seismically-Induced Liquefaction Features

In WLS COL FSAR Section 2.5.1.1.3.2.1, the applicant also discussed spatial distribution and ages of seismically-induced liquefaction features that occur in the Charleston area and along the South Carolina coast to constrain possible locations and recurrence rates for large earthquakes related to a Charleston area seismic source. The applicant noted that the meizoseismal area for the 1886 Charleston area contains the heaviest concentration of liquefaction features (Figure 2.5.1-5 of this report), and that the presence of liquefaction features associated with the 1886 Charleston earthquake and paleoliquefaction features attributed to pre-1886 earthquakes demonstrate the occurrence of repeated large magnitude earthquakes (RLMEs) in the Charleston area.

Eastern Tennessee Seismic Zone

In WLS COL FSAR Section 2.5.1.1.3.2.2, the applicant described the ETSZ. The applicant noted that this zone, located about 241 km (150 mi) west-northwest of WLS (Figure 2.5.1-4 of this report), is one of the most active seismic zones in eastern North America in terms of small magnitude (i.e., **M** < 5) earthquakes, and that no damaging historical earthquakes have occurred in the zone. The applicant indicated that earthquakes in the ETSZ occur in Precambrian (> 542 Ma) crystalline basement rocks at a mean focal depth of about 14 km (9 mi), which is well below the 5 km (3 mi) depth of the Appalachian basal décollement separating Precambrian basement from younger rocks in the overlying thrust sheets. The applicant reported that earthquakes within the ETSZ cannot be related to any known capable tectonic structure, but that the western margin of the zone is coincident with the prominent magnetic gradient defined by the NYAL magnetic lineament. Although the CEUS SSC model (NUREG-2115) did not define the ETSZ as an RLME, the applicant reported that, based on field data, Hatcher and others have suggested one or more prehistoric earthquakes of M 6.5 may have occurred in the ETSZ within the last 73,000 to 200,000 years. The applicant explained how the CEUS SSC model (NUREG-2115) incorporates the ETSZ into the Paleozoic Extended Crust (PEZ) seismotectonic zone in WLS COL FSAR Section 2.5.2.2.3.5, and discussed geologic investigations conducted in the zone more extensively in WLS COL FSAR Section 2.5.2.2.5.1.

Giles County Seismic Zone

In WLS COL FSAR Section 2.5.1.1.3.2.3, the applicant discussed the Giles County seismic zone (GCVSZ), which lies in Virginia approximately 257 km (160 mi) north of WLS (Figure 2.5.1-4 of this report). The applicant stated that the largest earthquake to occur in the seismic zone was an **M** 5.9 event in 1897. The applicant noted that earthquakes in this zone occur in Precambrian crystalline basement rocks lying beneath the Appalachian thrust sheets at depths of 5 to 26 km (3 to 16 mi). The applicant stated that no capable tectonic structures have been identified in the GCVSZ. The applicant explained how the CEUS SSC model (NUREG-2115) incorporates this seismic zone into the PEZ seismotectonic zone in WLS COL FSAR Section 2.5.2.2.3.5.

Concentrated Seismicity Outside the Site Region

In WLS COL FSAR Section 2.5.1.1.3.2.4, the applicant described two areas of concentrated seismicity outside the WLS region, specifically the New Madrid and Central Virginia seismic zones. Figure 2.5.1-4 of this report shows the locations of these two seismic zones.

The NMSZ extends from southeastern Missouri to southwestern Tennessee and lies more than 724 km (450 mi) west of WLS. The applicant reported that three large-magnitude historical earthquakes occurred in this zone between December 1811 and February 1812, with magnitudes interpreted by area experts ranging from **M** 7.2 to **M** 8.1. The applicant noted that recent published paleoseismic data suggest a mean earthquake recurrence interval of 500 years for the NMSZ. However, the applicant also noted that paleoseismic studies suggest seismic activity in this zone during the Holocene (0.01 Ma to present) may not be indicative of the long-term recurrence rate based on information provided in the CEUS SSC model (NUREG-2115). The applicant discussed temporal clustering models and uncertainties associated with the paleoliquefaction record and recurrence and explained how the CEUS SSC model (NUREG-2115) incorporated this zone as an RLME in WLS COL FSAR Section 2.5.2.2.4.2.

The Central Virginia seismic zone (CVSZ) is an area of persistent, low-level seismicity located more than 402 km (250 mi) north-northeast of WLS. This zone extends about 120 km (75 mi) in a north-south direction and about 144 km (90 mi) in an east-west direction from Richmond, VA, to Lynchburg, VA. The applicant noted that no causative surface faults have been identified in the zone. The applicant reported that the largest historical earthquake in the CVSZ occurred near Mineral, VA, in August 2011 with a magnitude (M_w) of 5.8, and discussed the investigations conducted for the Mineral earthquake in more detail in WLS COL FSAR Section 2.5.2.2.5.2.

2.5.1.2.2 Site Geology

WLS COL FSAR Section 2.5.1.2 describes geologic characteristics of the WLS area and site location, defined as the areas lying within an 8 km (5 mi) and a 1 km (0.6 mi) radius of the site, respectively. These characteristics include physiography and geomorphology, geologic setting and history, stratigraphy and lithology, structural geology, engineering geology, seismicity and paleoseismicity, and groundwater conditions. For the discussion of seismicity and paleoseismicity, the applicant also included the area lying within a 40 km (25 mi) radius of the site (i.e., the site vicinity). The following subsections of this report summarize the information provided by the applicant in WLS COL FSAR Section 2.5.1.2.

2.5.1.2.2.1.1 Site Area Physiography and Geomorphology

In WLS COL FSAR Section 2.5.1.2.1, the applicant described the physiography and geomorphology of the WLS area. The applicant indicated that the site lies within the Piedmont physiographic province of central South Carolina on the western side of the Broad River, bounded by the Coastal Plain province to the southeast and the Blue Ridge province to the northwest (Figure 2.5.1-1 of this report). The Broad River is the primary drainage in the site area. The applicant reported that gently to moderate rolling hills and well-drained mature valleys characterize topography of the site area, with elevations ranging from about 122 m to 305 m (400 to 1,000 ft) above mean sea level (MSL).

The applicant stated that residual soils and saprolite overlie igneous and metamorphic bedrock in the site area. The applicant noted that prominent lineaments associated with topographic ridges in the site area reflect patterns of stream erosion controlled by erosion-resistant rock units, rather than by surface displacement along tectonic structures, and concluded that no topographic features in the site area suggest surface faulting related to tectonic deformation.

2.5.1.2.2.2 *Site Area Geologic Setting and History*

In WLS COL FSAR Section 2.5.1.2.2, the applicant described the geologic setting and history of the WLS area. The applicant stated that metamorphosed and complexly deformed plutonic and volcanic rocks and associated metasedimentary rock units underlie the site area, which lies in the western part of the Charlotte terrane of the Carolina Zone (Figure 2.5.1-2 of this report). The applicant indicated that rocks comprising the Charlotte terrane exhibit a long and complex geologic history from Neoproterozoic (1000 to 542 Ma) through Triassic (251 to 201.6 Ma), including magmatic activity related to the opening of the modern Atlantic Ocean basin. The applicant reported that Charlotte terrane rocks in the site area show the effects of early Cambrian (542 to 535 Ma) metamorphism and deformation; tectonic and thermal events during Silurian (444 to 416 Ma), Devonian (416 to 359 Ma) and Carboniferous-Permian (the Alleghanian orogeny at 359 to 251 Ma) time; and Mesozoic (specifically Triassic-Jurassic, 251 to 145.5 Ma) extension and magmatism.

2.5.1.2.2.3 *Site Area Stratigraphy and Lithology*

In WLS COL FSAR Section 2.5.1.2.3, the applicant discussed site area stratigraphy and lithology. The applicant stated that rock units in the WLS area generally belong to the Battleground Formation, defined by Horton as Neoproterozoic (1,000 to 542 Ma) in age, with younger cross-cutting Triassic and Jurassic (251 to 145.5 Ma) intrusive igneous dikes. The applicant described the Battleground Formation as a volcaniclastic sequence intruded by its own parent magma, made up of metamorphosed felsic to intermediate composition rocks, metavolcaniclastic sequences with metaigneous intrusions of similar compositions, and interlayered metasedimentary rock units. The applicant explained that few primary features remain in this lithologic sequence for determining stratigraphic relationships, due primarily to metamorphism and intense deformation of the rock units. The applicant described individual rock units of the Battleground Formation in WLS COL FSAR Section 2.5.1.2.3, and reported that one of the metamorphosed intrusive igneous bodies of that formation, rock unit "Zto" (Figure 2.5.1-6 of this report), is the foundation unit for WLS. The applicant also stated that Quaternary alluvium made up of gravel, sand, and silt deposits occurs in river and stream valleys in the site area.

2.5.1.2.2.4 *Site Area Structural Geology*

In WLS COL FSAR Section 2.5.1.2.4, the applicant discussed the structural geology of the site area. The applicant stated that geologic structures observed in the site area and at the site location include deformation features that developed during five regional deformational episodes, D1 through D5, and affected rock units in the site vicinity and site area and at the site location. Based on observed field relationships, the applicant indicated geologic structures that developed during deformational episodes D1 and D2 formed during a metamorphic event about 549 to 535 Ma ago. The applicant reported that radiometric dates constrain the age of the D3, D4 and D5 deformational episodes to be no younger than 296 Ma. The following sections of this report summarize the information presented by the applicant in WLS COL FSAR Sections 2.5.1.2.4 on geologic structures in the WLS area.

2.5.1.2.2.4.1 Structures in the Site Area

In WLS COL FSAR Section 2.5.1.2.4, the applicant discussed site area geologic structures, including the Cherokee Falls and Draytonville synforms (i.e., folds with limbs closing downward, applied when the stratigraphic order of the folded sequence is unknown) and minor surfaces exhibiting slickensides (i.e., surficial striations that typically occur on a fault plane, which indicate direction, but not amount, of displacement along the fault). Based on field relationships, the applicant interpreted the Cherokee Falls and Draytonville synforms, both located about 7 km (4.5 mi) northwest of WLS, as D2 deformation features. The applicant also reported that several small-scale minor faults described in the CNS PSAR, located 6.4 km (4 mi) north and 9.7 km (6 mi) northwest of WLS in rocks that are Neoproterozoic to Cambrian in age (> 535 Ma), could not be traced laterally and are typical of older, minor geologic structures commonly found in the Piedmont.

The applicant discussed minor slickensided surfaces that occur at Cherokee Falls, SC, 4.8 km (3 mi) northwest of the site, and Draytonville, SC, 6.4 km (4 mi) west of the site, as described in the CNS PSAR. The applicant stated that these surfaces exhibiting slickensides could not be traced beyond a single exposure, a field observation suggesting the slickensides do not mark a through-going fault. The applicant also stated that field work performed for the WLS application did not reveal these reported slickensided surfaces. Therefore, the applicant concluded that the slickensided surfaces represent minor, localized structures similar to those found throughout the

Piedmont, and that they do not provide any evidence for Quaternary (2.6 Ma to present) fault movement.

2.5.1.2.2.5 *Site Location Geology*

In WLS COL FSAR Section 2.4.1.2.5, the applicant discussed the geology of the WLS location, which is the area located within a 1-km (0.6-mi) radius of the site. The applicant specifically addressed physiography and geomorphology, geologic setting and history, stratigraphy and lithology, structure, and geologic mapping in relation to site location.

2.5.1.2.2.5.1 Site Location Physiography and Geomorphology

In WLS COL FSAR Section 2.5.1.2.5.1, the applicant described the physiography and geomorphology of the site location. The applicant reported that the physiography and geomorphology observed at the site location are typical of the site area as described in WLS COL FSAR Section 2.5.1.2.1. Elevations at the site location range from 155 m to 250 m (510 to 820 ft) above MSL, with the observed relief primarily the result of stream drainage incision. The applicant indicated that an erosion-resistant quartzite unit holds up McKown's Mountain, the north-northeast-trending linear ridge located northwest of WLS (Figure 2.5.1-6 of this report).

2.5.1.2.2.5.2 Site Location Geologic Setting and History

In WLS COL FSAR Section 2.5.1.2.5.2, the applicant discussed the geologic setting and history of the site location. The applicant indicated that site location setting and history are generally congruent with the site area as described in WLS COL FSAR Section 2.5.1.2.2. However, the applicant noted that, of the five regional deformation episodes (D1 through D5) documented in the site area, rock units at the site location most strongly record only two of the five events (i.e., D1 and D2). The applicant stated that variations in strength and anisotropy between the metaplutonic rock mass comprising foundation unit Zto and the surrounding metavolcanic and metasedimentary country rocks resulted in the development of different tectonic structures in the different rock units, making it difficult to distinguish structures produced by deformation events D1 and D2.

2.5.1.2.2.5.3 Site Location Stratigraphy and Lithology

In WLS COL FSAR Section 2.5.1.2.5.3, the applicant described the stratigraphy and lithology of the WLS location. The applicant indicated that rock unit Zto (Figure 2.5.1-6 of this report) is the foundation unit for WLS, and that saprolite, with a maximum thickness of 30 m (100 ft), is typically 12 to 24 m (40 to 80 ft) thick at the site location. The applicant noted that Quaternary alluvium composed of gravel, sand, and silt deposits occurs in river and stream valleys.

2.5.1.2.2.5.4 Site Location Structure

In WLS COL FSAR Section 2.5.1.2.5.4, the applicant discussed the geologic structures mapped at the site location. These structures include the McKown's Creek antiform (i.e., a fold with limbs closing upward, applied when stratigraphic order of the folded sequence is unknown), shear and breccia zones, dilation fractures, joints, and slickensides. The applicant indicated that, although foundation rock unit Zto is generally massive, the unit locally exhibits discrete geologic structures expressed as fractures, joints, and shear and breccia zones at the site location.

McKown's Creek Antiform

Based on Schaeffer, the applicant interpreted the McKown's Creek antiform as a D2 deformation structure that developed around 542 Ma ago. The applicant noted that WLS foundation unit Zto lies in the nose of this proposed fold. The applicant reported that the more recent geologic map of Nystrom does not include the McKown's Creek antiform, and that most of the folds reflected in geologic map patterns at the site location resulted from deformation D2.

Shear and Breccia Zones, Dilation Fractures, and Joints

The applicant stated that shear and breccia zones, which occur at the site location in foundation rock unit Zto, show preferential development in smaller mafic dikes and along the margins of larger mafic dikes that intrude the foundation unit. The applicant indicated that the best-developed shear and breccia zones, initially studied during investigations performed for the Cherokee site, strike a few degrees east of north and dip steeply to the southeast, while a secondary set strikes northwest and dips moderately to the southwest. The applicant reported that the shear and breccia zones exhibit an early ductile fabric overprinted by a later-stage brittle fabric, and that a potassium-argon (K-Ar) radiometric age date constrains timing of the shearing and brecciation to be older than 219 Ma. The applicant obtained the K-Ar age date on potassium feldspar, collected during site characterization of the Cherokee site from an undeformed igneous vein cutting across one shear zone.

The applicant indicated that igneous veins containing undeformed minerals, including potassium feldspar, in rock unit Zto cut across the shear and breccia zones as dilation fractures (i.e., fractures opened by expansion perpendicular to the walls of the fracture, rather than by shearing parallel to the fracture). The applicant reported that the K-Ar radiometric age date discussed in the above paragraph also constrains timing of development of the dilation fractures to be older than 219 Ma. The applicant reported that steeply-dipping (i.e., 60 degrees to vertical) joints are common at the site location and exhibit a range of strike directions.

Slickensides

The applicant reported the occurrence of slickensides on joint surfaces and along contacts between different rock types at the site location, and stated that the slickensides post-date the dilation fractures and shear and breccia zones since the slickensided surfaces cross-cut the dilation fractures and shear and breccia zones. The applicant noted the common occurrence of chlorite mica on the slickensided surfaces, and concluded that minor movement along the surfaces must have occurred more than 219 Ma ago in connection with the metamorphic event that produced the chlorite (i.e., a greenschist facies metamorphic event, interpreted to occur within a temperature range of 300-500 degrees centigrade). The applicant indicated that slickensides also occur on discontinuous surfaces in partially weathered rock and saprolite.

2.5.1.2.2.5.5 Site Location Geologic Mapping

In WLS COL FSAR Section 2.5.1.2.5.5, the applicant summarized geologic mapping efforts conducted for WLS to document geologic characteristics of foundation bedrock at the site location. To confirm the results of previous geologic mapping of bedrock exposures in excavations at the CNS site as part of the investigations for the WLS application, the applicant performed geologic mapping to define lithologies and major tectonic features in foundation grade level bedrock exposed in a portion of the CNS Unit 2 excavation and compared that geologic map to scanned records for the CNS site located in the Duke archives. The applicant reported agreement between this confirmatory map and the original CNS Unit 2 geologic map of foundation grade level bedrock. The applicant also mapped available exposures of the top of sound rock (i.e., not foundation grade level bedrock) in the CNS Unit 3 excavation to preliminarily assess lithologies and major tectonic features at WLS Unit 2, which is coincident with CNS Unit 3.

The applicant likewise documented lithologies and tectonic features (i.e., faults, shear and breccia zones, and fractures) that occur in foundation grade level bedrock underlying CNS Unit 1, which coincides with WLS Unit 1. The documentation materials included detailed geologic maps of the foundation grade level bedrock surface in the CNS Unit 1 excavation that now lies under concrete, produced by digitizing and compiling the original field maps prepared for CNS Unit 1, and a report containing the compiled digital maps and associated data. Based on these materials for CNS Unit 1, the confirmatory mapping of a portion of CNS Unit 2, and the preliminary mapping of exposures at CNS Unit 3, the applicant confirmed the similarity of rock types and orientations and ages of tectonic structures in the excavations for WLS/CNS Unit 1, CNS Unit 2, and WLS Unit 2/CNS Unit 3. Regarding the age of tectonic features, based on undeformed mineral assemblages associated with a prominent fault zone (initially designated as “Fault Zone 6”) mapped in the excavation for the CNS Unit 1 reactor and auxiliary buildings, the applicant concluded that the fault zone must have developed prior to 170 Ma (i.e., middle Mesozoic). Closure temperatures and age relationships determined by the applicant for undeformed minerals in other minor fault zones also generally indicated that tectonic deformation at WLS is older than early Mesozoic (i.e., > 201.6 Ma). Therefore, based on results of the geologic mapping conducted at the site location as described above in combination with information on mineral closure temperatures and age dates, the applicant concluded that foundation bedrock at WLS does not contain any tectonic structures of Quaternary age (i.e., potentially capable tectonic structures).

2.5.1.2.2.6 *Site Area Engineering Geology*

In WLS COL FSAR Section 2.5.1.2.6, the applicant discussed the engineering geology of the WLS area and site location. The applicant indicated that unweathered crystalline rock of the Battleground Formation underlies the site area and site location, and that the foundation unit is a metamorphosed igneous intrusive rock body (i.e., Unit Zto, shown in Figure 2.5.1-6 of this report). The applicant stated that average shear wave velocities greater than 2,804 m/sec (9,200 ft/sec) characterize unweathered rock materials underlying the site. The applicant also stated that no mining operations, excessive extraction or injection of groundwater, or impoundments of water occur within the site area that could detrimentally affect geologic conditions, and that bedrock at the site is not susceptible to settlement or subsidence due to groundwater withdrawal. The applicant cross-referenced WLS COL FSAR Section 2.5.4 for additional details related to the engineering geology of the site area and site location. In WLS COL FSAR Section 2.5.4.5.3.1, the applicant specifically indicated that geologic mapping of the final exposed foundation-bearing rock surface would be performed upon completion of subsequent excavations related to the nuclear islands.

2.5.1.2.2.7 *Site Area Seismicity and Paleoseismology*

In WLS COL FSAR Section 2.5.1.2.7, the applicant discussed site area and site vicinity seismicity and paleoseismology. Based on the updated seismicity catalog used for the CEUS SSC model (NUREG-2115), the applicant noted that the largest earthquake within 40 km (25 mi) of the site was an E[M] 4.13 event in 1886. The applicant stated that the January 1913 E[M] 4.54 earthquake in Union County, SC, which does not have a well-defined epicenter but occurred at the margin of the site vicinity about 40 km (25 mi) south-southwest of WLS, produced an estimated Rossi-Forel shaking intensity at the site of VI. The applicant indicated that no causative tectonic feature has been identified for the Union County earthquake. The estimated Modified Mercalli Intensity at the site for the 1886 Charleston earthquake was VI.

Concerning geologic features that may indicate paleoseismic events occurred in the site area, the applicant stated that no published data indicate the presence of paleoseismic features in the site area. Based on extensive studies of outcrops during field reconnaissance investigations for WLS, the applicant also stated that no evidence exists for post-Miocene (< 5.3 Ma) earthquake activity in the site area.

2.5.1.2.2.8 *Site Area Groundwater Conditions*

WLS COL FSAR Section 2.5.1.2.8 cross-references WLS COL FSAR Section 2.4.12 for the detailed discussion of groundwater conditions presented by the applicant for WLS.

2.5.1.3 *Regulatory Basis*

NUREG-1793, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design,” and its supplements address the regulatory basis for information incorporated by reference.

The applicable regulatory requirements for geologic and seismic information are as follows:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 52.79(a)(1)(iii), “Contents of Applications: Technical Information in Final Safety Analysis Report,” as it relates to including in the WLS COL FSAR information on seismic and geologic characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

- 10 CFR 100.23, “Geologic and Seismic Siting Criteria,” for evaluating the suitability of a proposed site based on the consideration of geologic, geotechnical, geophysical, and seismic characteristics of the proposed site. Geologic and seismic siting factors must include the safe shutdown earthquake ground motion (SSE) for the site and the potential for surface tectonic and non-tectonic deformation.

In addition, the acceptance criteria associated with the relevant requirements of NRC regulations for basic geologic and seismic information are given in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” Section 2.5.1, as follows:

- Regional Geology: Requirements of General Design Criterion (GDC) 2 of 10 CFR Part 50, Appendix A, 10 CFR 52.17 and 10 CFR 100.23(c) are met and guidance in RG 1.206, RG 1.208, and RG 4.7 followed for this area of review if a complete and documented discussion is presented for the geologic setting, tectonic framework and conditions caused by human activities, that have the potential to affect the safe siting and design of the plant. This section should contain a review of regional stratigraphy, lithology, structural geology, geologic and tectonic history, tectonic features (with emphasis on the Quaternary period), seismology, geomorphology, paleoseismology, and physiography within the 320-km (200-mi) site region or beyond as necessary to provide a framework within which significance to safety can be evaluated concerning geology, seismology, and conditions caused by human activities. Geologic maps and cross-sections constructed at scales adequate to illustrate relevant regional features should be included in the application.

- Site Geology: Requirements of GDC 2 in 10 CFR Part 50, Appendix A, 10 CFR 52.17 and 10 CFR 100.23 (c) are met and guidance in RG 1.206, RG 1.208, and RG 4.7 followed for this area of review if it contains a description and evaluation of geologic features, tectonic features, and conditions caused by human activities at appropriate levels of detail for determining any potential natural hazards that might affect the design and operation of the proposed facility. This subsection should contain the following information:

    a. Structural geology, including identification and characterization of faults, joints, and other tectonic deformation features and discussion of the relationships between these features and regional tectonic structures.

    b. Geologic maps and cross-sections constructed at scales adequate to clearly illustrate pertinent features in the site vicinity, area and location shall be included in the application.

    c. Stratigraphy and lithology of rock units and discussion of their relationships to the regional lithostratigraphic framework.

    d. Geomorphologic features as tectonic strain markers or indicators of other potentially hazardous natural phenomena (e.g., landslides, karst development and dissolution collapse, growth faults).

    e. Geologic and tectonic history, particularly for the Quaternary Period, and discussion of the relationship to regional geologic and tectonic history.

f. Tectonic framework description, including identification of historical and instrumentally-recorded earthquakes; identification and characterization of any local tectonic features as they might be related to seismicity; discussion of the relationships between local and regional tectonic structures and any relationship to seismicity; and the nature of the crust beneath the site.

g. Evidence for paleoseismic features, including a description of investigations performed by the applicant to verify the presence or absence of the features.

h. Geologic features that have significance for geotechnical engineering:

    1) Zones of mineralization, alteration, irregular or deep weathering, or structural weakness in surface or subsurface materials

    2) Surface and subsurface dissolution features in soluble rock such as limestone, gypsum, or salt

Geologic characteristics should also be consistent with the appropriate sections from RG 1.132, Revision 2, “Site Investigations for Foundations of Nuclear Power Plants”; RG 1.138, Revision 2, “Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants”; RG 1.198, “Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites”; RG 1.206; and RG 1.208.

2.5.1.4 *Technical Evaluation*

The staff reviewed WLS COL FSAR Section 2.5.1 and the referenced DCD to ensure that the combination of information presented in the WLS COL FSAR and the DCD completely represents the required information related to basic geologic and seismic characteristics. The staff confirmed, as described below, that information contained in the application or incorporated by reference addresses the information required for this review topic. NUREG-1793 and its supplements document the results of the staff’s evaluation of information incorporated by reference into the WLS application.

The staff reviewed the following information in the WLS COL FSAR:

AP1000 COL Information Item

- WLS COL 2.5-1

The staff reviewed WLS COL 2.5-1 regarding the geologic, seismic, and geophysical information included in WLS COL FSAR Section 2.5.1. The COL information item in AP1000 DCD Section 2.5.1 states:

> Combined License applicants referencing the AP1000 certified design will address the following regional and site-specific geological, seismological, and geophysical information as well as conditions caused by human activities: (1) structural geology of the site, (2) seismicity of the site, (3) geologic history, (4) evidence of paleoseismicity, (5) site stratigraphy and lithology, (6) engineering

significance of geologic features, (7) site groundwater conditions, (8) dynamic behavior during prior earthquakes, (9) zones of alteration, irregular weathering, or structural weakness, (10) unrelieved residual stresses in bedrock, (11) materials that could be unstable because of mineralogy or physical properties, and (12) effect of human activities in the area.

WLS COL FSAR Section 2.5.1 contains geologic and seismic information collected by the applicant in support of the vibratory ground motion analysis and the site-specific GMRS provided in WLS COL FSAR Section 2.5.2. RG 1.208 recommends that applicants update the geologic, seismic, and geophysical database and evaluate any new data to determine whether revisions to the existing seismic source models are necessary. Consequently, the staff focused on geologic and seismic data published since the middle to late 1980s (i.e., after the original studies conducted by EPRI), including information in the CEUS SSC model as presented in NUREG-2115, to assess whether these data required updates of existing seismic source models. Through its review of WLS COL FSAR Section 2.5.1 and the report presenting the compiled geologic map for WLS/CNS Unit 1 excavation, the staff determined whether the applicant had complied with the applicable regulations and conducted the investigations at an appropriate level of detail in accordance with RG 1.208.

As part of the technical evaluation of WLS COL FSAR Section 2.5.1, the staff visited WLS between April 27 and May 2, 2008, on January 27 and 28, 2009, on July 12 to 14, 2011, and on February 10, 2014, to meet with the applicant regarding the geologic, seismic, geophysical, and geotechnical investigations conducted to characterize the site. Technical experts from the U.S. Geological Survey (USGS) accompanied the staff during the January 2009 site visit to assist with the evaluation of the geologic and seismic data.

During the site visits conducted in April to May 2008 and January 2009, the staff examined site location bedrock exposures in the excavations for the original CNS Unit 3 (i.e., WLS Unit 2 with the top of sound rock, not foundation grade level bedrock, exposed) and CNS Unit 2 (with foundation level bedrock exposed), and around the original CNS Unit 1 (WLS Unit 1) where no concrete had been previously placed. The staff also examined saprolite in the walls of the CNS excavations, core samples collected during site characterization investigations for the WLS application, and select bedrock exposures in the site area to assist with understanding the complex tectonic deformation history reflected in rock units at the site location.

The July 2011 site visit occurred after the applicant had provided Version 0 of the compiled geologic map report for the foundation grade level bedrock surface in the CNS/WLS Unit 1 excavation. The staff conducted a data documentation audit of Version 0 of the Duke report presenting the compiled geologic map and associated data for CNS Unit 1 on June 6, June 9, and June 10, 2011. The staff also audited Version 1 of the Duke report between October 25 and 27, 2011, to ensure that the final version of the report contained the necessary information related to documenting similarities in both lithologies and geologic features found in the CNS Unit 1 (proposed WLS Unit 1) excavation and in the adjacent excavations for CNS Unit 3 (proposed WLS Unit 2) and CNS Unit 2. During the site visit, the staff focused on analyzing lithologies and geologic structures shown on the compiled geologic map for the CNS Unit 1 excavation, which now lies under concrete. The staff directly examined lithologies and geologic structures in the existing adjacent excavations for the original CNS Unit 3 (i.e., WLS Unit 2 excavated to the top of sound rock) and CNS Unit 2 (located between WLS Units 1 and 2 and excavated to foundation grade level) and confirmed that the compiled geologic map accurately represents lithologies and geologic structures found in the CNS/WLS Unit 1 excavation. The data documentation audits and July site visit also enabled the staff to verify that lithologies and geologic features are similar among WLS/CNS Unit 1, WLS Unit 2/CNS Unit 3, and CNS Unit 2 and that no capable tectonic structures or other potentially detrimental geologic features occur in the concrete-covered excavation for proposed WLS Unit 1 (previous CNS Unit 1) or in the existing open excavations for CNS Unit 3 (proposed WLS Unit 2) and CNS Unit 2.

During the site visit in February 2014, the staff directly examined core from the following four select boreholes placed at the modified locations of the nuclear islands for WLS Unit 1, which is coincident with the previous CNS Unit 1, and WLS Unit 2, which is coincident with previous CNS Unit 3: B-2000 (total depth, or TD, = 38.4 m (126.0 ft)) and B-2002 (TD = 68.8 m (225.6 ft)) for Unit 1; B-2005 (TD = 68.6 m (225.0 ft)) and B-2006 (TD = 30.8 m (101.0 ft)) for Unit 2. The applicant moved the Unit 1 nuclear island footprint 20.1m (66 ft) south and 15.2 m (50 ft) east of CNS Unit 1 to avoid an area of deeply weathered saprolitic bedrock at the northwest corner of old CNS Unit 1. The applicant also moved the Unit 2 nuclear island footprint 20.1 m (66 ft) south. The site visit occurred after the staff had reviewed core logs and core images for all seven bore holes placed to characterize subsurface geologic conditions in the relocated nuclear island footprints in order to determine which cores were most appropriate for direct examination. The purpose of the site visit was to verify that lithologic units and tectonic deformation features intercepted by the boreholes in the subsurface at the locations of the relocated footprints were similar to what had been observed in boreholes from the previous footprint locations and in the foundation rock units exposed in the existing original excavations for the CNS site, including lithologic units and tectonic deformation structures previously mapped in the foundation grade level bedrock of the CNS Unit 1 excavation. Based on a detailed examination of the four cores and field observation of the locations of the new boreholes combined with consideration of the information derived from previous boreholes, examination of exposed bedrock that will comprise the foundation grade level units for WLS, and review of the geologic maps of the CNS Unit excavation, the staff concluded that lithologic units and tectonic deformation features similar to those previously described occur in the cores from the relocated footprints. Furthermore, B-2000 clearly illustrates that the applicant has avoided the zone of deeply weathered rock at the northwest corner of the original WLS Unit 1 since the core shows that, at 2.96 m (9.7 ft) depth, the borehole passes from the tightly-bonded concrete-bedrock interface into only slightly weathered bedrock.

Since tectonic deformation features (specifically ductile and brittle shear zones and other older tectonic features associated with the complex deformation history of the site) occur in bedrock in the site area and at the site location, on February 23 and July 20, 2009, the staff also audited materials used by the applicant to constrain the timing of the development of minor ductile and brittle shear zones mapped in the excavations at the original CNS site as part of the technical evaluation of WLS COL FSAR Section 2.5.1. These materials contained data on radiometric age dates previously acquired by the applicant for characterization of the original CNS site. Understanding the constraints on the timing of deformation provided by these original age dates is important because the applicant did not obtain additional radiometric ages during site characterization investigations for WLS, but rather relied on the previously acquired age dates to document the timing of the development of tectonic deformation features in the WLS application. In addition, the staff audited Revision 0 and Revision 1 of the report discussing the compiled geologic map and associated data on June 6, June 9 and 10, and October 25 through 27, 2011. The Duke report documented the characterization of the foundation grade level bedrock surface in the WLS Unit 1 excavation based on the previous geologic mapping of foundation grade level bedrock conducted in the co-located CNS Unit 1 excavation.

Through the multiple site visits and data documentation audits described above, the staff assessed the interpretations, assumptions, and conclusions made by the applicant regarding basic geologic and seismic information for WLS and acceptability of geologic conditions at WLS Units 1 and 2, particularly with regard to age constraints for the ductile and brittle shear zones. The staff confirmed that no geologic features found in foundation bedrock units represent capable tectonic structures.

Based on the discussion of the basic geologic and seismic information presented in WLS COL FSAR Section 2.5.1, and the independent staff evaluation described in Section 2.5.1.4 of this report, the staff concludes that the applicant provided the information required to satisfy COL Information Item 2.5-1.

Sections 2.5.1.4.1, "Regional Geology," and 2.5.1.4.2, "Site Geology," of this report present the staff's technical evaluation of the information provided by the applicant in WLS COL FSAR Section 2.5.1 and the applicant's responses to RAIs for WLS COL FSAR Section 2.5.1.
In addition to RAIs addressing specific technical issues related to regional and site geology of WLS, discussed in detail below, the staff prepared several editorial RAIs to clarify certain descriptive statements made by the applicant in the WLS COL FSAR and to qualify geologic features illustrated in WLS COL FSAR figures. This technical evaluation does not discuss these editorial RAIs because they do not alter the substantive technical information provided by the applicant. Also, this evaluation does not discuss RAIs related to geologic issues resolved in FSARs previously prepared for other sites in the CEUS, but rather addresses these issues by reference to and a summary of the information used to resolve them in those FSARs.

2.5.1.4.1 Regional Geology

The staff focused the review of WLS COL FSAR Section 2.5.1.1 on the descriptions provided for physiography, geomorphology, stratigraphy, tectonic setting, seismicity, and paleoseismology within a 320 km (200 mi) radius of WLS. The staff specifically focused on geologic features in the site region interpreted to be Quaternary (2.6 Ma to present) in age.

2.5.1.4.1.1 *Regional Physiography, Geomorphology, and Stratigraphy*

WLS COL FSAR Section 2.5.1.1.1 discusses the regional physiography, geomorphology, and stratigraphy of WLS, including the geologic setting of the Piedmont physiographic province in which WLS lies. Figure 02.05.01-1 of this report shows the location of WLS in relation to the parts of the five physiographic provinces that occur in the site region (i.e., the Appalachian Plateau, Valley and Ridge, Blue Ridge, Piedmont, and Atlantic Coastal Plain).

The staff focused the review of WLS COL FSAR Section 2.5.1.1.1 on the applicant's discussion of the Carolina Zone of the Piedmont physiographic province in which WLS lies. In RAI 59, Questions 02.05.01-3 and 02.05.01-4, the staff requested that the applicant clearly distinguish the Carolina Zone from adjacent lithotectonic terranes, and to incorporate pertinent information from more recent published references for describing the lithologic, stratigraphic, and tectonic characteristics of the Carolina Zone because WLS COL FSAR Section 2.5.1.1.1 lacked the information derived from these references. In response to RAI 59, Questions 02.05.01-3 and 02.05.01-4, the applicant clearly distinguished the Carolina Zone from the adjacent lithotectonic terranes by modifying WLS COL FSAR Figures 2.5.1-202a and 2.5.1-202b, as well as by including new WLS COL FSAR Figure 2.5.1-235 to show the correlations between physiographic provinces and recent classifications of lithotectonic terranes in the site region. The applicant also presented information derived from the more recent published references that describe the lithologic, stratigraphic, and tectonic characteristics of the Carolina Zone. The applicant incorporated the three figures and the updated descriptions of the Carolina Zone into WLS COL FSAR Section 2.5.1.

Based on review of the responses to RAI 59, Questions 02.05.01-3 and 02.05.01-4 and the figures and descriptions provided in WLS COL FSAR Section 2.5.1, as well as an independent examination of the current literature cited by the applicant, the staff concludes that the applicant adequately distinguished and characterized the Carolina Zone in which the site lies. The staff draws this conclusion because the applicant incorporated information from up-to-date references into WLS COL FSAR Section 2.5.1. Accordingly, the staff considers RAI 59, Questions 02.05.01-3 and 02.05.01-4 resolved.

Based on review of F WLS COL FSAR Section 2.5.1.1.1, the responses to RAI 59, Questions 02.05.01-3 and 02.05.01-4, and an independent examination of current literature cited by the applicant, the staff finds that the applicant provided a thorough and accurate description of regional physiography, geomorphology, and stratigraphy in support of the WLS application.

2.5.1.4.1.2 *Regional Tectonic Setting*

WLS COL FSAR Section 2.5.1.1.2 discusses the regional tectonic setting of WLS, including regional geologic history, tectonic stress in the mid-continent region, gravity and magnetic data of the site region and site vicinity, and principal regional tectonic structures. The following paragraphs present the staff's evaluation of WLS COL FSAR Sections 2.5.1.1.2.1 through 2.5.1.1.2.4. The staff performed the most detailed evaluation for the 15 potential Quaternary tectonic structures identified in the site region because these structures represent potentially capable tectonic features.

2.5.1.4.1.2.1 Regional Geologic History

In WLS COL FSAR Section 2.5.1.1.2.1, the applicant discussed geologic history of WLS region. The staff focused the review of WLS COL FSAR Section 2.5.1.1.2.1 on the applicant's discussion of the possibility that earthquakes in the eastern part of the Piedmont physiographic province and beneath the Coastal Plain province may be spatially associated with buried normal faults that initially developed during the early and middle Mesozoic rifting of the continental crust. The applicant indicated that there is no definitive spatial correlation of seismicity with any faults known to bound Mesozoic rift basins in the site region. The staff evaluated this interpretation further as described in Section 2.5.1.4.1.2.4 below.

Based on review of WLS COL FSAR Section 2.5.1.1.2.1 and an independent examination of references cited by the applicant, the staff concludes that the applicant presented information pertinent for describing the geologic history of the WLS region. The staff makes this conclusion because the applicant included information to adequately address the geologic history for the region including WLS from Paleozoic (542 to 251 Ma) to Quaternary (2.6 Ma to present) time.

Based on review of WLS COL FSAR Section 2.5.1.1.2.1 and an independent examination of references cited by the applicant, the staff finds that the applicant provided a thorough and accurate description of regional geologic history in support of the WLS application.

2.5.1.4.1.2.2 Tectonic Stress in the Mid-Continent Region

In WLS COL FSAR Section 2.5.1.1.2.2, the applicant discussed tectonic stress in the mid-continent region, including data from the CEUS SSC model (NUREG-2115) compilations. The staff focused the review of WLS COL FSAR Section 2.5.1.1.2.2 on information used by the applicant to document that no new concerns exist regarding the potential for tectonic activity along geologic structures in the site region due to changes in orientation of the regional stress field. Based on review of WLS COL FSAR Section 2.5.1.1.2.2 and an independent examination of the references cited by the applicant in that WLS COL FSAR section, the staff concludes that the data presented by the applicant document a continuing northeast-southwest orientation for maximum horizontal compressive stress in the site region. The staff draws this conclusion because all published regional stress data presented by the applicant and independently reviewed by the staff fully support it.

Based on review of WLS COL FSAR Section 2.5.1.1.2.2 and an independent examination of references cited by the applicant in that WLS COL FSAR section, the staff finds that the applicant provided a thorough and accurate description of tectonic stress in the mid-continent region in support of the WLS application.

2.5.1.4.1.2.3 Gravity and Magnetic Data

In WLS COL FSAR Section 2.5.1.1.2.3, the applicant discussed gravity and magnetic data for the site region and site vicinity, including those data reprocessed and published as part of the CEUS SSC model (NUREG-2115) database. The staff focused the review of WLS COL FSAR Section 2.5.1.1.2.3 on the applicant's statements related to determining the presence of Cenozoic tectonic structures based on regional gravity and magnetic data because the applicant did not provide any explanation of how these regional data sets can be used to determine that a specific tectonic feature is Cenozoic in age. In RAI 59, Question 02.05.01-5, the staff requested that the applicant define the criteria applied for determining the presence of Cenozoic tectonic structures based on regional gravity and magnetic data, and to discuss the significance of magnetic anomalies in the site vicinity in relation to geologic structure and rock type. In the response to RAI 59, Question 02.05.01-5, the applicant indicated that the statements made about gravity and magnetic data in WLS COL FSAR Section 2.5.1.1.2.3 should not be interpreted to imply that these data can be used to explicitly define the ages of geologic features. The applicant incorporated revised text into WLS COL FSAR Section 2.5.1.1.2.3 to correct this misconception. Also in the response to RAI 59, Question 02.05.01-5, the applicant explained that the low magnetic anomalies, which occur in Charlotte terrane rocks of the site vicinity both southeast and northeast of WLS, reflect the presence of Paleozoic (542 to 251 Ma) intrusive igneous bodies that are relatively non-magnetic, rather than faults or shear zones. The applicant provided two additional figures to illustrate magnetic anomaly patterns in the site vicinity and site area, and incorporated revised text and the two new figures into the WLS COL FSAR to better describe the magnetic anomalies.

Based on review of the response to RAI 59, Question 02.05.01-5, figures and descriptions provided in WLS COL FSAR Section 2.5.1.1.2.3, and an independent examination of references cited by the applicant in that WLS COL FSAR section, the staff concludes that the applicant corrected the statement implying that regional gravity and magnetic data could be used to define the presence of Cenozoic structures and qualified that low magnetic anomalies in the site vicinity generally reflect lithologies rather than tectonic structures. The staff draws this conclusion with regard to the magnetic anomalies because published field evidence cited by the applicant in the FSAR and reviewed by the staff shows that intrusive igneous bodies occur at the locations of the magnetic lows southeast and northeast of the site.

Based on review of WLS COL FSAR Section 2.5.1.1.2.3, an independent examination of references cited by the applicant in that WLS COL FSAR section, and the response to RAI 59, Question 02.05.01-5, the staff finds that the applicant provided a thorough and accurate description of regional gravity and magnetic data in support of WLS application. Accordingly, the staff considers RAI 59, Question 02.05.01-5 resolved.

2.5.1.4.1.2.4 Principal Regional Tectonic Structures

In WLS COL FSAR Section 2.5.1.1.2.4, the applicant discussed Paleozoic (542-251 Ma), Mesozoic (251-65.5 Ma), Cenozoic (65.5 Ma to present), and Quaternary (2.6 Ma to present) tectonic structures that occur within 320 km (200 mi) of WLS (i.e., the site region). The applicant specifically assessed the 15 potential Quaternary tectonic structures postulated to occur in the site region.

The staff focused the review of WLS COL FSAR Section 2.5.1.1.2.4 primarily on the Quaternary tectonic structures postulated to occur in the site region since they represent potentially capable tectonic features. The staff also focused on understanding age constraints proposed by the applicant for the regional faults interpreted to be pre-Quaternary in age. This secondary focus was important for documenting the timing of last displacement on these structures to ensure that none are Quaternary in age (i.e., potentially capable tectonic features).

Regional Geophysical Anomalies and Lineaments

In WLS COL FSAR Section 2.5.1.1.2.4.1, the applicant discussed the geophysical anomalies and lineaments that occur in the site region. From southeast to northwest, these features include the ECMA; the southeastern limit of Iapetan normal faulting; the Clingman and Ocoee lineaments; the NYAL; the Appalachian gravity gradient; the northwest boundary of Iapetan normal faulting; the Appalachian thrust front; and the Grenville Front.

Based on review of the information presented in WLS COL FSAR Section 2.5.1.1.2.4.1 regarding the association of regional geophysical anomalies and lineaments with old tectonic features, as well as an independent examination of references cited by the applicant in that WLS COL FSAR section, the staff concludes that none of the anomalies or lineaments represent a capable tectonic structure. The staff draws this conclusion because the preponderance of available data strongly supports the interpretation that the anomalies and lineaments show an association with tectonic features that initially developed more than 65.5 Ma ago. In addition, the staff notes that WLS COL FSAR Section 2.5.2.2.3 explains how the CEUS SSC model incorporated regional gravity and magnetic anomalies to define the boundaries of seismotectonic zones in and adjacent to the site region. Therefore, the staff further concludes that the applicant considered the latest regional geophysical data by using the CEUS SSC model in the PSHA for WLS.

Based on review of WLS COL FSAR Section 2.5.1.1.2.4.1 and an independent examination of references cited by the applicant in that WLS COL FSAR section, the staff finds that the applicant provided a thorough and accurate description of regional geophysical anomalies and lineaments in support of the WLS application.

Regional Paleozoic Tectonic Structures

In WLS COL FSAR Section 2.5.1.1.2.4.2, the applicant discussed 13 regional tectonic structures of Paleozoic (542 to 251 Ma) age, which occur in the WLS region, specifically the Kings Mountain, Gold Hill-Silver Hill, Middleton-Lowdensville, Beaver Creek, Modoc, Chappells, Hyco and Boogertown-Southwest Extension shear zones and the Tinsley Bridge, Brevard, Cross Anchor, Brindle Creek, and Reedy River faults. The staff focused the review of WLS COL FSAR Section 2.5.1.1.2.4.2 on the information used to document ages of the faults and shear zones interpreted to be Paleozoic in age, which occur in the site region, to ensure that none of these structures have any potential for reactivation as capable tectonic features. In RAI 59, Questions 02.05.01-11 through 02.05.01-14, the staff requested that the applicant provide the published information used to document a Paleozoic age for the faults and shear zones discussed in WLS COL FSAR Section 2.5.1.1.2.4.2, but also include some tectonic structures interpreted to be Paleozoic in age that the applicant did not specifically discuss in that WLS COL FSAR section. In the responses to RAI 59, Questions 02.05.01-11 through 02.05.01-14, the applicant provided additional information and updated references to clearly document a Paleozoic age for the last movement on the faults and shear zones interpreted to be Paleozoic. The applicant also modified WLS COL FSAR Figures 2.5.1-209 and 2.5.1-210 to locate certain structures and incorporated changes in WLS COL FSAR Section 2.5.1.1.2.4.2, to present the additional information, updated references, and modified figures.

Based on review of the applicant's responses to RAI 59, Questions 02.05.01-11 through 02.05.01-14 and the revisions made in WLS COL FSAR Section 2.5.1.1.2.4.2, including both text and WLS COL FSAR Figures 2.5.1-209 and 2.5.1-210, as well as independent examination of references cited by the applicant in that WLS COL FSAR section, the staff concludes there is strong evidence that these faults and shear zones are Paleozoic in age and that none are capable tectonic structures. The staff makes this conclusion because the age dates presented by the applicant constrain the last displacements along these regional tectonic structures to Paleozoic time, and because the structures commonly exhibit deformation fabrics indicative of a deep-seated, high-temperature metamorphic environment (i.e., the geologic environment in which Paleozoic deformational events occurred in the site region).

Based on review of WLS COL FSAR Section 2.5.1.1.2.4.2, the applicant's responses to RAI 59, Questions 02.05.01-11 through 02.05.01-14, the revisions incorporated into WLS COL FSAR

Section 2.5.1.1.2.4.2, and an independent examination of references cited by the applicant in that WLS COL FSAR section, the staff finds that the applicant provided a thorough and accurate description of regional Paleozoic tectonic structures in support of the WLS application. Accordingly, the staff considers RAI 59, Questions 02.05.01-11 through 02.05.01-14 resolved.

Regional Mesozoic Tectonic Structures

In WLS COL FSAR Section 2.5.1.1.2.4.3, the applicant discussed six regional tectonic structures of Mesozoic (251 to 65.5 Ma) age that occur in the WLS region, specifically Mesozoic rift basins and the Wateree Creek, Summers Branch, Ridgeway, Longtown, and Mulberry Creek faults. The staff focused the review of WLS COL FSAR Section 2.5.1.1.2.4.3 on the information used by the applicant to document the ages of the structures interpreted to be Mesozoic in age to ensure that none of these structures have any potential for reactivation as capable tectonic features. The staff also focused on the data used by the applicant to conclude that faults bounding Mesozoic rift basins do not show any spatial relationship with seismicity.

<u>Wateree Creek, Summers Branch, and Ridgeway Faults</u>

In RAI 59, Question 02.05.01-16, the staff requested that the applicant summarize the information used to constrain the timing of fault displacement along the Summers Branch and Ridgeway faults since the applicant interpreted these two faults to be Mesozoic (251 to 65.5 Ma) in age based primarily on their association with the Wateree Creek fault. In response to RAI 59, Question 02.05.01-16, based on field data from Secor and others, the applicant reported that igneous dikes of probable Triassic (251 to 201.6 Ma) to early Jurassic (201.6 to 176 Ma) age cross the Wateree Creek fault without offset, constraining the age of last movement on that fault to be older than 176 Ma. The applicant stated that Secor and others noted strong similarities between the Wateree Creek and Ridgeway faults, including fault length and a northerly strike direction. In addition, the applicant pointed out that Secor and others presented field data showing the Ridgeway fault does not offset an overlying Mesozoic (specifically Upper Cretaceous, 99.6 to 65.2 Ma) stratigraphic unit, indicating that the last movement on the Ridgeway fault is older that Upper Cretaceous. The applicant stated that, while evidence for the Summers Branch fault is speculative, the strike direction and length of this fault are similar to the strike direction and length of the Wateree Creek and Ridgeway faults as defined by Secor and others. The applicant reported that a more recent geologic map prepared by Maher and others, which included Secor as a co-author, did not show the Summers Branch fault. Based on field observations that suggest strong similarities between the Summers Branch (if it exists), Wateree Creek, and Ridgeway faults, the applicant concluded that the last displacements on these three faults are not younger than Mesozoic (i.e., > 65.5 Ma).

Based on review of the applicant's response to RAI 59, Question 02.05.01-16 and direct field examination of the Wateree Creek Fault performed by the staff with USGS geologists during a March 2009 site visit related to the review of the FSAR for the V.C. Summer site, the staff concludes that existing field evidence strongly supports the interpretation that these three faults are Mesozoic in age and none represent capable tectonic structures. The staff draws this conclusion because of the well-defined Mesozoic age constraint on the Wateree Creek fault; the similarities between the three faults described by the applicant; and the field characteristics of the Wateree Creek fault observed by the staff, which indicate that this fault offsets older

Paleozoic rock units and does not exhibit a deformation fabric clearly related to late-stage brittle failure. Accordingly, the staff considers RAI 59, Question 02.05.01-16 resolved.

<u>Longtown and Mulberry Creek Faults</u>

In WLS COL FSAR Section 2.5.1.1.2.4.3, based on Barker and Secor, the applicant reported that four igneous dikes of probable Triassic (251 to 201.6 Ma) age cross the Longtown fault without offset, establishing a minimum age of Triassic for this fault. Regarding the Mulberry Creek fault, the applicant noted that silicified breccia and microbreccia characterize this fault and suggested a Late Triassic to Early Jurassic (235 to176 Ma) age for the fault based on the fact that, as documented by West, silicified fault zones found in North and South Carolina are most commonly Mesozoic in age. The staff concludes that the applicant provided sufficient information to document a Mesozoic age for both the Longtown and Mulberry Faults because published field data cited by the applicant and reviewed by the staff clearly constrain timing of the last displacement along these two structures to Mesozoic.

<u>Mesozoic Rift Basins</u>

In WLS COL FSAR Section 2.5.1.1.2.4.3, the applicant reported that Mesozoic rift basins have long been considered potential earthquake sources in the CEUS. Although the applicant concluded, in WLS COL FSAR Section 2.5.1.1.2.4.3, that no correlation exists between seismicity and Mesozoic rift basins in the site region, the applicant stated in WLS COL FSAR Section 2.5.3.1.5 that two September 2006 earthquakes, which occurred near Bennettsville, SC, about 121 km (75 mi) east-southeast of the site, showed a spatial association with a small Mesozoic extensional basin lying beneath the Coastal Plain as mapped by Benson. In RAI 59, Question 02.05.01-17, the staff requested that the applicant provide information to support the conclusion that faults bounding Mesozoic rift basins do not exhibit any spatial association with seismicity, and to summarize the logic for stating that Mesozoic structures in the site region are not capable tectonic sources. In response to RAI 59, Question 02.05.01-17, the applicant stated that, based on the most current data available, no positive correlation exists between earthquakes and Mesozoic rift basins in the site region; that there is a definitive lack of spatial correlation between Mesozoic basins and seismicity within 80.5 km (50 mi) of WLS, based on an assessment of tectonic features and seismicity within that area; and that no data demonstrate Quaternary reactivation of any Mesozoic basin-bounding faults in the site region. The applicant also stated that the two September 2006 earthquakes that occurred near Bennettsville had large uncertainties in location and could not be definitively correlated with any specific tectonic feature.

Based on review of the applicant’s response to RAI 59, Question 02.05.01-17, the staff concludes that faults bounding Mesozoic rift basins do not exhibit any definitive spatial association with seismicity and that no Mesozoic structures that occur in the site region represent capable tectonic sources. The staff draws these conclusions because a preponderance of data independently reviewed by the staff strongly supports these two interpretations. In addition, staff notes that the Extended Continental Crust-Atlantic Margin (ECC-AM) seismotectonic zone of the CEUS SSC model (NUREG-2115), one of the multiple zones included in the seismic hazard calculation for WLS, contains Mesozoic tectonic structures of the site region. The applicant discussed the ECC-AM seismotectonic zone in WLS COL

FSAR Section 2.5.2.2.3.2. Accordingly, the staff considers RAI 59, Question 02.05.01-17 resolved.

Based on review of WLS COL FSAR Section 2.5.1.1.2.4.3, an independent examination of references cited by the applicant in that WLS COL FSAR section, and the applicant's responses to RAI 59, Questions 02.05.01-16 and 02.05.01-17, the staff finds that the applicant provided a thorough and accurate description of regional Mesozoic tectonic structures in support of the WLS application.

Regional Cenozoic Tectonic Structures

In WLS COL FSAR Section 2.5.1.1.2.4.4, the applicant discussed regional tectonic structures of Cenozoic (65.5 Ma to present) age that occur in WLS region. The applicant stated that only a few structures in the site region show possible evidence of Cenozoic activity, including the Camden fault, the Prowell faults, and the Cape Fear and Yamacraw arches.

The staff focused the review of WLS COL FSAR Section 2.5.1.1.2.4.4 on the data used by the applicant to conclude that the Cape Fear and Yamacraw arches are not active tectonic features. In RAI 59, Question 02.05.01-19, the staff requested that the applicant refer to the primary sources of data that render this conclusion about these features plausible, rather than relying on the data compilation of Crone and Wheeler. In response to RAI 59, Question 02.05.01-19, the applicant cited information from the published literature and stated that there is only limited evidence to constrain timing of the most recent movement on the Cape Fear and Yamacraw arches. The applicant used the most recent field data from Gohn to state that warping related to the Cape Fear arch did not affect stratigraphic units younger than late Tertiary (i.e., > 2.6 Ma, so pre-Quaternary in age), and that tectonic history of the Yamacraw arch is likely analogous since they both represent broad structural upwarps formed by the same processes. The applicant incorporated changes into the WLS COL FSAR to present the information derived from Gohn.

Based on review of the applicant's response to RAI 59, Question 02.05.01-19, independent examination of the references cited by the applicant, and text revisions in WLS COL FSAR Section 2.5.1.1.2.4.4, the staff concludes that the Cape Fear and Yamacraw arches likely do not represent active tectonic structures. The staff draws this conclusion because the most recent field data support the interpretation that upwarping related to the Cape Fear and Yamacraw arches did not affect stratigraphic units younger than late Tertiary. Accordingly, the staff considers RAI 59, Question 02.05.01-19 resolved.

Based on a review of WLS COL FSAR Section 2.5.1.1.2.4.4, independent examination of references cited by the applicant in that WLS COL FSAR section, and the applicant's response to RAI 59, Question 02.05.01-19, the staff finds that the applicant provided a thorough and accurate description of regional Cenozoic tectonic structures in support of the WLS application.

Regional Quaternary Tectonic Structures

In WLS COL FSAR Section 2.5.1.1.2.4.5, the applicant identified fifteen potential Quaternary tectonic features postulated to occur in WLS region based on data compiled by Crone and Wheeler and Wheeler. These authors classified potential Quaternary features as Class A, B, C, or D based on strength of evidence for Quaternary deformation derived from published information. These 15 potential Quaternary tectonic features include the Class A Charleston area, Bluffton, and Georgetown liquefaction features; the Class B Pembroke faults; and the Class C Fall Lines of Weems, Belair and Lindside fault zones, Giles County and Eastern Tennessee seismic zones, East Coast fault system, Cape Fear arch, and Pen Branch, Hares Crossroads, Stanleytown-Villa Heights, and Cooke faults. Figure 2.5.1-3 of this report shows the locations of these potential Quaternary features in relation to WLS.

The staff's technical evaluations of the information provided by the applicant in the WLS COL FSAR and in responses to RAIs in regard to the Cape Fear arch, Charleston area features (i.e., the Cooke fault, East Coast Fault System, and Charleston, Bluffton, and Georgetown liquefaction features), and the Eastern Tennessee and Giles County seismic zones are in various parts of this report. The paragraphs that immediately follow present the staff's evaluations of information provided by the applicant in the WLS COL FSAR and in responses to RAIs related to the Fall Lines of Weems, the Belair and Lindside fault zones, and the Pen Branch, Hares Crossroads, Stanleytown-Villa Heights, and Pembroke faults. The staff focused the review on information used by the applicant to determine that none of these features represent capable tectonic structures.

<u>Fall Lines of Weems (Class C)</u>

For the Fall Lines of Weems, based on published literature, field reconnaissance, and review of the staff's detailed evaluation of these features for the North Anna ESP application as dodumented in NUREG-1835, the applicant concluded that these features developed because of contrasting resistance to erosion of adjacent rock types. The detailed evaluation performed by the NRC for the North Anna ESP application (NUREG-1835) provided the primary basis for staff to also conclude that the Fall Lines of Weems are not tectonic in origin and do not represent capable tectonic structures.

<u>Belair Fault Zone (Class C)</u>

Based on information from Prowell and O'Connor, the applicant constrained the timing of last movement on the Belair fault zone to between 33.9 Ma (i.e., late Eocene) and 26,000 years ago. The applicant stated that the Belair fault zone may be related to the regional Augusta fault zone and, although available data do not clearly demonstrate Quaternary movement along the Belair zone, these data also do not preclude Quaternary displacement on the Belair fault zone. In RAI 59, Question 02.05.01-20, the staff requested that the applicant discuss how the inference of possible Quaternary movement on the Belair fault zone, coupled with a possible structural relationship to the Augusta fault zone, might affect seismic hazard at WLS. In response to RAI 59, Question 02.05.01-20, the applicant noted that existing data do not confirm a common slip history or similar sense of slip for the Augusta and Belair fault zones. The applicant also noted that none of the faults in the site region exhibiting Cenozoic reactivation show any evidence of Quaternary displacement and, therefore, are not capable tectonic features that must be specifically assessed in regard to seismic hazard at WLS.

Based on the applicant's response to RAI 59, Question 02.05.01-20 and an independent examination of references cited by the applicant, the staff concludes that there is no definitive evidence to document Quaternary displacement on the Belair or Augusta fault zones and that the Belair fault zone is not a capable tectonic structure. The staff draws these conclusions because the published field data cited by the applicant and independently reviewed by the staff support the interpretation that the Belair fault does not show conclusive evidence for Quaternary movement. Accordingly, the staff considers RAI 59, Question 02.05.01-20 resolved.

Pen Branch Fault (Class C)

Based on information collected for the Savannah River site and the results of investigations performed for the Vogtle ESP application, the applicant concluded that no evidence exists for post-Eocene (i.e., < 33.9 Ma) displacement along the Pen Branch fault and the fault is not a capable tectonic structure. Based on an independent review of the information presented by Cumbest and others and data included in Vogtle ESP application, the staff also concludes that the Pen Branch fault does not represent a capable tectonic structure. The staff makes this conclusion because timing of last movement on the fault is constrained by field data to be older than Quaternary (i.e., > 2.6 Ma).

Hares Crossroads Fault (Class C)

The applicant noted that the postulated Hares Crossroads fault occurs in a single roadcut exposure in Coastal Plain sediments and concluded that this structure is a local feature that most likely formed as a result of landsliding rather than tectonic faulting. Because this postulated fault occurs in a single exposure of Coastal Plain sediments and is not laterally continuous, the staff concludes that this feature is non-tectonic in origin and does not represent a capable tectonic structure.

Lindside Fault Zone (Class C)

The applicant reported that the Lindside fault zone does not exhibit any evidence of Quaternary displacement, and that orientation of the fault zone is unfavorable for reactivation in the current stress field. Based on a lack of field data suggesting Quaternary slip on the fault zone, the staff concludes that the Lindside fault zone is not a capable tectonic structure.

Stanleytown-Villa Heights Faults (Class C)

The applicant stated that field evidence derived from published literature suggests the postulated Stanleytown-Villa Heights faults are likely the result of landsliding, rather than a tectonic event. These postulated faults juxtapose Quaternary (2.6 Ma to present) alluvium against Cambrian (542 to 488 Ma) rock units and occur as short features characterized by downslope displacements along the surfaces which define the postulated fault surfaces. In RAI 59, Question 02.05.01-21, the staff requested that the applicant concisely summarize the primary evidence for a non-tectonic, landslide mechanism for these postulated faults since they affect Quaternary age deposits. In response to RAI 59, Question 02.05.01-21, based on Conley and Toewe, the applicant reported that these proposed faults do not exhibit any shear fabrics and have a limited lateral extent, and that the juxtaposition of Quaternary and Cambrian units can be readily interpreted as a depositional contact between alluvium and bedrock. The applicant also pointed out that no publications more recent than that of Conley and Toewe address these features.

Since a preponderance of published field evidence cited by the applicant and reviewed by the staff suggests that the postulated Stanleytown-Villa Heights faults are likely non-tectonic in origin and related either to landsliding or a depositional contact between Quaternary alluvium and weathered Cambrian bedrock, the staff concludes that these features are not capable tectonic structures. Accordingly, the staff considers RAI 59, Question 02.05.01-21 resolved.

Pembroke Faults (Class B)

The applicant reported that the postulated Pembroke faults exhibit no geomorphic expression, occur in alluvial terrace deposits of latest Pliocene (3.6 to 2.6 Ma) to Quaternary (2.6 Ma to present) age, and overlie dissolution-prone Ordovician (488 to 444 Ma) carbonate rocks. The applicant stated that these features may be directly related to collapse over subsurface dissolution features rather than a tectonic event. In RAI 59, Question 02.05.01-22, the staff requested that the applicant summarize the primary information on fault geometry and fault length, and to present evidence regarding whether the postulated Pembroke faults are tectonic or non-tectonic (i.e., related to dissolution collapse) in origin. In response to RAI 59, Question 02.05.01-22, the applicant stated that, because these features occur in a single roadcut, fault length cannot be determined. Based on Peavy and Sayer, the applicant reported that numerous sinkholes attributed to dissolution of underlying Ordovician carbonates occur in the area. The applicant also noted Law and others suggested that subsurface cavities detected in sandy clays developed as a result of dissolution collapse and consequent upward migration of collapse structures from underlying carbonate bedrock into the overlying sediments.
In addition, the applicant reported that the same researchers who initially proposed that the Pembroke faults formed in response to tectonic processes more recently suggested karst dissolution as an equally viable origin for these features. Therefore, the applicant did not interpret the Pembroke faults as capable tectonic structures.

Based on the response to RAI 59, Question 02.05.01-22 and an independent review of references cited by the applicant, the staff concludes that the postulated Pembroke faults are likely related to collapse of alluvial terrace deposits as a result of dissolution of underlying carbonate bedrock. The staff draws this conclusion because carbonate rocks underlie the terrace deposits at the location of these features, and researchers who initially interpreted the Pembroke faults as tectonic in origin more recently suggested dissolution collapse as the causative mechanism. Accordingly, the staff considers RAI 59, Question 02.05.01-22 resolved.

Based on review of WLS COL FSAR Section 2.5.1.1.2.4.5, an independent examination of references cited by the applicant in that WLS COL FSAR section, and the applicant's responses to RAI 59, Questions 02.05.01-20 through 02.05.01-22, the staff finds that the applicant provided a thorough and accurate description of the Fall Lines of Weems, the Belair and Lindside fault zones, and the Pen Branch, Hares Crossroads, Stanleytown-Villa Heights, and Pembroke faults, all of which are potential Quaternary tectonic structures postulated to occur in the site area, in support of the WLS application.

2.5.1.4.1.3 *Regional Seismicity and Paleoseismology*

WLS COL FSAR Section 2.5.1.1.3 discusses seismicity and paleoseismology of WLS region, with emphasis on the Charleston, East Tennessee, and Giles County seismic zones. WLS COL FSAR Section 2.5.1.1.3 also addresses two seismic zones that lie outside the site region, the

New Madrid and Central Virginia seismic zones. The applicant noted that only Charleston and the New Madrid seismic zone are RLME sources as defined in the CEUS SSC model (NUREG-2115). Figure 2.5.1-4 of this report shows the locations of these 5 seismic source zones relative to WLS. Sections 2.5.1.4.1.3.1 through 2.5.1.4.1.3.1.5 of this report present the staff's technical evaluation of the information provided by the applicant in WLS COL FSAR Section 2.5.1.1.3 related to regional seismicity and paleoseismology.

2.5.1.4.1.3.1 Charleston Tectonic Features

In WLS COL FSAR Section 2.5.1.1.3.2.1, the applicant discussed ten source faults and three seismic source zones postulated to occur in the Charleston area, as well as seismically-induced liquefaction features. The applicant noted that, although the 1886 Charleston earthquake is the largest historic seismic event in the eastern United States, researchers have not yet identified a causative fault, but rather infer an earthquake source based on geomorphology, geologic features, and instrumented seismicity.

Potential Charleston Area Source Faults and Seismic Zones

In WLS COL FSAR Section 2.5.1.1.3.2.1, the applicant discussed ten potential source faults for the 1886 Charleston earthquake (i.e., the postulated East Coast Fault system, Helena Banks fault zone, and Adams Run, Ashley River, Charleston, Cooke, Sawmill Branch, Dorchester, Summerville, and Woodstock faults). All of these structures lie within the meizoseismal area of the 1886 earthquake, except the Helena Banks fault zone, as shown in Figure 2.5.1-5 of this report. Also in WLS COL FSAR Section 2.5.1.1.3.2.1, the applicant described three zones of increased seismicity identified in the Charleston area as possible source zones for the 1886 earthquake, specifically the Middleton Place-Summerville, Bowman, and Adams Run seismic zones. Figure 2.5.1-4 of this report shows the locations of these three seismic zones.

The applicant noted that the CEUS SSC model (NUREG-2115) includes three alternative geometries for the Charleston seismic source (i.e., Charleston Local, Charleston Narrow, and Charleston Regional), as discussed in WLS COL FSAR Section 2.5.2.2.4.1. These three alternatives encompass the ten causative source faults and the three seismic source zones postulated for the 1886 Charleston earthquake.

The staff acknowledges that the alternate source zone geometries of the CEUS SSC model (NUREG-2115) contain all potential source faults and postulated seismic source zones for the 1886 Charleston earthquake. Therefore, because the applicant used the CEUS SSC model to analyze seismic hazard at WLS, the staff concludes that the applicant took into account the ten potential source faults and the three postulated seismic source zones.

Charleston Area Seismically-Induced Liquefaction Features

In WLS COL FSAR Section 2.5.1.1.3.2.1, the applicant discussed seismically-induced liquefaction features in the Charleston area, including liquefaction features related to the 1886 Charleston earthquake and paleoliquefaction features found along the South Carolina coast which pre-date the 1886 earthquake. Figure 2.5.1-5 of this report shows the locations of these liquefaction and paleoliquefaction features. The three alternate source zone geometries of the CEUS SSC model encompass the locations of all seismically-induced liquefaction and paleoliquefaction features found in the Charleston area. As discussed in WLS COL FSAR Section 2.5.2.2.4.1, the Charleston Local seismic source geometry specifically encompasses the area with the densest concentration of liquefaction features associated with the 1886 earthquake and prehistoric earthquakes, as well as the meizoseismal area of the 1886 earthquake and the majority of local tectonic features. The Charleston Narrow geometry captures location and orientation of postulated faults and tectonic features in the Charleston area. The Charleston Regional seismic source geometry includes the Local and Narrow zones, along with outlying paleoliquefaction sites and other proposed tectonic features. It was not necessary for the applicant to define a specific causative tectonic structure for any of the liquefaction or paleoliquefaction features because the three alternate source zones used in the PSHA analysis for WLS captured these seismically-induced features.

The staff acknowledges that the CEUS SSC model used by the applicant to assess seismic hazard at WLS incorporates the most recent information on earthquake recurrence interval derived from analysis of the liquefaction and paleoliquefaction features found in the Charleston area, as discussed by the applicant in WLS COL FSAR Section 2.5.1.1.3.2.1. Therefore, because the applicant used the CEUS SSC model to analyze seismic hazard at WLS, the staff concludes that the applicant took into account liquefaction data related to the 1886 event, as well as paleoliquefaction data that suggest three to five possible prehistoric seismic events going back to 5500 years before present.

2.5.1.4.1.3.2 Eastern Tennessee Seismic Zone

The applicant addressed the ETSZ in WLS COL FSAR Section 2.5.1.1.3.2.2 and also discussed the geologic investigations conducted in the zone in WLS COL FSAR Section 2.5.2.2.5.1. Figure 2.5.1-4 of this report shows the location of the ETSZ, which is not a RLME source as defined in the CEUS SSC model (NUREG-2115) but rather lies in the PEZ seismotectonic zone of that model. The applicant concluded that no new information requires treating the ETSZ as a RLME, including the data discussed by Hatcher and others who interpreted initial field evidence to suggest that one or more pre-historic earthquakes of M 6.5 may have occurred in the zone within the last 73,000 to 200,000 years.

Based on review of WLS COL FSAR Sections 2.5.1.1.3.2.2 and 2.5.2.2.5.1 and an independent examination of references cited by the applicant in those WLS COL FSAR sections, the staff concludes that the applicant properly included the ETSZ within the PEZ seismotectonic zone as defined in the CEUS SSC model (NUREG-2115). The staff makes this conclusion because there is no firm evidence as yet that the ETSZ should be included as a RLME for assessment of seismic hazard at WLS.

Based on review of WLS COL FSAR Sections 2.5.1.1.3.2.2 and 2.5.2.2.5.1 and an independent examination of references cited by the applicant in those WLS COL FSAR sections, the staff finds that the applicant provided a thorough and accurate description of the ETSZ in support of the WLS application.

2.5.1.4.1.3.3 Giles County Seismic Zone

In WLS COL FSAR Section 2.5.1.1.3.2.3, the applicant discussed the GCVSZ, reporting that no geologic evidence exists to demonstrate the occurrence of prehistoric earthquakes larger than any historical events known to have occurred in this seismic zone. Consequently, the applicant concluded that there is no new information requiring the zone to be treated in a manner other than incorporating it into the PEZ seismotectonic zone as defined in the CEUS SSC model (NUREG-2115).

Based on review of WLS COL FSAR Section 2.5.1.1.3.2.3 and an independent examination of references cited by the applicant in that WLS COL FSAR section, the staff concludes that the applicant correctly determined that no new data exist to warrant consideration of the GCVSZ differently than including the zone within the PEZ seismotectonic zone as defined for the CEUS SSC model (NUREG-2115). The staff draws this conclusion because no new information exists that requires the seismic zone to be characterized differently.

Based on review of WLS COL FSAR Section 2.5.1.1.3.2.3 and an independent examination of references cited by the applicant in that WLS COL FSAR section, the staff finds that the applicant provided a thorough and accurate description of the Giles County seismic zone in support of the WLS application.

2.5.1.4.1.3.4 Areas of Concentrated Seismicity Outside the Site Region

In WLS COL FSAR Section 2.5.1.1.3.2.4, the applicant discussed two areas of concentrated seismicity outside WLS region, namely, the NMSZ and CVSZ. Figure 2.5.1-4 of this report shows the locations of these two seismic zones. The applicant also discussed detailed information related to characterization of the NMSZ as a RLME in the CEUS SSC model (NUREG-2115) in WLS COL FSAR Section 2.5.2.2.4.2, and investigations of the 2011 Mineral, VA earthquake that occurred in the CVSZ in WLS COL FSAR Section 2.5.2.2.5.2.

For the NMSZ, the applicant presented data in WLS COL FSAR Sections 2.5.1.1.3.2.4 and 2.5.2.2.4.2 documenting that this seismic zone can be characterized as a RLME for assessment of seismic hazard at WLS. Based on a review of WLS COL FSAR Sections 2.5.1.1.3.2.4 and 2.5.2.2.4.2 and an independent examination of references cited by the applicant in those WLS COL FSAR sections, the staff concludes that sufficient data exist to characterize the NMSZ as a RLME. The staff makes this conclusion because there are considerable data presented in the CEUS SSC model (NUREG-2115) to support it.

For the CVSZ, the applicant presented data in WLS COL FSAR Sections 2.5.1.1.3.2.4 and 2.5.2.2.5.2 documenting that the ECC-AM seismotectonic zone of the CEUS SSC model (NUREG-2115) captures seismicity in this zone. The applicant concluded that no new data, including information related to the 2011 Mineral, VA earthquake, required revision to the CEUS SSC model for the CVSZ. Based on review of WLS COL FSAR Sections 2.5.1.1.3.2.4 and 2.5.2.2.5.2 and an independent examination of references cited by the applicant in those WLS COL FSAR sections, the staff concludes that there is no new information requiring revision to how the CEUS SSC model incorporates the CVSZ. The staff draws this conclusion because no evidence exists that the CVSZ represents a RLME.

Based on review of WLS COL FSAR Sections 2.5.1.1.3.2.4, 2.5.2.2.4.2, and 2.5.2.2.5.2 and an independent examination of references cited by the applicant in those WLS COL FSAR sections, the staff finds that the applicant provided a thorough and accurate description of areas of concentrated seismicity outside the site region in support of the WLS application.

2.5.1.4.1.4 *Staff Conclusions on Regional Tectonic Setting, Seismicity, and Paleoseismology*

Based on review of WLS COL FSAR Sections 2.5.1.1.2 and 2.5.1.1.3 and the applicant's responses to RAIs on those two WLS COL FSAR Sections, as well as independent examination of references cited by the applicant in those two sections, the staff finds that the applicant provided thorough and accurate descriptions of the regional tectonic setting and regional seismicity and paleoseismology for WLS, including regional geologic history, tectonic stress in the mid-continent region, gravity and magnetic data in the site region and site vicinity, principal regional tectonic structures, and seismic sources defined by regional seismicity in the CEUS both inside and outside the site region. The staff also finds that the descriptions provided in WLS COL FSAR Sections 2.5.1.1.2 and 2.5.1.1.3 reflect the current literature and state of knowledge and meet the requirements of 10 CFR 52.79 and 10 CFR 100.23.

2.5.1.4.2 Site Geology

The staff focused the review of WLS COL FSAR Section 2.5.1.2 on descriptions provided by the applicant for physiography and geomorphology, geologic setting and history, stratigraphy and lithology, structural geology, engineering geology, seismicity and paleoseismology, and ground water conditions in the WLS vicinity and site area. The staff also focused on descriptions provided by the applicant in WLS COL FSAR Section 2.5.1.2.5 related to certain of these topics specifically for WLS location.

2.5.1.4.2.1 *Site Area Physiography and Geomorphology*

WLS COL FSAR Section 2.5.1.2.1 discusses physiography and geomorphology of the WLS vicinity and site area. The applicant described five lineaments that occur in the site vicinity, one of which, Lineament 1, extends into the site area. The applicant interpreted these lineaments to be non-tectonic in origin, with development controlled by stream drainage patterns and varying resistance of bedrock units to erosion, and concluded that the lineaments do not represent capable tectonic structures. However, Lineament 1 parallels a northeast-striking ridge held up by a resistant quartzite unit, and the orientation of this ridge parallels the northeastern orientation of regional geologic structures that are characteristic of the Appalachian orogenic belt.

The staff focused the review of WLS COL FSAR Section 2.5.1.2.1 on determining that the lineaments occurring in the site vicinity and site area do not represent capable tectonic structures. In RAI 59, Question 02.05.01-29, the staff requested that the applicant discuss the possibility that certain lineaments may reflect regional joint trends, if not capable tectonic features. In RAI 59, Question 02.05.01-31, the staff requested that the applicant discuss the influence of geologic features on development of Lineament 1. In response to RAI 59, Question 02.05.01-29, the applicant stated that, while some exceptions exist, drainage orientations commonly exhibit local control due to variable resistance of bedrock units to erosion, rather than control by jointing, and that no field evidence exists to suggest the five lineaments described in WLS COL FSAR Section 2.5.1.2.1 have a tectonic origin. The applicant compared orientations of regional joint patterns with drainage systems, and determined that joints and structural fabrics strike N25E to N60E, while drainages have a predominant strike of N30W to N60W. In response to RAI 59, Question 02.05.01-31, based on field observations made for the WLS application and information provided in the WLS COL PSAR for the CNS site, the applicant documented that Lineament 1 parallels an erosion-resistant quartzite ridge and terminates to the northeast along strike at the Broad River. Therefore, the applicant concluded that none of these lineaments represent capable tectonic structures.

Based on review of WLS COL FSAR Section 2.5.1.2.1 and the responses to RAI 59, Questions 02.05.01-29 and 02.05.01-31, staff examination of quartzite ridges in the field during the January 2009 site visit, and an independent review of information presented in the CNS PSAR, the staff concludes that erosion-resistant rock units, and not capable tectonic structures, controlled development of the five lineaments discussed in WLS COL FSAR Section 2.5.1.2.1, including Lineament 1, which occurs in the site area. The staff makes this conclusion because field data reported by the applicant and field observations made by the staff during the January 2009 site visit support the interpretation that the lineaments developed along stream drainages aligned parallel to ridges capped by erosion-resistance rock units, and that Lineament 1 is a relatively local feature which terminates along strike at the Broad River. Accordingly, the staff considers RAI 59, Questions 02.05.01-29 and 02.05.01-31 resolved.

Based on review of WLS COL FSAR Section 2.5.1.2.1, the responses to RAI 59, Questions 02.05.01-29 and 02.05.01-31, examination of quartzite ridges in the field, and an independent review of information presented in the CNS PSAR, the staff finds that the applicant provided a thorough and accurate description of site area physiography and geomorphology in support of the WLS application.

2.5.1.4.2.2 *Site Area Geologic Setting and History*

WLS COL FSAR Section 2.5.1.2.2 describes geologic setting and history of the WLS vicinity and site area, specifically addressing the Charlotte terrane in which the site lies and summarizing the deformation and metamorphic history of the site area. However, WLS COL FSAR Section 2.5.1.2.2 presents an inconsistent terminology for the rock units that comprise the Charlotte terrane, making it difficult to correlate rock units described in WLS COL FSAR Section 2.5.1.2.2 with those shown on the geologic maps included in the WLS COL FSAR. In addition, WLS COL FSAR Section 2.5.1.2.2 does not concisely summarize the complete deformation history of the site area, but leaves out Ordovician (488 to 444 Ma) history and any discussion of Mesozoic (251 to 145.5 Ma) cataclasites (i.e., rocks which developed as a result of pervasive fracturing, crushing, and grinding by brittle faulting).

The staff focused the review of WLS COL FSAR Section 2.5.1.2.2 on clarifying both terminology for the rock units comprising the Charlotte terrane in which the site occurs and deformation history of the Charlotte terrane in the site area. In RAI 59, Question 02.05.01-32, the staff requested that the applicant delineate the lithologic units that make up the Charlotte terrane to enable correlation with geologic maps presented in the WLS COL FSAR. In RAI 59, Question 02.05.01-35, the staff requested that the applicant summarize the complete deformation history of the site area to aid with assessing potential geologic hazard at WLS. In the responses to RAI 59, Questions 02.05.01-32 and 02.05.01-35, the applicant indicated that the confusion about site area lithologies results largely from the fact that different researchers have applied different classification schemes and terminologies over time. The applicant provided changes to WLS COL FSAR Section 2.5.1.2.2 that clarified terminology for the rock units occurring in the site area and more completely summarized site area deformation history to include all orogenic events proposed for the Charlotte terrane based on information in Hibbard and others.

Based on review of WLS COL FSAR Section 2.5.1.2.2, the responses to RAI 59, Questions 02.05.01-32 and 02.05.01-35, and an independent examination of the clarifying references cited by the applicant, the staff concludes that the applicant clearly delineated the lithologic units that make up the Charlotte terrane and adequately discussed the deformation history of that terrane. The staff makes this conclusion because the applicant incorporated additional information from appropriate published sources prepared by area experts. Accordingly, the staff considers RAI 59, Questions 02.05.01-32 and 02.05.01-35 resolved.

Based on review of WLS COL FSAR Section 2.5.1.2.2, the responses to RAI 59, Questions 02.05.01-32 and 02.05.01-35, and an independent examination of the references cited by the applicant, the staff finds that the applicant provided a thorough and accurate description of site area geologic history and setting in support of WL application**.**

2.5.1.4.2.3 *Site Area Stratigraphy and Lithology*

WLS COL FSAR Section 2.5.1.2.3 describes site area stratigraphy and lithology. The applicant indicated that rock units in WLS area are part of the Battleground Formation, defined by Horton as Neoproterozoic (1000 to 542 Ma) in age. A Neoproterozoic metamorphosed plutonic rock mass, Zto, intrudes the metavolcanic sequence and comprises the foundation rock unit for WLS Units 1 and 2.

The staff focused the review of WLS COL FSAR Section 2.5.1.2.3 on understanding the extent of foundation rock unit Zto and the relationships between the various rock units that occur in the site area and at the site location. In RAI 59, Question 02.05.01-36, the staff requested that the applicant provide a consistent nomenclature and description for rock unit Zto since this is the foundation rock mass for WLS Units 1 and 2. In RAI 59, Question 02.05.01-37, the staff requested that the applicant discuss in more detail the relationships between rock units found in the site area and at the site location, to include specifying locations of the geologic structures interpreted to deform, and consequently influence the observed relationships between, rock units in the site area and at the site location. In response to RAI 59, Question 02.05.01-36, the applicant stated that variations in descriptive terminology for the Battleground Formation and rock unit Zto occur because different researchers mapped the Battleground Formation and Zto at different times using different map scales in adjacent quadrangles. The applicant incorporated changes into WLS COL FSAR Section 2.5.1.2.3 to more concisely describe the Battleground Formation and foundation rock unit Zto, including multiple references clarifying the variations in nomenclature for Zto. The applicant stated that, while the composition of Zto is spatially variable, the most abundant rock type is metagranodiorite based on petrographic analyses presented in the CNS PSAR. In response to RAI 59, Question 02.05.01-37, the applicant further modified WLS COL FSAR Section 2.5.1.2.3 by locating the South Fork and McKowns Creek antiforms on the site area geologic map shown in WLS COL FSAR Figure 2.5.1-219a to accompany the text changes provided in response to RAI 59, Question 02.05.01-36. These text changes clarified the relationships between rock units in the site area and at the site location as influenced by the intense deformation which accompanied development of the antiformal folds, leading to the interpretation that rock units are relatively younger to the northwest across the site area.

Based on review of WLS COL FSAR Section 2.5.1.2.3, the responses to RAI 59, Questions 02.05.01-36 and 02.05.01-37, and an independent examination of references cited by the applicant related to rock nomenclature and the relationships between rock units in the site area and at the site location, the staff concludes that the applicant clarified the nomenclature for and description of foundation rock unit Zto and qualified the relationships between the different rock units in the site area and at the site location. The staff makes this conclusion because the applicant provided information documenting the primary rock type that characterizes rock unit Zto and the effects of deformation on the relationships between rock units in the site area and at the site location. Accordingly, the staff considers RAI 59, Questions 02.05.01-36 and 02.05.011-37 resolved.

Based on review of WLS COL FSAR Section 2.5.1.2.3, the responses to RAI 59, Questions 02.05.01-36 and 02.05.01-37, and an independent examination of references cited by the applicant related to rock nomenclature and relationships between rock units in the site area and at the site location, the staff finds that the applicant provided a thorough and accurate description of site area stratigraphy and lithology in support of the WLS application.

2.5.1.4.2.4 *Site Area Structural Geology*

WLS COL FSAR Section 2.5.1.2.4 describes structural geology of the site area, including deformation history. WLS COL FSAR Section 2.5.1.2.4.1 discusses the relevant geologic structures that occur within WLS area, specifically the Cherokee Falls and Draytonville synforms and slickensides.

The staff focused the review of WLS COL FSAR Sections 2.5.1.2.4 and 2.5.1.2.4.1 on the applicant's discussions of the Cherokee Falls and Draytonville synforms and slickensides. Sections 2.5.1.4.2.4.1 through 2.5.1.4.2.4.3 of this report present the staff's technical evaluation of the information provided by the applicant in WLS COL FSAR Sections 2.5.1.2.4 and 2.5.1.2.4.1 related to site area structural geology and structures within the site area.

2.5.1.4.2.4.1 Deformation History

In WLS COL FSAR Section 2.5.1.2.4, based on Schaeffer, the applicant discussed the field evidence for five regional deformation episodes, D1 through D5, which occurred during development of the Appalachian orogenic belt and affected WLS area as well as the site region, site vicinity, and site location. The applicant presented field evidence documenting that D1 and D2 structures formed during deformation accompanied by a metamorphic event around 542 Ma ago, and radiometric age dates constraining the timing of D3, D4 and D5 structures to be more than 296 Ma old.

Based on review of the information presented in WLS COL FSAR Section 2.5.1.2.4 regarding ages of deformation episodes D1 through D5, independent review of the references cited by the applicant in that WLS COL FSAR section, four data documentation audits conducted by the staff in February 2009, July 2009, June 2011, and October 2011, and the staff's direct examination of deformation features during four site visits conducted in April-May 2008, January 2009, July

2011, and February 2014, the staff concludes that deformation events D1 through D5 are all older than 296 Ma. The staff draws this conclusion because field evidence related to growth of metamorphic minerals associated with the deformation events supports the interpretation that none of these events are younger than 296 Ma.

2.5.1.4.2.4.2 Cherokee Falls and Draytonville Synforms

In WLS COL FSAR Section 2.5.1.2.4.1, the applicant presented field evidence documenting that the Cherokee Falls and Draytonville synforms are D2 deformation features. The applicant also indicated that more recent mapping by Nystrom did not include either of these synforms, although the portion of the geologic map shown in WLS COL FSAR Figure 2.5.1-219a as being based on Nystrom includes the synforms. In RAI 59, Questions 02.05.01-40 and 02.05.01-41, the staff requested that the applicant clarify why these two structures are shown on the map derived from Nystrom even though he did not include them on his map. In response to RAI 59, Questions 02.05.01-40 and 02.05.01-41, the applicant stated that the map prepared by Nystrom did not include structural data, including fold axes, and the portion of the geologic map in WLS COL FSAR Figure 2.5.1-219a attributed to Nystrom actually presents geologic data derived from multiple sources, certain of whom mapped the axial traces of the Cherokee Falls and Draytonville synforms. The applicant revised notation on WLS COL FSAR Figure 2.5.1-219a to indicate that geology came from Nystrom, Howard, and Horton and Dicken; that Hibbard and others provided information on fault locations; and that Murphy and Butler provided the locations of fold axes.

Based on review of WLS COL FSAR Section 2.5.1.2.4.1, including the modifications to WLS COL FSAR Figure 2.5.1-219 and the responses to RAI 59, Questions 02.05.01-40 and 02.05.01-41, the staff concludes that the applicant clarified the basis for including the axial traces of the Cherokee Falls and Draytonville synforms on WLS COL FSAR Figure 2.5.1-219a. The staff draws this conclusion because the applicant documented the various sources of information compiled to develop this figure, including the mapping of Murphy and Butler that defined these two structures, and the staff reviewed the cited information and finds it reliable and sufficient. Accordingly, the staff considers RAI 59, Questions 02.05.01-40 and 02.05.01-41 resolved.

2.5.1.4.2.4.3 Slickensides

Slickensides are potentially important structural features because they provide evidence for at least minor displacement along the surfaces on which they occur. Based on the CNS PSAR, the applicant stated that, although slickensided surfaces occur in the site area, none could be readily traced beyond a single exposure, which suggests the lack of a through-going fault. The applicant also reported that similar minor, locally-developed slickensided surfaces are common throughout the Piedmont. Since the applicant did not show the slickenside locations on a geologic map in the WLS COL FSAR, it is uncertain whether they may define a single extended linear trace. It is also uncertain whether unsheared minerals occur on all slickensided surfaces, indicating that the slickensides represent old and deep-seated, rather than young (i.e., Quaternary, 2.6 Ma to present in age) and near-surface, deformation. Therefore, to enable an assessment of whether the slickensided surfaces may indicate the presence of a capable tectonic structure, in RAI 59, Question 02.05.01-42, the staff requested that the applicant summarize information taken from the CNS PSAR and used to conclude that the slickensides are old and do not indicate the existence of a potentially capable Quaternary tectonic structure of finite length in the site area. The staff also requested that the applicant provide a reference for the statement that slickensided surfaces and other similar minor, locally developed features with no tectonic significance occur throughout the Piedmont. In response to RAI 59, Question 02.05.01-42, the applicant cited the CNS PSAR and stated that no features associated with the locations of the slickensides exhibited any topographic expression, as may be expected for an active tectonic structure, and that orientation of the slip surfaces varied at the different locations, making them inconsistent with a single tectonic feature. The applicant also noted that the mineral epidote occurred on at least one of slickensided surfaces, and deformation history of the Piedmont, including the site area, indicates the conditions required to grow this mineral have not occurred in the site area since late Paleozoic time (> 251 Ma). The applicant cited Garihan and others, who reported the widespread occurrence of brittle deformation features that commonly show no correlation with active faults throughout the Piedmont.

Based on review of WLS COL FSAR Section 2.5.1.2.4.1, the response to RAI 59, Question 02.05.01-42, and an independent examination of references cited by the applicant, the staff concludes that the slickensides found in the site area do not suggest the presence of any capable tectonic structures. The staff makes this conclusion because the applicant documented the local, discontinuous nature of the slickensided surfaces. Accordingly, the staff considers RAI 59, Question 02.05.01-42 resolved.

Based on review of WLS COL FSAR Sections 2.5.1.2.4 and 2.5.1.2.4.1 and the responses to RAI 59, Questions 02.05.01-40 through 02.05.01-42, the staff finds that the applicant provided a thorough and accurate description of site area structural geology, including deformation history and structures within the site area, in support of the WLS application.

2.5.1.4.2.5 *Site Location Geology*

WLS COL FSAR Section 2.5.1.2.5 discusses geology of the WLS location, including physiography and geomorphology, geologic setting and history, stratigraphy and lithology, structure, and geologic mapping. Sections 2.5.1.4.2.5.1 through 2.5.1.4.2.5.5 of this report present the staff's technical evaluation of the information provided by the applicant in WLS COL FSAR Section 2.5.1.2.5. In Section 2.5.1.4.2.5.5 of this report, the staff also evaluated the compiled geologic map and associated data for the CNS Unit 1 site (i.e., proposed WLS Unit 1) excavation, which documented lithologies and the absence of capable tectonic structures or other potentially detrimental geologic features in the foundation rocks of proposed WLS Unit 1. The staff focused the review of WLS COL FSAR Section 2.5.1.2.5 on understanding timing of deformation at the site location in light of the information presented by the applicant in WLS COL FSAR Section 2.5.1.2.4 related to the five regional deformation episodes that affected the site area, and on assessment of the compiled geologic map of the CNS/WLS Unit 1 excavation.

2.5.1.4.2.5.1 Site Location Physiography and Geomorphology

In WLS COL FSAR Section 2.5.1.2.5.1, the applicant discussed physiography and geomorphology of WLS location. The applicant reported that physiography and geomorphology at the site location are similar to these same characteristics as observed in the site area.

Based on review of the information presented in WLS COL FSAR Section 2.5.1.2.5.1 regarding physiography and geomorphology of WLS location, as well as field observations made during site visits conducted by the staff in April-May 2008, January 2009, July 2011, and February 2014, the staff concludes that physiographic and geomorphic characteristics of the site location are similar to these same characteristics as described for the site area. The staff draws this conclusion because field observations made during the four site visits confirm that physiography and geomorphology at the site location are typical of the site area and no unique physiographic or geomorphic features occur at the site location.

Based on review of WLS COL FSAR Section 2.5.1.2.5.1 and the field observations made during the four site visits, the staff finds that the applicant provided a thorough and accurate description of site location physiography and geomorphology in support of the WLS application.

2.5.1.4.2.5.2 Site Location Geologic Setting and History

In WLS COL FSAR Section 2.5.1.2.5.2, the applicant discussed geologic setting and history of the site location. The applicant stated that geologic setting and history of the site location are the same as the geologic setting and history of the site area even though, of the five regional deformation episodes (D1 through D5) documented in the site area, rock units at the site location most strongly exhibit only the first two of the five events (i.e., D1 and D2, with D2 being the most prominent).

Based on review of WLS COL FSAR Section 2.5.1.2.5.2 and field observations made during site visits conducted by the staff in April-May 2008, January 2009, July 2011, and February 2014, the staff concludes that the metavolcanic and metasedimentary country rock sequence responded differently to deformation than did foundation rock unit Zto due to rheological differences between these different lithologies. The staff makes this conclusion because field evidence observed during the site visits listed above documents the imprint of all five deformation events, D1 through D5, in the country rock sequence and corroborates the applicant’s statement that rock unit Zto best exhibits deformation events D1 and D2, with D2 being the most obvious.

Based on review of WLS COL FSAR Section 2.5.1.2.5.2 and the field observations made during the four site visits, the staff finds that the applicant provided a thorough and accurate description of site location geologic setting and history in support of the WLS application.

2.5.1.4.2.5.3 Site Location Stratigraphy and Lithology

In WLS COL FSAR Section 2.5.1.2.5.3, the applicant discussed stratigraphy and lithology of the site location. The applicant indicated that stratigraphy and lithology at the site location are the same as stratigraphy and rock types in the site area.

Based on review of WLS COL FSAR Section 2.5.1.2.5.3 and field observations made during site visits conducted by the staff in April-May 2008, January 2009, July 2011, and February 2014, the staff concludes that stratigraphy and lithology at the site location are congruent with site area stratigraphy and lithology. The staff makes this conclusion because field evidence observed by the staff documents that stratigraphy and lithology at the site location are the same as the stratigraphy and rock types described for the site area.

Based on review of WLS COL FSAR Section 2.5.1.2.5.3 and the field observations made during the four site visits, the staff finds that the applicant provided a thorough and accurate description of site location stratigraphy and lithology in support of the WLS application.

2.5.1.4.2.5.4 Site Location Structure

In WLS COL FSAR Section 2.5.1.2.5.4, the applicant addressed structural geology of the site location. The applicant discussed geologic structures that occur at the site location, including the McKown's Creek antiform, shear and breccia zones, joints and dilation fractures, and slickensides.

McKown's Creek Antiform

The applicant reported that, while regional deformation episodes D1 through D5 characterize deformation style in the site area, geologic structures at the site location, including the McKown's Creek antiform, largely reflect geologic structures and rock fabrics resulting from deformation D2. The applicant noted that foundation unit Zto is a massive metamorphosed intrusive igneous body and does not contain good marker horizons for registering deformation events. Therefore, although the applicant provided a chronological listing of regional deformation events that affected the site area in WLS COL FSAR Section 2.5.1.2.5.4, it is not completely clear to the staff which deformation event produced a given geologic structure (e.g., foliation surfaces, shear and breccia zones, dilation fractures and joints, and slickensides) at the site location. In RAI 59, Question 02.05.01-46, the staff requested that the applicant clarify which deformation event produced which geologic structure and rock fabric at the site location. In response to RAI 59, Question 02.05.01-46, the applicant incorporated new Table 2.5.1-204 into WLS COL FSAR Section 2.5.1.2.5.4 to correlate structures developed at the site location with site area deformation episodes, including a summary of age constraints for the deformation events and structures. This table clarified the relationships described by the applicant between site location structures and site area deformation episodes, and equated the latest stage of fracture development with Mesozoic (251 to 145.5 Ma) extension. All other tectonic structures (i.e., folds, shear and breccia zones, dilation fractures, and joints mineralized by chlorite and calcite and often slickensided) are older than 219 Ma based on radiometric age date results.

Shear and Breccia Zones, Dilation Fractures, Joints, and Slickensides

Shear and breccia zones, dilation fractures, joints, and slickensides occur at the site location in rock mass Zto, the foundation unit for proposed WLS Unit 1 and Unit 2. The applicant reported that a K-Ar radiometric age date constrains timing of the shearing, brecciation, dilation fracturing, and development of slickensides on joint surfaces coated with chlorite mica in unweathered rock to be older than 219 Ma.

Based on review of WLS COL FSAR Section 2.5.1.2.5.4, including WLS COL FSAR Table 2.5.1-204, and the response to RAI 59, Question 02.05.01-46, as well as direct field observations made by the staff during site visits in April-May 2008, January 2009, July 2011, and February 2014 and review of information during data documentation audits conducted by NRC staff in February and July 2009 and June and October 2011, the staff concludes that the applicant provided information sufficient to enable correlation of structures developed at the site location with site area deformation episodes. The staff makes this conclusion because information reviewed during the data documentation audits, which included radiometric age dates, and field evidence examined during the site visits documented the deformation history presented by the applicant for the site location and confirmed that none of the structures produced by the deformation episodes represent capable tectonic features. Accordingly, the staff considers RAI 59, Question 02.05.01-46 resolved.

Based on review of WLS COL FSAR Section 2.5.1.2.5.4 and the response to RAI 59, Question 02.05.01-46, as well as field observations made during the four site visits and examination of data during the four data documentation audits, the staff finds that the applicant provided a thorough and accurate description of site location structure in support of the WLS application.

2.5.1.4.2.5.5 Site Location Geologic Mapping

WLS COL FSAR Section 2.5.1.2.5.5 discusses geologic mapping performed at the site location.

Based on geologic mapping results for both WLS and the original CNS site, including detailed geologic maps of CNS Unit 1 foundation grade level bedrock produced by digitizing and compiling the original field maps prepared for CNS Unit 1, as well as in-situ and laboratory test results summarized in WLS COL FSAR Section 2.5.1.2.6 and cross-referenced in WLS COL FSAR Section 2.5.1.2.5.5, the applicant concluded that the foundation rock mass at WLS is suitable. Since the applicant used information in the CNS PSAR to supplement geologic data for WLS, WLS COL FSAR Section 2.5.1.2.5.5 also discusses confirmatory testing of the original CNS geologic map of CNS Unit 2, implemented by the applicant as part of site characterization for WLS. The applicant did not provide a geologic map of that part of the CNS Unit 2 excavation used to confirm the accuracy of previous mapping at the original CNS site. In RAI 59, Question 02.05.01-47, the staff requested that the applicant summarize the data collected during previous geologic mapping at the CNS site being used to supplement geologic data for WLS. The staff also requested that the applicant provide a geologic map documenting the reported confirmation of the previous geologic mapping at the CNS site. Also in WLS COL FSAR Section 2.5.1.2.5.5, the applicant concluded that the western boundary of plutonic rock mass Zto, the foundation unit at WLS, is a lithologic contact with surrounding country rock rather than a fault, although Nystrom mapped this boundary as a straight line segment that could be interpreted as a fault. The applicant did not provide a summary of the pertinent information that made the conclusion about the non-tectonic character of this contact possible. In RAI 59, Question 02.05.01-49, the staff requested that the applicant summarize the data used to determine that the western boundary of foundation unit Zto is not a fault or shear zone in light of the fact that Nystrom mapped this boundary as a straight line that could be interpreted as a fault.

In response to RAI 59, Question 02.05.01-47, the applicant indicated that the primary objective of the confirmatory mapping was to validate the historical final foundation geologic mapping records developed for the original CNS site. The applicant used these records to supplement geologic data collected for the WLS application. The applicant noted that this mapping was particularly important for validating the unpublished historical geologic map data for CNS Unit 1, which now lies under concrete, because the original CNS Unit 1 site coincides with the proposed WLS Unit 1 site. The applicant stated that the most prominent lithologies, shear zones, and individual shears appear on both the historical CNS and WLS geologic maps. The applicant provided a geologic map showing WLS mapping results overlain on the part of the historical CNS Unit 2 final foundation geologic map being validated to support this statement. Careful examination of these two sets of geologic map data by staff documented that the most prominent lithologies, shear zones, and individual shears occur on both the historical CNS Unit 2 and the confirmatory WLS geologic maps.

In response to RAI 59, Question 02.05.01-49, using field evidence derived from trenches located to investigate geologic characteristics of the western boundary of metamorphosed plutonic rock mass Zto and from boreholes drilled for CNS site investigations and the WLS application, the applicant determined that this boundary is lithologic contact between the pluton and the adjacent country rock and not a fault. The applicant modified WLS COL FSAR Figure 2.5.1-226 to show the irregular nature of this boundary and correct the misconception that it is a straight line, which could be interpreted as related to faulting.

Based on review of WLS COL FSAR Section 2.5.1.2.5.5, the responses to RAI 59, Questions 02.05.01-47 and 02.05.01-49, and information in the CNS PSAR, as well as field observations made by the staff during the site visits and examination of data by the staff during the data documentation audits, including review of the report presenting the compiled geologic map and associated data for the WLS/CNS Unit 1 excavation, the staff makes the following conclusions: (1) the most prominent lithologies, shear zones, and individual shears appear in both the historical CNS and WLS geologic maps; (2) the western boundary of foundation rock unit Zto is an irregular intrusive contact rather than a fault; (3) similarities exist in both lithologies and geologic features found in the CNS Unit 1 (proposed WLS Unit 1) excavation and the adjacent excavations for CNS Unit 2 and CNS Unit 3 (proposed WLS Unit 2); and (4) no capable tectonic structures or other potentially detrimental geologic features occur in the excavation for proposed WLS Unit 1. The staff makes these conclusions for the following reasons with field evidence based on observations made by the staff and independent evaluation of data presented by the applicant: (1) field data shown on the historical CNS Unit 2 and the confirmatory WLS geologic maps document that lithologies and major geologic features are similar between the two maps; (2) field evidence indicates the western boundary of foundation unit Zto is not a fault; (3) the data documentation audits and site visits confirm that similarities exist in both lithologies and geologic features found in the CNS Unit 1 (proposed WLS Unit 1) excavation and in the adjacent excavations for CNS Unit 2 and CNS Unit 3 (proposed WL Unit 2); and (4) field evidence indicates that no capable tectonic structures or other potentially detrimental geologic features occur in the excavation for proposed WLS Unit 1. Accordingly, the staff considers RAI 59, Questions 02.05.01-47 and 02.05.01-49 resolved.

Based on review of WLS COL FSAR Section 2.5.1.2.5.5 and the responses to RAI 59, Questions 02.05.01-47 and 02.05.01-49, as well as the field observations made during the site visits and examination of data during data documentation audits, including the report presenting the compiled geologic map for CNS Unit 1/WL Unit 1, the staff finds that the applicant provided a thorough and accurate description of site location geologic mapping in support of the WLS application.

2.5.1.4.2.6 *Site Area Engineering Geology*

In WLS COL FSAR Section 2.5.1.2.6, the applicant described engineering geology of the site area. The applicant indicated that, although further excavation will likely expose weathered zones, joints, and fractures in the foundation rock unit (Zto), the weathered rock would be treated and removed as described in WLS COL FSAR Section 2.5.4.12. Based on in-situ and laboratory test results summarized in WLS COL FSAR Section 2.5.1.2.6, as well as information derived from geologic mapping at the site location, the applicant concluded that the foundation rock unit at WLS is suitable.

Based on review of WLS COL FSAR Section 2.5.1.2.6, in-situ and laboratory results, and results of geologic mapping at the site location, the staff finds that the applicant provided a thorough and accurate description of site area engineering geology in support of the WLS application**.**

2.5.1.4.2.7 *Site Area Seismicity and Paleoseismology*

In WLS COL FSAR Section 2.5.1.2.7, the applicant discussed site area seismicity and paleoseismology. The applicant stated that no published reports identified paleoseismic features in the site area, and that, based on extensive new studies performed for the WLS application, there is no evidence for post-Miocene (< 5.3 Ma) earthquake activity in the site area. However, the applicant did not present any information to support the statement about the new studies.

The staff focused the review of WLS COL FSAR Section 2.5.1.2.7 on the applicant’s statement that no evidence exists for post-Miocene (< 5.3 Ma) earthquake activity in the site area. In RAI 59, Question 02.05.01-50, the staff requested that the applicant provide information documenting that field investigations performed for the WLS application did not reveal any evidence for post-Miocene earthquake activity. In response to RAI 59, Question 02.05.01-50, the applicant stated that the new studies included comprehensive literature reviews and interviews with site area experts; interpretation of aerial photographs; and geologic investigations comprising aerial and ground reconnaissance of the site area, including examination of Quaternary alluvial surfaces and deposits that are potentially subject to earthquake-induced liquefaction and outcrops along the Broad River and its tributaries. The applicant stated that none of these efforts revealed paleoliquefaction features, other indicators of post-Miocene earthquake activity, or evidence for Miocene or younger faulting in the site area.

Based on review of WLS COL FSAR Section 2.5.1.2.7 and the applicant’s response to RAI 59, Question 02.05.01-50, the staff concludes that no evidence exists for earthquake-induced paleoliquefaction, other indicators of post-Miocene earthquake activity, or Miocene or younger faulting. The staff makes this conclusion because the applicant applied the standard field methods necessary for assessing the presence of paleoseismic features such as those that would be represented by earthquake-induced liquefaction or young (i.e., post-Miocene, including Quaternary age) faulting.

Based on review of WLS COL FSAR Section 2.5.1.2.7 and the fact that the applicant applied standard field methods to assess the presence of paleoseismic features, the staff finds that the applicant provided a thorough and accurate description of site area seismicity and paleoseismology in support of the WLS application.

2.5.1.4.2.8 *Site Area Groundwater Conditions*

WLS COL FSAR Section 2.5.1.2.8 cross-references WLS COL FSAR Section 2.4.12, for the detailed discussion of groundwater conditions presented by the applicant for WLS.

2.5.1.5 *Post Combined License Activities*

There are no post COL activities related to WLS COL FSAR Section 2.5.1. However, Section 2.5.3.4.8 of this report identifies a geologic mapping License Condition for WLS Unit 2 as the responsibility of the applicant and explains why a geologic mapping License Condition is not necessary for WLS Unit 1. Section 2.5.3.5 of this report defines the applicant's responsibility for geologic mapping at WLS Unit 2 as License Condition (2-2).

2.5.1.6 *Conclusion*

The staff reviewed the WLS COL FSAR and the referenced DCD. Based on these reviews, the staff confirmed that the applicant addressed the required information related to basic geologic and seismic characteristics, and that no additional outstanding information must be discussed in WLS COL FSAR related to these characteristics. NUREG-1793 and its supplements document the results of the staff's technical evaluation of the information incorporated by reference into the WLS application.

As set forth above, the staff has reviewed the information in WLS 2.5-1 and finds that the applicant provided a thorough characterization of basic geologic and seismic information for WLS, as required by 10 CFR 100.23 and 10 CFR 52.79(a)(1)(iii). In addition, the staff concludes that the applicant identified and appropriately characterized all seismic sources significant for determining the GMRS, or SSE, for this COL site, in accordance with NRC regulations provided in 10 CFR 100.23 and 10 CFR 52.79(a)(1)(iii) and guidance provided in RG 1.208. Based on the applicant's geologic investigations performed for the site region, site vicinity, site area, and site location, the staff concludes that the applicant properly characterized regional and site lithology, stratigraphy, geologic and tectonic history, and structural geology, as well as subsurface soil materials and rock units at WLS. The staff also concludes that there is no potential for effects of human activity (i.e., mining activity or ground water injection or withdrawal) to compromise site safety. Therefore, the staff concludes that the proposed WLS site is acceptable from the standpoint of geologic and seismic information and meets the requirements of 10 CFR 100.23 and 10 CFR 52.79(a)(1)(iii).

WLS COL 2.5-1 addresses the provision of regional and site-specific geologic, seismic, and geophysical information, as well as conditions caused by human activity. Based on the discussion of the basic geologic and seismic information presented in WLS COL FSAR Section 2.5.1, and the technical evaluation presented above in FSER Section 2.5.1.4, the staff concludes that the applicant provided the information required to satisfy WLS COL 2.5-1.

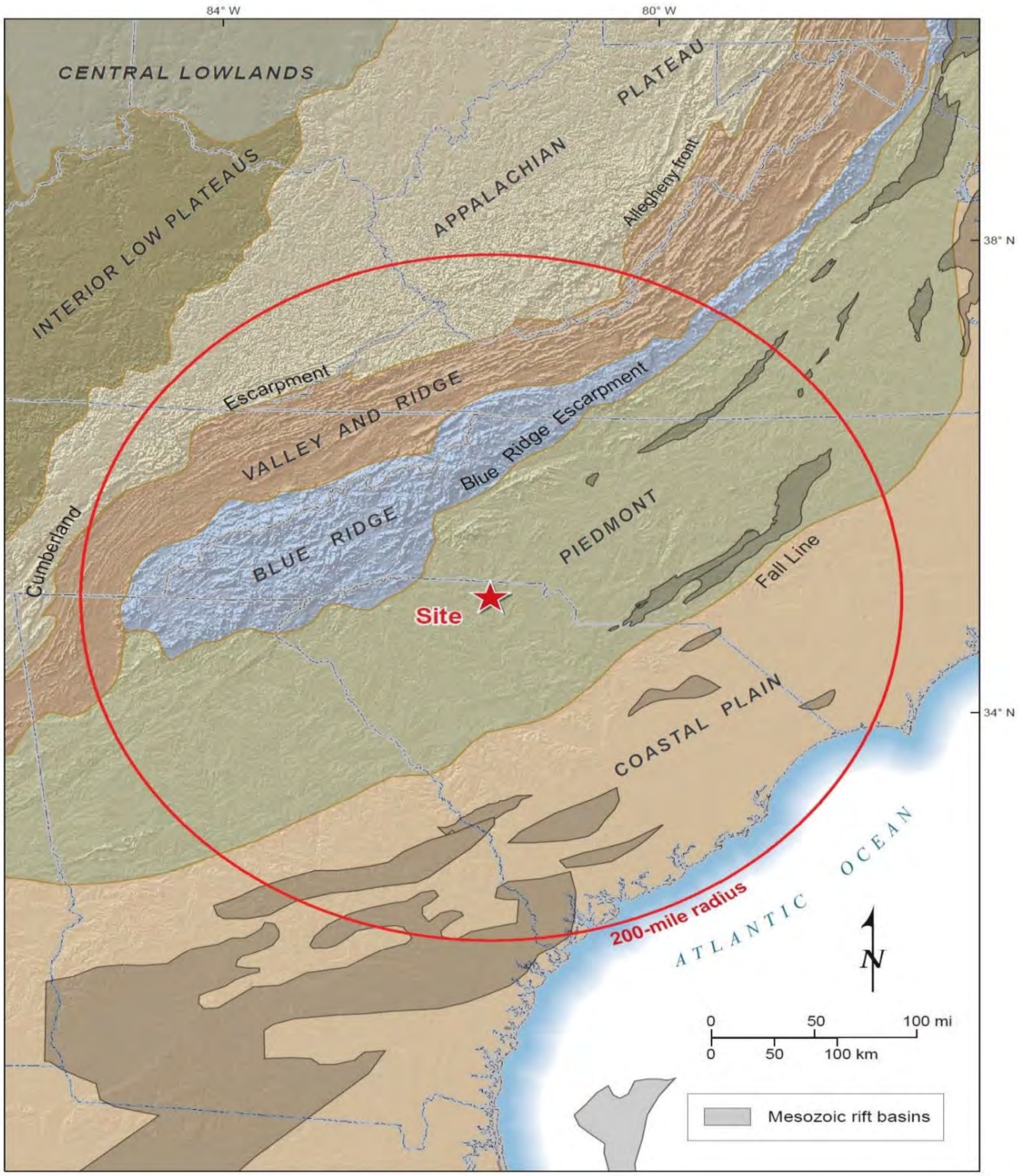


Figure 2.5.1-1 Location of WLS in relation to physiographic provinces and Mesozoic rift basins that occur in the site region (Reproduced from WLS COL FSAR Revision 0, Figure 2.5.1-201)

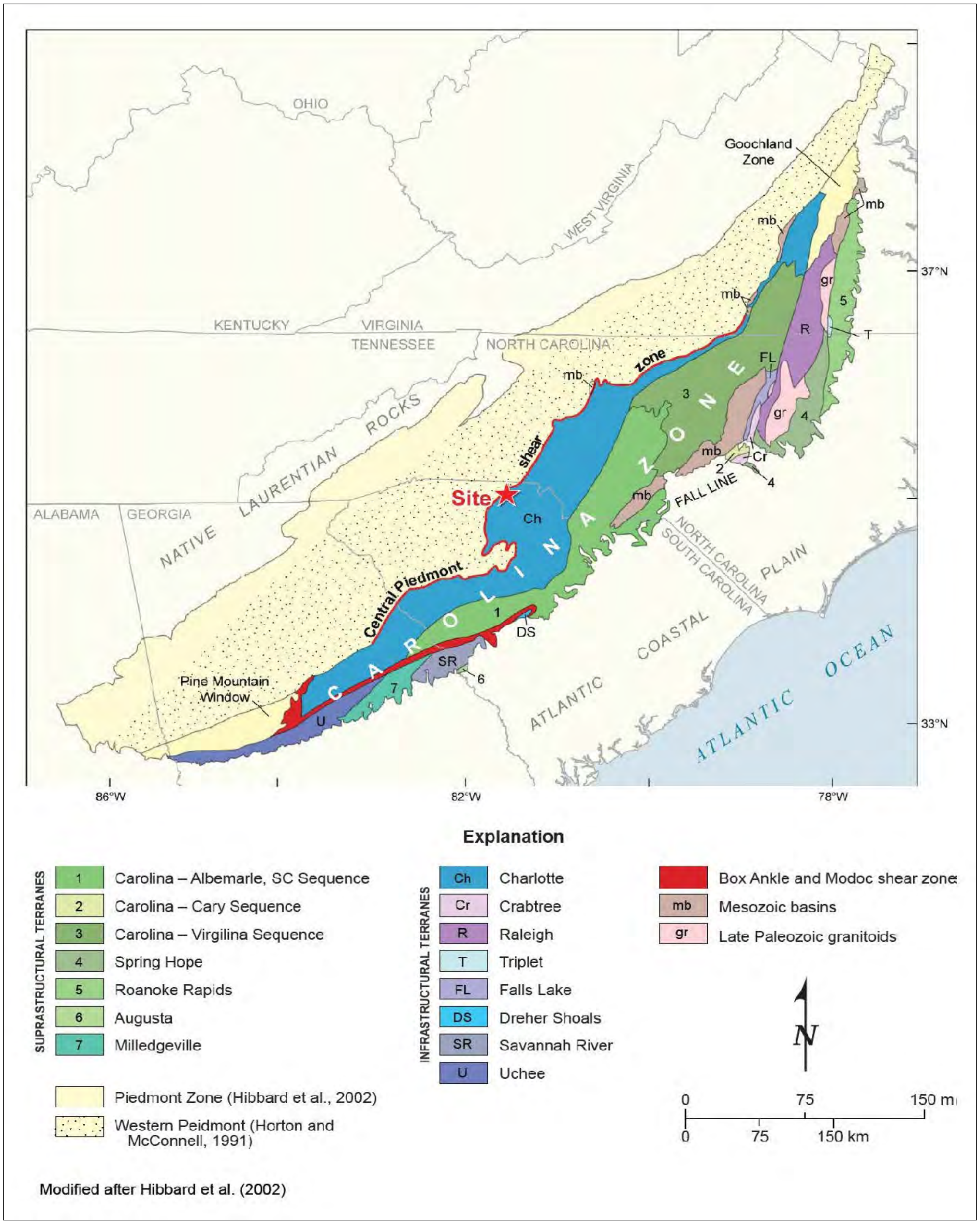


Figure 2.5.1-2 Location of WLS in Relation to The Charlotte Terrane of The Carolina Zone (Reproduced from WLS COL FSAR Revision 2, Figure 2.5.1-202a)

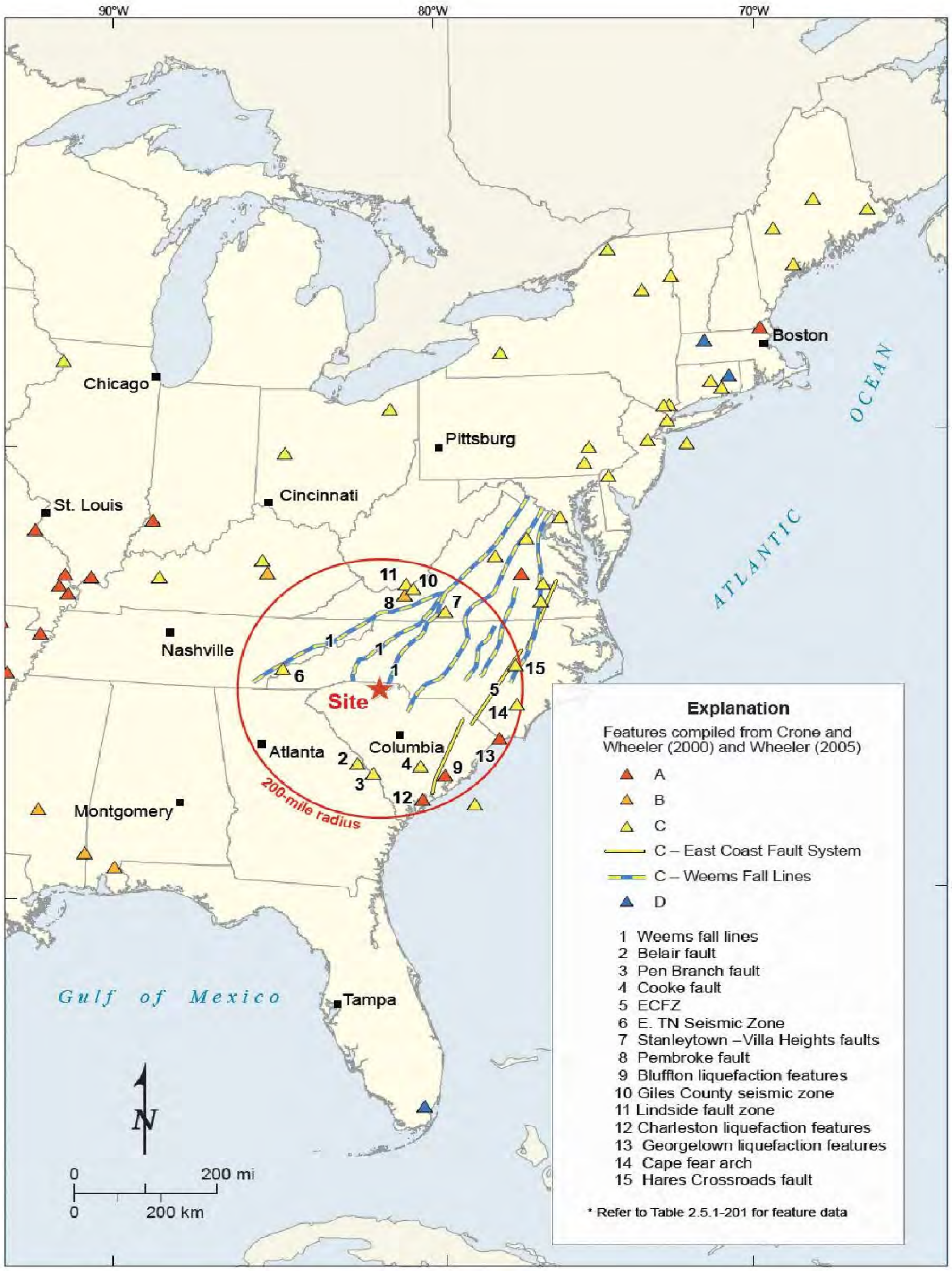


Figure 2.5.1-3 Location of WLS in relation to potential Quatrnary age (2.6 Ma to Present) tectonic features in the site region (Reproduced from WLS COL FSAR Revision 0, Figure 2.5.1-213)

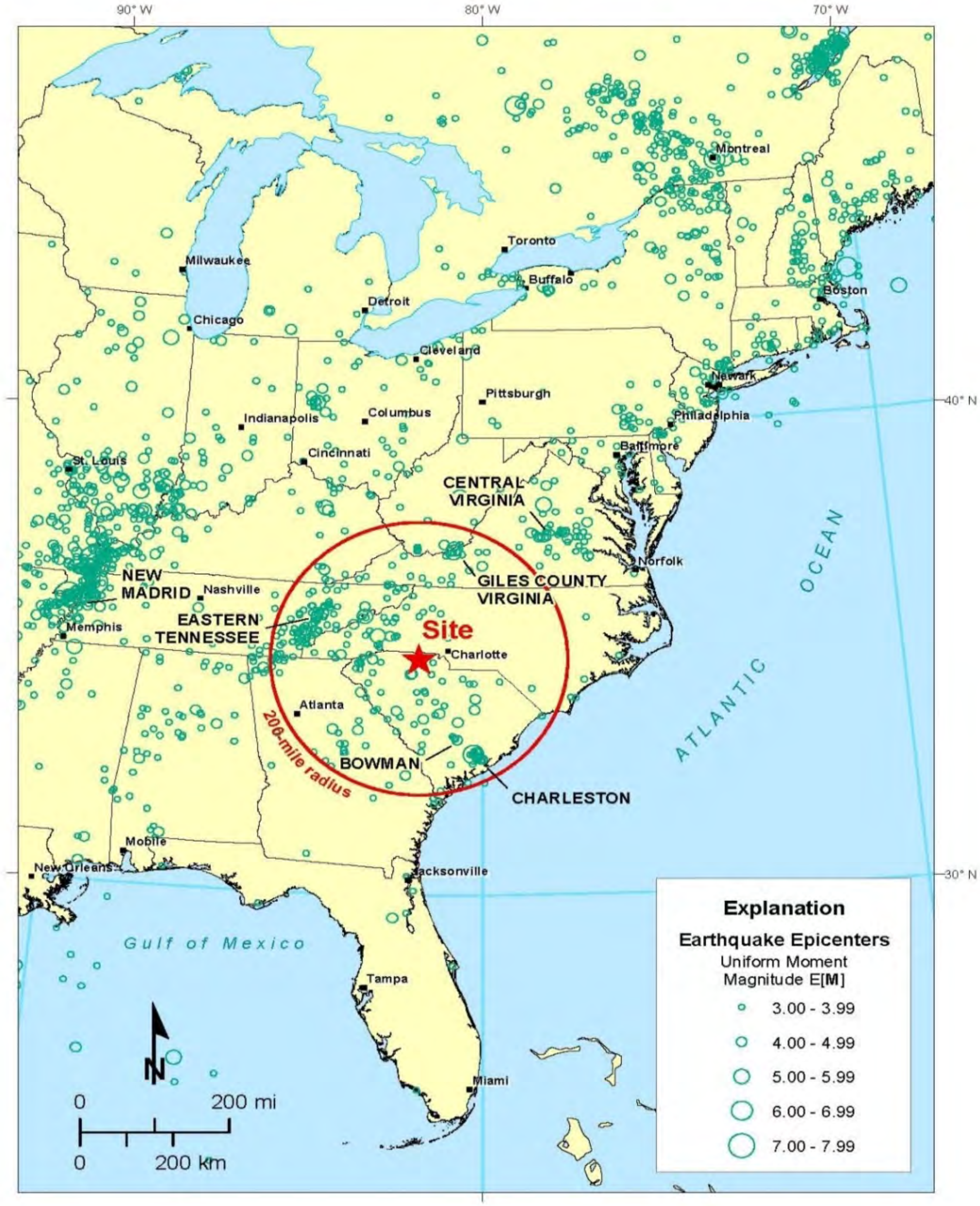


Figure 2.5.1-4 Location of WLS in Relation to Seismicity in the CEUS, Five Principal Areas of Seismicity in The Site Region, and Two Seismic Zones Outside the Site Region. Areas Within The Site Region Include The East Tennessee and Giles County Seismic Zones, the Middleton Place-Summerville and Adams Run seismic Zones at The Mapped Position of Charleston, and The Bowman Seismic Zone. The New Madrid and Central Virginia Seismic Zones Are The Two Located Outside The Site Region. (Reproduced from WLS COL FSAR Revision 8, Figure 2.5.1-214)

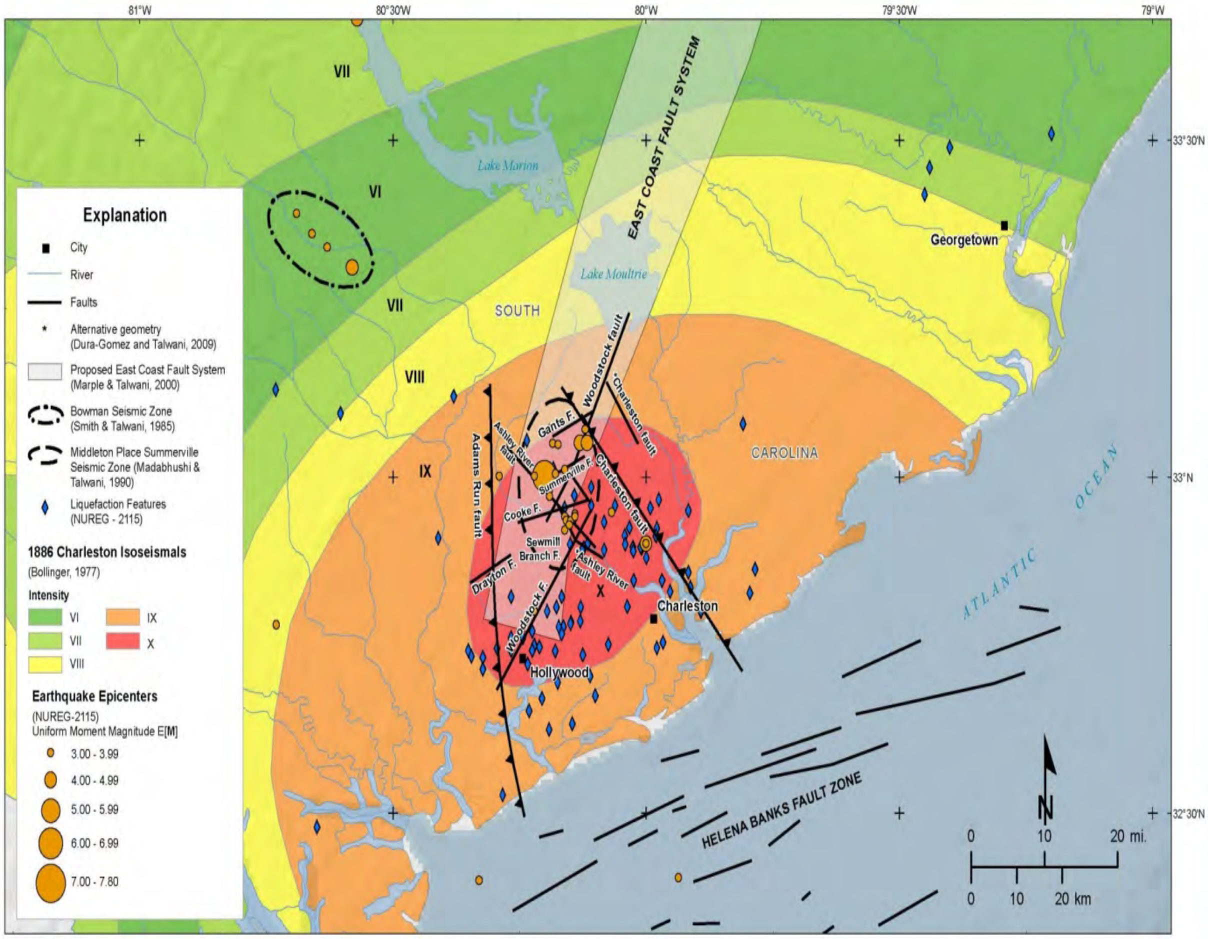


Figure 2.5.1-5 Locations of Tectonic Features, Including Postulated Faults and Seismically-Induced Liquefaction Features, in The Charleston Area. Not Shown is The Northwest-Striking Dorchester Fault, Inferred by Bartholomew and Rich (2007) to Lie Between The Postulated Northwest-Striking Adams Run and Charleston Faults. (Reproduced from WLS COL FSAR Revision 8, Figure 2.5.1-216)

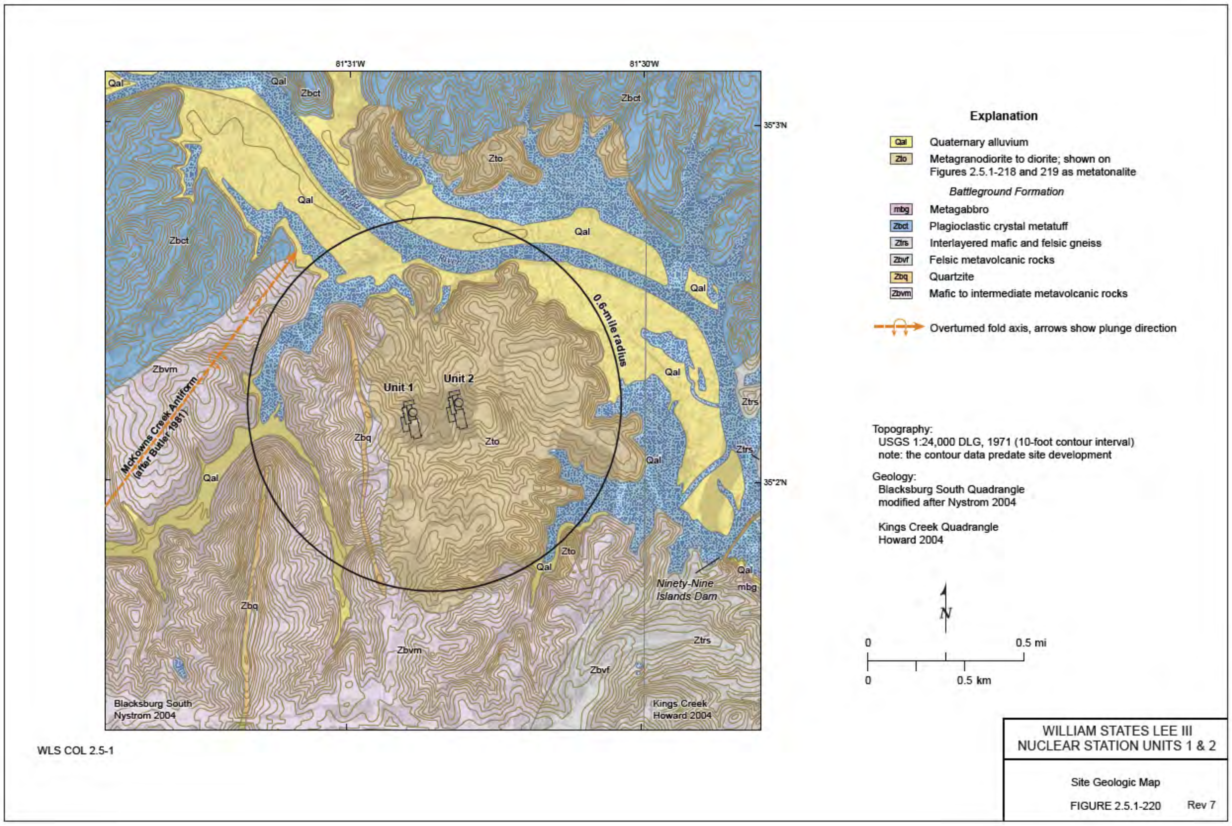


Figure 2.5.1-6 Locations of Lee Nuclear Site Units 1 and 2 Superimposed on The Geologic Map of the Site Location. The Map Clearly Shows That Rock Unit Zto is The Foundation Lithology for WLS COL Units 1 and 2. (Reproduced from WLS COL FSAR Revision 7, Figure 2.5.1-220)

2.5.2 Vibratory Ground Motion

2.5.2.1 *Introduction*

The vibratory ground motion is evaluated based on seismological, geological, geophysical, and geotechnical investigations carried out to determine the site-specific GMRS, which must meet the regulations for SSE provided in 10 CFR 100.23. The GMRS is defined as the free-field horizontal and vertical ground motion response spectra at the plant site. The development of the GMRS is based upon a detailed evaluation of earthquake potential, taking into account the regional and local geology, Quaternary tectonics, seismicity, and site-specific geotechnical engineering characteristics of the site subsurface material. The specific investigations necessary to determine the GMRS include the seismicity of the site region and the correlation of earthquake activity with seismic sources. Seismic sources are identified and characterized, including the rates of occurrence of earthquakes associated with each seismic source. Seismic sources that have any part within 320 km (200 mi) of the site must be identified. More distant sources that have a potential for earthquakes large enough to affect the site must also be identified. Seismic sources can be capable tectonic sources or seismogenic sources. The review covers the following specific areas: (1) seismicity; (2) geologic and tectonic characteristics of the site and region; (3) correlation of earthquake activity with seismic sources; (4) probabilistic seismic hazard analysis and controlling earthquakes; (5) seismic wave transmission characteristics of the site; (6) site-specific ground motion response spectrum; and (7) any additional information requirements prescribed within the "Contents of Application" sections of the applicable Subparts to 10 CFR Part 52, "Licenses, certifications, and approvals for nuclear power plants."

2.5.2.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 2.5, incorporates by reference AP1000 DCD, Revision 19, Section 2.5.2.

In addition, in WLS COL FSAR Section 2.5.2, the applicant provided site-specific information to address the following:

AP1000 COL Information Item

- WLS COL 2.5-2

The applicant provided additional information in WLS COL 2.5-2 to address COL Information Item 2.5-2. WLS COL 2.5-2 addresses the provision for site-specific information related to vibratory ground motion aspects of the site including: seismicity, geologic and tectonic characteristics, correlation of earthquake activity with seismic sources, probabilistic seismic hazard analysis (PSHA), seismic wave transmission characteristics and the SSE ground motion.

- WLS COL 2.5-3

The applicant provided additional information in WLS COL 2.5-3 to resolve COL Information Item 2.5-3, which addresses the provision for performing site-specific evaluations, if the site-specific GMRS at foundation level exceed the response spectra in AP1000 DCD Figures 3.7.1-1 and 3.7.1-2 at any frequency, or if soil conditions are outside the range evaluated for the AP1000 DCD.

2.5.2.2.1 Seismicity

WLS COL FSAR Section 2.5.2.1 states that the applicant used the most recent earthquake catalog published as part of NUREG-2115, "Central and Eastern United States Seismic Source Characterization for Nuclear Facilities," in its seismic hazard assessment at the WLS Site. The NUREG-2115 earthquake catalog covers earthquakes in the CEUS region from 1568 through 2008. The applicant indicated that since the compilation of the NUREG-2115 earthquake catalog, the moment magnitude (**M**) 5.8, 2011, Virginia earthquake was the most significant earthquake. The epicenter of this earthquake was located approximately 450 km (280 mi) from the WLS site. Although this earthquake falls outside the WLS Site region, the applicant discussed the earthquake as part of the discussion of the Central Virginia seismic zone (WLS COL FSAR Section 2.5.1.1.3.2.4) and in the source characterization (WLS COL FSAR Section 2.5.2.2.5.2). Figure 2.5.2-1 of this report shows the seismicity of the WLS site region and its surroundings. The largest earthquake within 40 km (25 mi) of the WLS Site in the NUREG-2115 earthquake catalog is the 1886 E[**M**] 4.13 event.

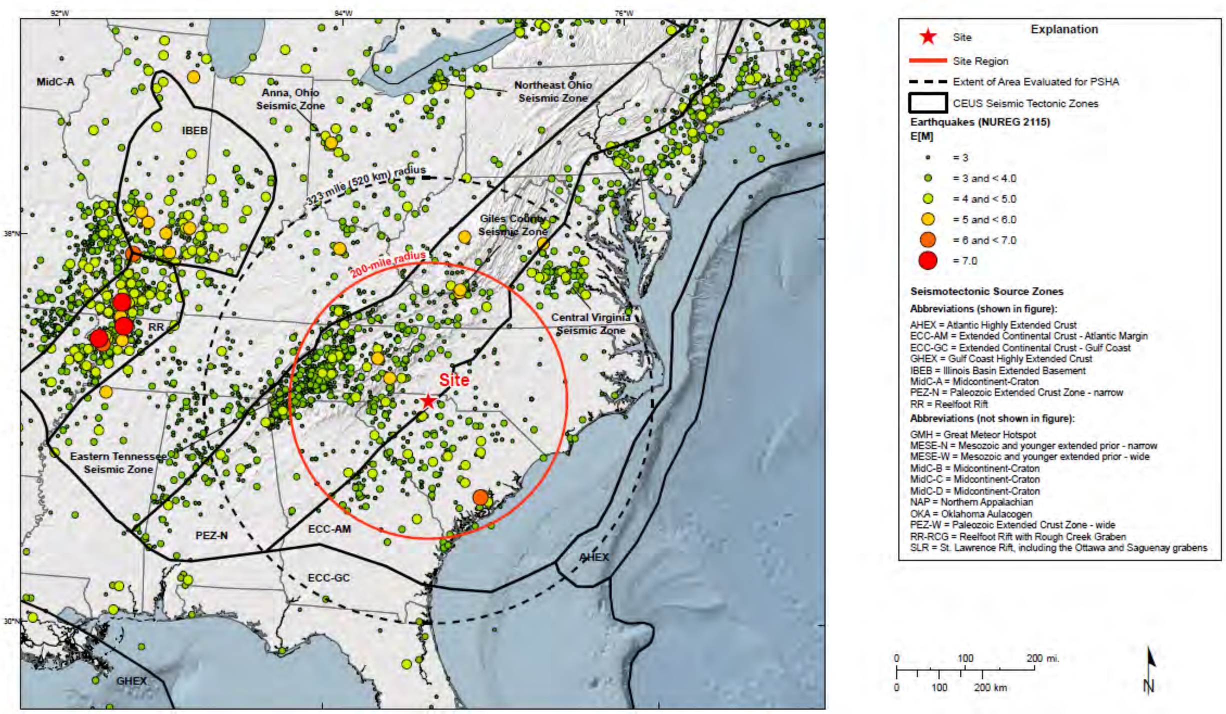


Figure 2.5.2-1 Map Showing The Earthquake Activity in The CEUS Region and The WLS Site (Ref. WLS COL FSAR Revision 8, Figure 2.5.2-248)

2.5.2.2.1.1 *Reservoir-Induced Seismicity*

In WLS COL FSAR Section 2.5.2.1.3, the applicant evaluated the potential for Reservoir-Induced Seismicity (RIS) associated with the construction and operation of Make-Up Pond C. The applicant stated that no documented RIS is associated with the Make-Up Pond B constructed as a part of the former Cherokee Nuclear Station, but noted that factors controlling the presence or absence of RIS are strongly dependent on local geologic properties, including reservoir rock type, fault and fracture characteristics, local and regional tectonics, and reservoir operation characteristics.

Following NUREG/CR-5503, the applicant stated the importance of making a distinction between triggered seismicity in regions of active faulting that are characterized by tectonic earthquakes of **M** greater than 5 in the historical record, such as the region west of the Rocky Mountains, and RIS in regions that are not associated with ongoing seismic activity and generally lack historical seismicity of **M** greater than 4. Triggered seismicity implies that a tectonic earthquake that was likely to occur at a later date is triggered and occurs earlier as a result of perturbations of elastic stresses and/or pore pressures associated with reservoir operations.

The applicant summarized the documented cases of RIS for reservoirs with hydraulic heights less than 60 m (196 ft) considering all U.S. Bureau of Reclamation reservoirs located in metamorphic terranes and all earthquakes located within 30 km (18 mi) of the reservoirs. In those cases post-impoundment maximum magnitudes have been less than **M**4 for reservoirs located in regions of low historical seismicity and have been less than or equal to **M**5 for reservoirs located in regions where historical pre-impoundment maximum magnitudes were greater than or equal to **M**5.5.

In the event that RIS associated with Make-Up Pond C occurs, the applicant concludes that it is unlikely that the induced magnitudes would exceed a **M** greater than 4, well below the short-period controlling earthquake. The applicant concluded that the potential for RIS associated with the Make-Up Pond C impoundment is low with a negligible risk to safe operations for WLS Units 1 and 2.

2.5.2.2.2 Geologic and Tectonic Characteristics of the Site and Region

WLS COL FSAR Section 2.5.2.2 describes the seismic sources and seismic model parameters that the applicant used to calculate the seismic ground motion hazard at the WLS site. The applicant used the NUREG-2115 regional seismic source characterization model developed for the CEUS region as a starting point for its seismic ground motion hazard analysis. The NUREG-2115 seismic source model is a model published in January 2012. The model development followed the Senior Seismic Hazard Analysis Committee (SSHAC) Level 3 procedures as outlined in NUREG/CR-6372, “Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts,” and NUREG-2117, “Practical Implementation Guidelines for SSHAC Level 3 and 4 Hazard Studies.” The NUREG-2115 model is a regional seismic source model to be used as a starting model in seismic hazard calculations for nuclear facilities in the CEUS region. The applicant stated that it conducted a review of the NUREG-2115 model to identify whether there is a need to update any of the seismic sources. Based on its review results, the applicant stated that the regional model, as published, is adequate for use in seismic hazard calculations for the WLS site. The following describes a summary of the NUREG-2115 model.

2.5.2.2.2.1 *Summary of NUREG-2115 Seismic Source Model*

The applicant stated that the CEUS SSC model described in NUREG-2115 contains two types of seismic sources: (1) distributed seismicity sources; and (2) repeated large magnitude earthquake sources. While the distributed seismicity sources were developed based on available earthquake locations and regional geologic/tectonic characterizations, the RLME sources were based on geologic and paleo-earthquake records. The RLME sources represent the zones of repeated (two or more) large magnitude earthquakes (**M** > 6.5) in the CEUS region.

The NUREG-2115 model categorizes the distributed seismicity sources into two subgroups: Mmax zones and seismotectonic zones. These subgroups represent uncertainties in source characterizations and differences of opinions in seismic source identification in this region. In hazard estimates, the Mmax and seismotectonic sources are weighted by 40 percent and 60 percent, respectively, to determine their contributions to the total seismic hazard at the site. The Mmax zones are broad seismic sources identified based on limited tectonic information and represent potential seismic sources of future earthquakes. The seismotectonic sources are those developed by extensive analyses of regional geology, tectonics, and seismicity in the CEUS region. Both the Mmax and the seismotectonic zones also include alternative source geometries, accommodating inherent uncertainty in seismic source characterization. The RLME sources are superimposed on the distributed seismicity sources.

2.5.2.2.2.2 *Post-NUREG-2115 Seismic Source Characterization Studies*

In WLS COL FSAR Section 2.5.2.2.5, the applicant discussed geologic and seismic investigations of the site region and beyond that provide information that could be used to evaluate and potentially update the NUREG-2115 model relevant to the WLS site PSHA. Specifically, the applicant discussed ongoing investigations of the: (1) Eastern Tennessee seismic zone (ETSZ); and (2) **M**5.8, 2011, Virginia earthquake.

Geologic Investigations of the Eastern Tennessee Seismic Zone

The ETSZ can be identified as a narrow trend of concentrated seismicity east of the New York-Alabama magnetic lineament and seismicity associated with the ETSZ is located within the WLS site region. The applicant stated that in spite of the high rate of seismic activity in the ETSZ, the largest historical earthquake in the region is magnitude 4.6 (magnitude scale unspecified). The applicant stated that recent geologic studies of the ETSZ either post-date the NUREG-2115 model or were published during development of the NUREG-2115 model. These studies suggest that the ETSZ may have produced large prehistoric earthquakes.

The applicant discussed the findings of Vaughn (2010), Obermeier (2010), Howard (2011), Warrell (2012), and Hatcher (2012). Vaughn (2010) found minor surface faulting, fracturing, and disrupted features in terrace alluvium, along with minor paleoliquefaction, northeast of Knoxville, TN. Similarly, Obermeier (2010) documented Douglas Reservoir fracture systems and sandy intrusions in terrace deposits that they interpret as paleoseismic in origin, although the applicant stated the significance of these features is unclear. Howard (2011) and Warrell (2012) document fractures, small faults, and displacements in Quaternary alluvium along Douglas Reservoir that they suggest resulted from earthquakes with magnitudes greater than 6.0 and 6.5 (magnitude scale unspecified). The applicant explained that Hatcher (2012) continued studying the Douglas Reservoir by coupling geologic observations with preliminary optically stimulated luminescence age dating of Quaternary deposits. Hatcher concluded that one or more "probable minimum" **M**6.5 earthquakes could be associated with the ETSZ within the last 73 to approximately 200 ka.

The applicant stated that while these recent studies strengthen the argument that the ETSZ has experienced at least one moderate-sized earthquake in the late Quaternary, the studies do not quantify parameters such as recurrence interval and magnitude, which are necessary parameters to demonstrate that the ETSZ produces repeating large-magnitude events as defined in NUREG-2115. Therefore, the applicant concluded that the ETSZ is best modeled within its NUREG-2115 host seismic source zones, the Mesozoic-and-younger extended crust Mmax zone (MESE) and the Paleozoic extended seismotectonic zones (PEZ), and that creation of an ETSZ RLME source is not warranted.

Investigations of the **M**5.8, 2011, Virginia Earthquake

The **M**5.8, 2011, Virginia earthquake was the largest historical event in the region and the largest instrumentally recorded earthquake in eastern North America since the 1988 **M**5.84 Saguenay earthquake. The **M**5.8, 2011, event occurred after completion of the NUREG-2115 earthquake catalog and is not included in the applicant's earthquake catalog. Numerous scientific studies have been conducted to identify the surface rupture associated with the earthquake, but the applicant reports that the consensus results of these investigations suggest that the earthquake occurred on a previously unrecognized structure, the dimensions of which are unknown and not observable at the ground surface. Therefore, the applicant did not include the **M**5.8, 2011, earthquake as a new fault or RLME source in the WLS site PSHA. The applicant stated that without slip-rate, recurrence, or maximum magnitude constraints for the structure defined by the distribution of aftershock hypocenters that likely produced the earthquake, it is most appropriate to consider it as an event captured by the host seismic source zones (ECC-AM, MESE-N, MESE-W, and Study Region) in the NUREG-2115 model framework. In addition, the applicant stated that because of the distance to the WLS site (approximately 450 km (280 mi)), and the buffer between the earthquake magnitude and lower end of the host seismic source zones' maximum magnitude distribution. The applicant concluded that no changes to the NUREG-2115 model were required due to the **M**5.8, 2011, event.

2.5.2.2.3 Correlation of Earthquake Activity with Seismic Sources

In WLS COL FSAR Section 2.5.2.3, the applicant discussed the correlation of the NUREG-2115 earthquake catalog to the specific NUREG-2115 seismic sources used by the applicant. Since the applicant did not update the NUREG-2115 earthquake catalog or seismic sources, the applicant chose to rely on the correlation of earthquake activity with seismic sources defined in NUREG-2115.

2.5.2.2.4 Probabilistic Seismic Hazard Analysis and Controlling Earthquakes

WLS COL FSAR Section 2.5.2.4 presents the results of the applicant's PSHA for the WLS site. In performing this analysis, the applicant followed the guidance provided in RG 1.208 to determine the seismic hazard curves and controlling earthquakes for the WLS site. The applicant based its analyses on the NUREG-2115 seismic source model and the EPRI (2013) ground motion prediction equations. The PSHA curves generated by the applicant represent generic hard rock conditions characterized by a shear wave velocity (V_S) in excess of 2.8 kilometers per second (km/s) (9,200 feet per second (fps)). The applicant also described the earthquake potential for the site in terms of a uniform hazard response spectra (UHRS) and the controlling earthquakes, the most likely earthquake magnitudes and source-site distances. The applicant determined the low- and high-frequency controlling earthquakes by deaggregating the PSHA curves at selected probability levels. The summary of the applicant's PSHA study is described below.

2.5.2.2.4.1 *PSHA Inputs*

To conduct the PSHA and obtain the UHRS at the site, it is necessary to study the site location and its surrounding regions to determine geological and seismological properties, as outlined in

RG 1.208. This requires determinations of active seismic source zones in the area, the seismic sources' model parameters, and appropriate ground motion models (GMM) for the region. The following subsections summarize the applicant's efforts in these areas.

2.5.2.2.4.1.1 Seismic Source Models and Parameters

The input model for the WLS PSHA study is the NUREG-2115 seismic source model. The applicant stated that it included all Mmax zones defined by the NUREG-2115 model in its hazard calculation for the WLS Site, truncated at a distance from the site of approximately 520 km (325 mi). The seismotectonic zones that the applicant included in the hazard calculation for the WLS site are the Atlantic Highly Extended Crust (AHEX), Extended Continental Crust-Atlantic Margin (ECC-AM), Extended Continental Crust-Gulf Coast (ECC-GC), Illinois Basin Extended Basement (IBEB), Paleozoic Extended Crust (PEZ), Midcontinent-Craton (MidC) seismotectonic zones (MidC-A through MidC-D), and Reelfoot Rift zone-Rough Creek graben (RR-RCG) zones. The applicant truncated each seismotectonic zone at a distance of 520 km (325 mi) from the site. The applicant determined that the RLME sources that contribute significantly to hazard at the WLS site are Charleston and New Madrid RLME sources, so the applicant included the Charleston and New Madrid RLME sources in its hazard calculations.

2.5.2.2.4.1.2 Ground Motion Models

In WLS COL FSAR Section 2.5.2.4.1, the applicant stated that it used the CEUS ground motion prediction model published in 2013 by EPRI for its PSHA calculations. These models were reviewed by the staff. In an August 28, 2013, letter to the Nuclear Energy Institute, the staff concluded that the EPRI (2013) GMM is an acceptable ground motion attenuation model for use by CEUS plants in developing plant-specific GMRS until such time as the Next Generation Attenuation project for eastern North America (NGA-East) project is completed and has been reviewed and approved by the staff. The NGA-East project is being conducted as a higher level SSHAC project than the EPRI (2013) GMM and will benefit from the development of the new GMM based on an expanded suite of earthquake recordings and simulations. Once complete, the NGA-East GMM will replace this EPRI (2013) GMM.

2.5.2.2.4.2 *PSHA Methodology and Calculation*

Using the updated NUREG-2115 seismic source characteristics and the EPRI (2013) GMM, the applicant performed PSHA calculations for peak ground acceleration (PGA) and spectral acceleration at ground motion frequencies of 0.5, 1.0, 2.5, 5, 10, and 25Hz. The applicant performed PSHA calculations for the WLS site assuming generic hard rock conditions at the site with V_S of 2.8 km/s (9,200 fps). The applicant first calculated mean and fractile rock seismic hazard curves at seven particular spectral frequencies (0.5, 1.0, 2.5, 5, 10, 25, and PGA (100 Hz)) and annual frequencies of exceedance (10^{-4}, 10^{-5}, and 10^{-6}). Then, the applicant deaggregated the results as described in RG 1.208 to calculate the controlling earthquakes for low- and high-frequency ground motions. Finally, the applicant used the WLS controlling earthquakes, and hard rock spectral shapes for CEUS earthquake ground motions recommended in NUREG/CR-6728 to calculate the final WLS generic hard rock UHRS.

2.5.2.2.4.3 *PSHA Results*

In WLS COL FSAR Section 2.5.2.4.2, the applicant stated that the local background, Charleston and New Madrid seismic sources contribute to seismic hazard at the WLS site. The applicant plotted its deaggregation results in WLS COL FSAR Figures 2.5.2-231 through -236. For a 10^{-4} annual frequency of exceedance, the applicant stated that the background and Charleston sources are the largest contributors to seismic hazard at both low- (1 and 2.5 Hz) and high-frequencies (5 and 10 Hz). For a 10^{-5} annual frequency of exceedance, the applicant obsevered that the background and Charleston sources are the largest contributor to seismic hazard for low-frequencies and the background source is the largest contributor to seismic hazard at high-frequencies. For a 10^{-6} annual frequency of exceedance, the applicant stated that the Charleston contribution is smaller at low-frequencies and is absent for high-frequencies and that the local background sources representing seismicity dominate for all annual frequencies of exceedance for high-frequencies.

The applicant also calculated the controlling earthquakes' distances and magnitudes for the low- and high-frequency earthquakes using the generic rock hazard curves. Table 2.5.2-1 of this report shows the results of the applicant's calculations.

Table 2.5.2-1 Controlling Earthquakes for the WLS Site
(Ref. WLSA COL FSAR Revision 8, Table 2.5.2-218)

	Mean 10^{-4}	Mean 10^{-5}	Mean 10^{-6}
Low Frequency **M** (R>100 km)	7.2	7.3	7.4
Low Frequency R (R>100 km)	250 km	230 km	190 km
High Frequency **M**	6.1	6.0	6.2
High Frequency R	35 km	16 km	12 km

Following the calculations of the controlling earthquake distances and magnitudes, the applicant calculates PSHA results using the NUREG-2115 seismic source model and the EPRI (2013) ground motion prediction models at the seven defined frequencies. The applicant determined the smoothed UHRS at the generic rock level and the applicant's UHRS curves for 10^{-4}, 10^{-5}, and 10^{-6} annual exceedance levels are shown as the blue, red, and green curves, respectively, in Figure 2.5.2-2 of this report.

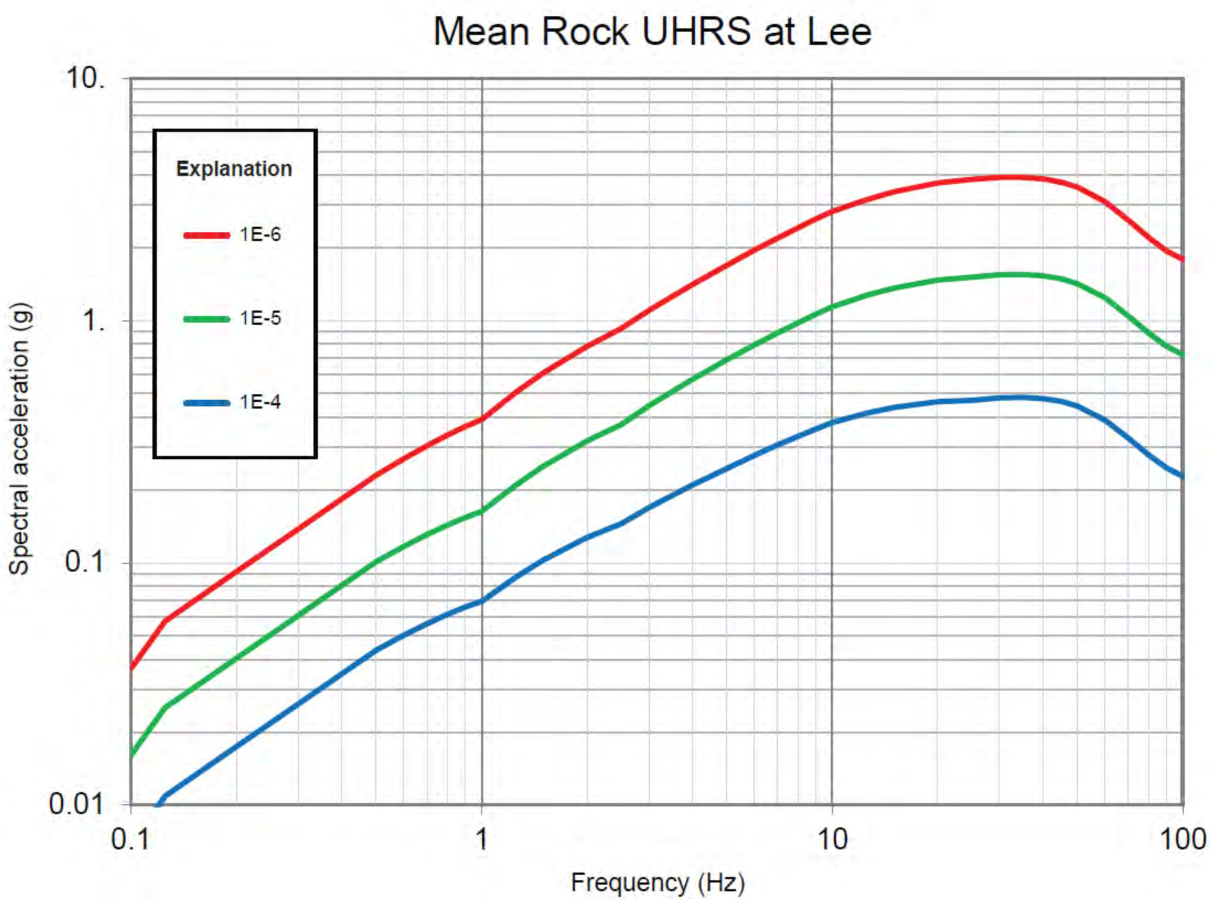


Figure 2.5.2-2 Smoothed Uniform Hazard Response Spectra For the Generic Rock Conditions at the WLS Site. (Ref. WLS COL FSAR Revision 8, Figure 2.5.2-266b)

2.5.2.2.5 Seismic Wave Transmission Characteristics of the Site

In WLS COL FSAR Section 2.5.2.5, the applicant characterized the WLS site as a hard rock site located on igneous and metamorphic rocks of Paleozoic age. WLS COL FSAR Section 2.5.1.2 describes the geology of the site area. The applicant referred to WLS COL FSAR Section 2.5.4 for a detailed discussion of dynamic and static properties of the site foundation materials. The majority of V_s measurements at the site exceed 2.8 km/s (9,200 fps), the hard rock definition used by CEUS attenuation relationships (2.83 km/s (9,282 fps)). Some near-surface V_s measurements fall below 2.8 km/s (9,200 fps). The applicant stated that variation in V_s measurements of several hundred feet per second centered at the hard rock condition result in a negligible variation in site response calculation results.

Based on the majority of near-surface V_s exceeding 2.8 km/s (9,200 fps), the applicant concluded that the WLS site is a hard rock site and therefore the EPRI (2013) ground motion equations were used directly, without calculation of site response. The UHRS reflects this hard rock condition.

2.5.2.2.6 Ground Motion Response Spectra

WLS COL FSAR Section 2.5.2.6 describes the method used by the applicant to develop the horizontal and vertical site-specific GMRS. The applicant first developed the horizontal GMRS and then obtained the vertical GMRS using vertical-to-horizontal (V/H) ratios. The applicant stated that it did not use the cumulative absolute velocity (CAV) model in its final hazard calculation.

2.5.2.2.6.1 *Horizontal GMRS*

The applicant calculated a horizontal, site-specific, performance-based GMRS using the method described in RG 1.208. The performance-based method achieves the annual target performance goal (PF) of 10^{-5} per year for frequency of onset of significant inelastic deformation. This damage state represents a minimum structural damage state, or essentially elastic behavior, and falls well short of the damage state that would interfere with functionality. The GMRS is calculated using the following relationship.

$$\text{GMRS} = \text{UHRS} * \text{DF}$$

Where

$$\text{UHRS} = \text{Mean } 10^{-4} \text{ UHRS}$$

$$\text{DF} = \max \{1.0, 0.6\ (\text{AR})^{0.8}\}$$

$$\text{AR} = 10^{-5} \text{ UHRS} / 10^{-4} \text{ UHRS}$$

RG 1.208 also states, if AR as defined above is greater than 4.2, then this relationship is no longer valid. In this case, RG 1.208 recommends setting the GMRS to 45 percent of the 10^{-5} site-specific surface UHRS curve. Figure 2.5.2-3 of this report shows the horizontal GMRS curve calculated for the WLS Site.

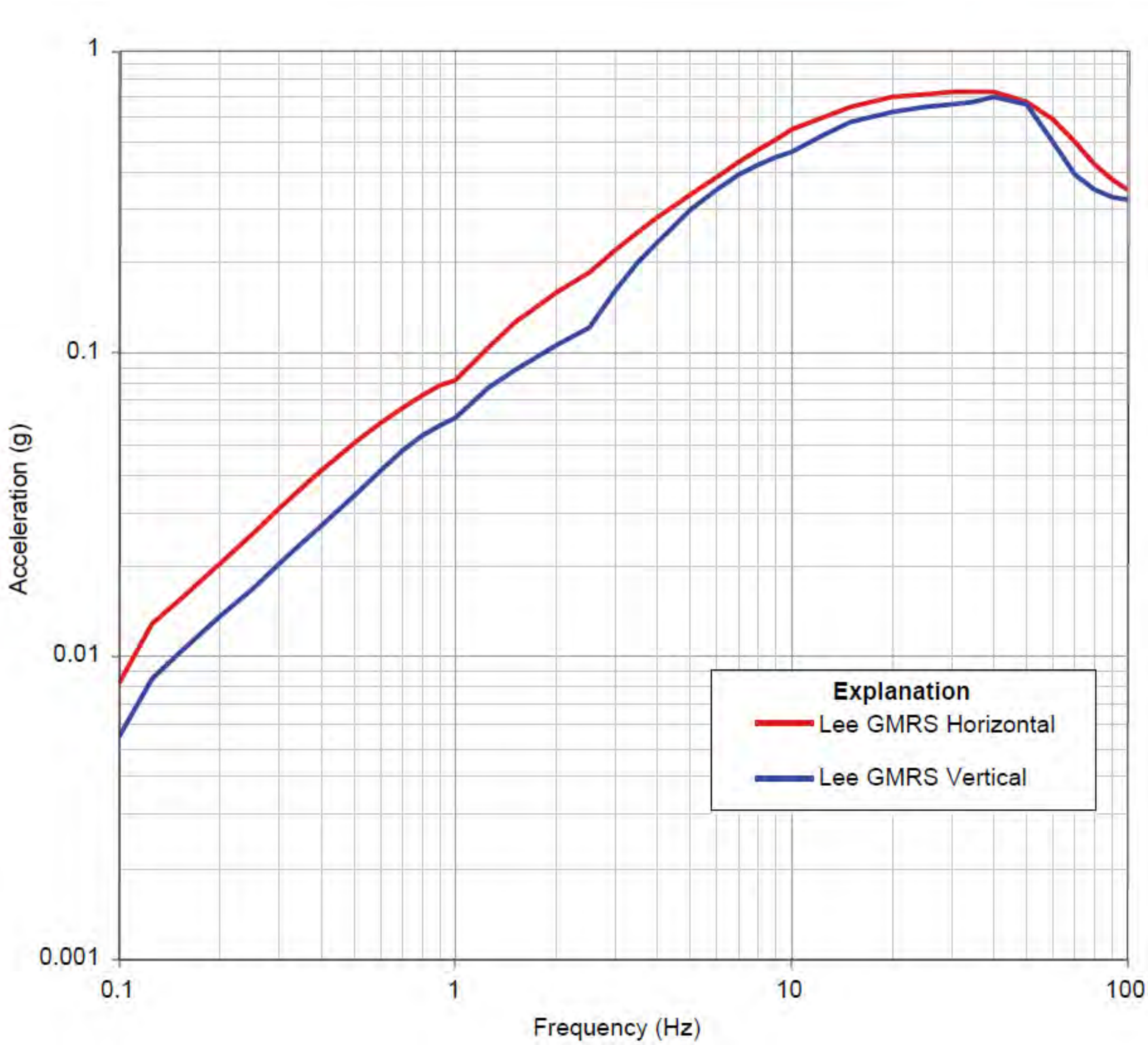


Figure 2.5.2-3 GMRS For Horizontal and Vertical Motion
(Ref. WLS COLFSAR Revision 8, Figure 2.5.2-239)

2.5.2.2.6.2 *Vertical GMRS*

In WLS COL FSAR Section 2.5.2.6, the applicant calculated the vertical GMRS by integrating the horizontal hazard curves with distributions of V/H ratios, which the applicant used to maintain the same exceedance probability as the horizontal hazard. V/H ratios can vary with source distance, so the range of ratios used by the applicant was intended to cover the deaggregation distances. The applicant presented a subset of the median estimates of the computed V/H ratios and demonstrated that there is little change at distances beyond about 10 to 15 km (6 to 9 mi), with an abrupt jump in the ratios within about 10 km (6 mi). The ratios are largely independent of frequency with a peak near 60 Hz, and range in amplitude from about 0.5 to about 1 as distance decreases. These values, at low frequency, are lower than empirical hard rock central and eastern North America (CENA) V/H ratios, which average about 0.8, decreasing from about 0.9 at 1 Hz to about 0.7 at 10 Hz. While these empirical V/H ratios are for Fourier amplitude spectra and not 5 percent damped response spectra and are dominated by small earthquakes of **M** less than or equal to 4, and large distances greater than or about 201 km (125 mi), the results illustrate the large uncertainty in vertical hard rock hazard for CENA and suggest large distant ratios may be greater than model predictions at low frequency. The applicant adopted a minimum V/H ratio of 0.7, the average of the empirical and simulations. To accommodate the change in source distance with both annual exceedance probability and structural frequency shown in the deaggregation plots (WLS COL FSAR Figures 2.5.2-231 through -236), V/H ratios computed at a suite of distances were given relative weights. The distances selected are 27, 6.4, and 0 km (17, 4, and 0 mi) to cover ratios reflecting distant, intermediate, and near source contributions. The resulting vertical GMRS is shown in Figure 2.5.2-3 of this report.

2.5.2.2.7 Development of FIRS for Units 1 and 2

In WLS COL FSAR Section 2.5.2.7, the applicant described how it preformed analyses to develop the site-specific foundation input response spectra (FIRS). While the applicant described development of FIRS in WLS COL FSAR Section 2.5.2.7, the staff's summary and technical evaluation of the FIRS is described in Section 3.7.1 of this report.

2.5.2.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed within the FSER related to the AP1000 DCD.

In addition, the applicable regulatory requirements for reviewing the applicant's discussion of vibratory ground motion are as follows:

- 10 CFR 100.23, as it relates to obtaining geologic and seismic information necessary to determine site suitability and ascertain that any new information derived from site-specific investigations does not impact the GMRS derived by a probabilistic seismic hazard analysis. In complying with this regulation, the applicant also meets guidance in RG 1.132 and RG 1.208.

- 10 CFR 52.79(a)(1)(iii), as it relates to consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity and period of time in which the historical data have been accumulated.

The related acceptance criteria from NUREG-0800, Section 2.5.2 are summarized as follows:

- Seismicity: To meet the requirements in 10 CFR 100.23, this subsection is accepted when the complete historical record of earthquakes in the region is listed and when all available parameters are given for each earthquake in the historical record.

- Geologic and Tectonic Characteristics of Site and Region: Seismic sources are identified and characterized.

- Correlation of Earthquake Activity with Seismic Sources: To meet the requirements in 10 CFR 100.23, acceptance of this subsection is based on the development of the relationship between the history of earthquake activity and seismic sources of a region.

- Probabilistic Seismic Hazard Analysis and Controlling Earthquakes: For CEUS sites relying on NUREG-2115 methods and data bases, the staff will review the applicant's PSHA, including the underlying assumptions and how the results of the site investigations are used to update the existing sources in the PSHA, how they are used to develop additional sources, or how they are used to develop a new data base.

- Seismic Wave Transmission Characteristics of the Site: In the PSHA procedure described in RG 1.208, the controlling earthquakes are determined for generic rock conditions.

- Ground Motion Response Spectra: In this subsection, the staff reviews the applicant's procedure to determine the GMRS.

In addition, the geologic and seismic characteristics should be consistent with appropriate sections from: RG 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants"; RG 1.132, "Site Investigations for Foundations of Nuclear Power Plants"; RG 1. 206, "Combined License Applications for Nuclear Power Plants (LWR Edition)"; and RG 1.208, "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion."

2.5.2.4 *Technical Evaluation*

The staff reviewed WLS COL FSAR Section 2.5.2 and checked the referenced DCD to ensure that the combination of the DCD and the information in the COL represent the complete scope of information relating to this review topic. The staff's review confirmed that the information contained in the application and incorporated by reference addresses the required information relating to the vibratory ground motion. AP1000 DCD, Revision 19, Section 2.5.2 was reviewed by the staff under Docket Number 56-006. The staff's technical evaluation of the information incorporated by reference related to the vibratory ground motion is documented in the staff's safety evaluation report on the DC application for the AP1000 design.

The staff reviewed the following information contained in the WLS COL FSAR:

AP1000 COL Information Item

- WLS COL 2.5-2

The staff reviewed WLS COL 2.5-2 related to COL Information Item 2.5-2, which addresses the provision for site-specific information related to the vibratory ground motion aspects of the site including: seismicity; geologic and tectonic characteristics; correlation of earthquake activity with seismic sources; probabilistic seismic hazard analysis; seismic wave transmission characteristics and the SSE ground motion.

- WLS COL 2.5-3

The staff reviewed WLS COL 2.5-3 related to COL Information Item 2.5-3, which addresses the provision for performing site-specific evaluations, if the site-specific GMRS at foundation level exceed the response spectra in AP1000 DCD Figures 3.7.1-1 and 3.7.1-2 at any frequency, or if soil conditions are outside the range evaluated for the AP1000 DCD.

Section 2.5.2.4 of this report provides the staff's evaluation of the seismic, geologic, geophysical, and geotechnical investigations carried out by the applicant to determine the site-specific GMRS, or the SSE ground motion for the site. The development of the GMRS is based upon a detailed evaluation of earthquake potential, taking into account the regional and local geology, Quaternary tectonics, seismicity, and site-specific geotechnical engineering characteristics of the WLS site subsurface material.

During the early site investigation stage, the staff visited the site and interacted with the applicant regarding the geologic, seismic and geotechnical investigations conducted for the WLS application. To thoroughly evaluate the geologic, seismic and geophysical information presented by the applicant, the staff visited the WLS site over April 27 to May 2, 2008, and January 27 and 28, 2009, Technical experts from the USGS accompanied the staff during the January 2009 site visit to assist with evaluation of the geologic and seismic data. The staff's evaluation of the information the applicant presented in WLS COL FSAR Section 2.5.2 and in the applicant's responses to RAIs is presented below.

As discussed at the beginning of this report (Section 2.5, "Geology, Seismology, and Geotechnical Engineering"), the staff issued several RAIs to the applicant and evaluated the responses received during the review process. However, following the Fukushima accident in Japan in March 2011, and the subsequent NRC NTTF recommendations as well as the NRC March 12, 2012, letter, "Request for information pursuant to Title 10 of the *Code of Federal Regulations* 50.54(f) regarding recommendations 2.1, 2.3, and 9.3 of the near term task force review of insights from the Fukushima Dai-ichi accident," requesting the operating nuclear power plants to re assess seismic hazards at their sites using the most recent seismic source models, the staff issued an RAI to all COL and ESP applicants (RAI 105, Question 01.05-1 was issued to WLS) to reassess the seismic hazard at their sites using the new seismic source models. In a January 30, 2014, response, the applicant revised the WLS COL FSAR significantly, especially WLS COL FSAR Section 2.5.2 related to seismic hazard calculations. As part of this WLS COL FSAR revision, the applicant replaced the EPRI (1986, 1989) seismic source models previously used in the seismic hazard calculations at the site with the newly published NUREG-2115 CEUS SSC model. With this change in the base seismic source model, and the move of the Unit 1 nuclear island footprint 20.1 m (66 ft) south and 15.2 m (50 ft) east of CNS Unit 1 to avoid an area of deeply weathered saprolitic bedrock at the northwest corner of the old CNS Unit 1, many of the earlier RAIs became irrelevant and were closed without any specific resolution. The staff's evaluations of these earlier RAIs are not part of this report. However, new RAIs that the staff developed in response to the revised WLS COL FSAR are applicable to the staff's review and are discussed below.

2.5.2.4.1 Seismicity

WLS COL FSAR Section 2.5.2.1 states that the earthquake catalog used for the WLS site seismic hazard assessment is the NUREG-2115 earthquake catalog. The earthquake catalog published as part of the NUREG-2115 seismic source model covers the entire CEUS region from 1568 through 2008 and includes a uniform moment magnitude scale for all earthquakes listed in the catalog. The NUREG-2115 earthquake catalog covers the seismicity in the WLS site region through 2008 and provides critical data to assess seismic source model parameters used in the WLS PSHA study. Seismic source model parameters, such as maximum magnitude and earthquake recurrence rates, are primarily determined based on information available in the earthquake catalog. Since the staff reviewed the NUREG-2115 earthquake catalog previously, the staff's technical evaluation of WLS COL FSAR Section 2.5.2.1 focused on whether earthquakes that have occurred since completion of the NUREG-2115 earthquake catalog should be used to update the catalog for use in the WLS site PSHA.

As part of its review of WLS COL FSAR Section 2.5.2.1, the staff performed confirmatory analysis by developing a supplemental earthquake catalog covering the CEUS region from 2009 through September 15, 2014. The staff used this supplemental earthquake catalog to determine whether there are new earthquakes in the CEUS region since the development of the NUREG-2115 earthquake catalog that might impact the parameters of seismic sources identified in the NUREG-2115 model used in the WLS site PSHA study. The staff used the USGS Advanced National Seismic Network earthquake catalog (ANSS)[23] for this analysis. The staff searched for earthquakes with magnitudes 3.0 and above within the time window covering 2009 through September 15, 2014, throughout the CEUS as defined by NUREG-2115. The staff's supplemental earthquake catalog confirmed that the applicant's use of the NUREG-2115 earthquake catalog without updates adequately characterizes seismicity for the WLS site.

The staff's supplemental catalog showed that there are 912 earthquakes in the CEUS region (Figure 2.5.2-4 of this report) that occurred between 2009 and September 15, 2014. Five of these earthquakes are greater than or equal to **M**5.0. The earthquake with a **M** greater than or equal to **M**5.0 that is the closest to the WLS site is also the largest of these five events and it is the **M**5.8, 2011, Virginia earthquake, which is located approximately 450 km (280 mi) from the WLS site. The other large magnitude events occurred in November 2011, in Oklahoma and August 2011, in southern Colorado. The staff identified 60 earthquakes in the range between **M**4.0 and 4.9 distributed over the CEUS region. The majority of the earthquakes (847 of the 912) in the updated catalog are small magnitude earthquakes (**M** < 4.0).

[23] Advanced National Seismic System (ANSS), ANSS Catalog Search, http://www.ncedc.org/anss/catalog-search.html.

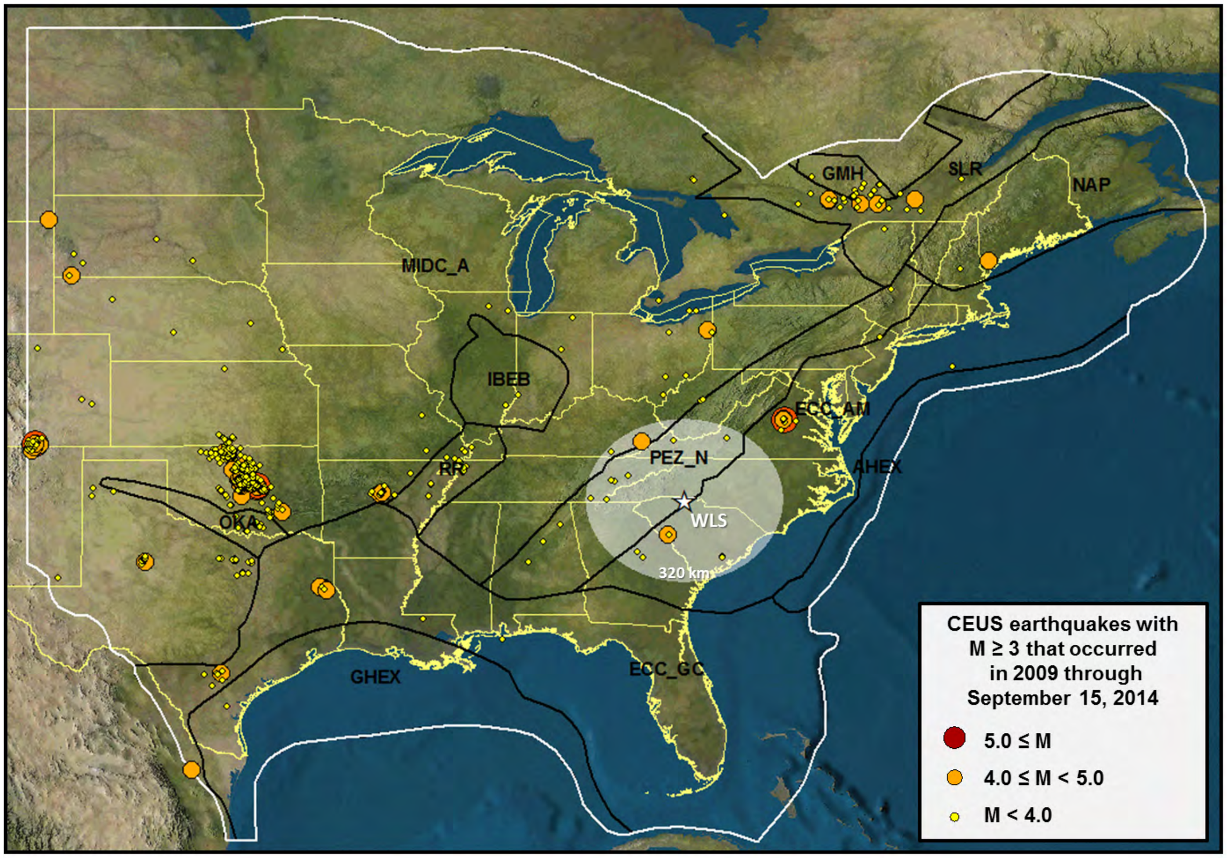


Figure 2.5.2-4 Earthquakes With Moment Magnitudes (M) Greater Than Or Equal to 3.0 In The CEUS That Occurred Between 2009 and September 15, 2014. The Outer White Polygon Defined the NUREG-2115 CEUS Region and the Black Lines and Labels Identify the NUREG-2115 CEUS-SS

Within the WLS site region, there were 15 earthquakes that occurred during 2009 through September 15, 2014. Of the 15 earthquakes within the WLS site region, there was a **M**4.2 on November 10, 2012, northeast of the WLS site in southeastern Kentucky and a **M**4.1 on February 15, 2012, southeast of the WLS site in eastern South Carolina. The other 13 earthquakes within the WLS site region that appear in the staff's supplemental earthquake catalog are all less than or equal to **M**3.3.

There are ten NUREG-2115 seismic sources zones that occur within the WLS site region. Those seismic source zones are modeled using maximum magnitude distributions that range from **M**5.6 through **M**8.1. The maximum magnitude distributions of the NUREG-2115 seismic sources zones that occur with the WLS site region are well above the 15 earthquake magnitudes that appear in the staff's supplemental earthquake catalog and occur within the WLS site region. Since all the large magnitude events (**M** > 5.0) are located outside of the WLS site region and all earthquakes in the supplemental catalog occurred within existing seismic sources with maximum magnitude distributions larger than the magnitudes of the observed events, the staff concludes from its confirmatory analysis that the earthquakes in the staff's supplemental earthquake catalog do not add any new information that might impact the parameters of the seismic sources of the seismic sources used in the WLS site PSHA study and, therefore, the applicant's use of NUREG-2115 earthquake catalog without updates adequately characterizes seismicity for the WLS site.

Reservoir-Induced Seismicity

Reservoir-induced seismicity is the triggering of earthquakes by the physical processes that accompany the impoundment of reservoirs (i.e., the effect of the mass of the added water on the underlying rocks, or the pore-fluid pressure changes from the added water). The applicant evaluated the potential for RIS associated with the construction and operation of Make-Up Pond C. To do this, the applicant reviewed worldwide cases of RIS near reservoirs with equivalent or greater hydraulic heights to Make-up Pond C, RIS near reservoirs in the Carolina Piedmont, and the U.S. Bureau of Reclamation records of RIS near reservoirs located in metamorphic terranes with equivalent or greater hydraulic heights to Make-up Pond C. Make-Up Pond C will be located in the Carolina Piedmont on metamorphic terrane, so the information the applicant reviewed is analogous to conditions at Make-Up Pond C and can be used to inform the analysis of the potential for RIS associated with the impoundment of that reservoir. Additionally, the applicant followed NUREG/CR-5503, "Techniques for Identifying Faults and Determining Their Origins," to analyze the relationships between faults, fault activity, and associated RIS.

The applicant conducted a thorough global review of reservoirs similar to or larger than Make-up Pond C. Since there are no active seismogenic faults near the location of Make-Up Pond C and historical seismic activity is low, any triggered seismicity associated with the impoundment of that reservoir is likely to be **M** < 5. Specifically for RIS associated with reservoirs located in the Carolina Piedmont on metamorphic terrane in regions of low historical seismicity, RIS magnitudes are **M** < 4. Additionally, no documented RIS was associated with the impoundment of Make-up Pond B during the construction of the former CNS located next to the WLS site. Lastly, as shown in Table 2.5.2-1 of this report, the local earthquakes that control the short-period hazard at the WLS are larger than potential RIS (minimum controlling earthquake **M** is equal to 6.0 for 10^{-4}, 10^{-5}, and 10^{-6} annual frequencies of exceedance). Therefore, the staff concludes it is unlikely that seismicity induced from the impoundment of Make-Up Pond C at the WLS site would exceed the **M** of the short-period controlling earthquakes and the potential for RIS associated with the Make-Up Pond C impoundment is low with a negligible risk to safe operations at the WLS site.

Staff Conclusions Regarding Seismicity

Based upon its review of the applicant's WLS COL FSAR Section 2.5.2.1 and the staff's confirmatory analysis, the staff concludes that the applicant used a complete and accurate earthquake catalog for the region surrounding the WLS site. The staff concludes that the seismicity catalog as described by the applicant in WLS COL FSAR Section 2.5.2.1 forms an adequate basis for the seismic hazard characterization of the site and meets the requirements of 10 CFR 52.79 and 10 CFR 100.23.

2.5.2.4.2 Geologic and Tectonic Characteristics of the Site and Region

WLS COL FSAR Section 2.5.2.2 describes the seismic sources and seismicity parameters used by the applicant to calculate the seismic ground motion hazard for the WLS site. Specifically, the applicant described the seismic source model published as part of NUREG-2115 in 2012. The staff previously reviewed the NUREG-2115 seismic source model and approved its use as a starting regional model for nuclear power plant applications. However, the NUREG-2115 model is a regional model and NUREG-2115 specifically states that it should be compared against the local data and information, and if needed, appropriate local adjustments must be conducted. As such, the staff primarily focused on the applicant's investigation of potential local seismic source and source parameter adjustments to the NUREG-2115 model.

2.5.2.4.2.1 *Modifications to NUREG-2115 model due to Post-NUREG-2115 Seismic Source Characterization Studies*

In WLS COL FSAR Section 2.5.2.2.5, the applicant discussed geologic and seismic investigations of the site region and beyond that provide information that could be used to potentially update the NUREG-2115 model relevant to the WLS site PSHA. Specifically, the applicant discussed ongoing investigations of the: (1) ETSZ; and (2) **M**5.8, 2011, Virginia earthquake. The following describes the staff's assessment of the applicant's evaluations of post-NUREG-2115 model seismic source characterization studies and their relevance to the WLS site PSHA.

<u>Geologic Investigations of the Eastern Tennessee Seismic Zone</u>

In WLS COL FSAR Section 2.5.2.2.5, the applicant discussed recent geologic studies of the ETSZ that either post-date the NUREG-2115 model or were published during development of the model. The seismicity associated with the ETSZ is located within the WLS site region. These studies suggest that the ETSZ may have produced large prehistoric earthquakes. Specifically, the applicant discussed the findings of Vaughn (2010), Obermeier (2010), Howard (2011), Warrell (2012), and Hatcher (2012). The applicant stated that while these recent studies strengthen the argument that the ETSZ has experienced at least one moderate-sized earthquake in the late Quaternary, the studies do not quantify parameters such as recurrence interval and magnitude, which are necessary parameters to demonstrate that the ETSZ produces repeating large-magnitude events as defined in NUREG-2115. Therefore, the applicant concluded that the ETSZ is best modeled within its NUREG-2115 host seismic source zones and that creation of an ETSZ RLME source is not warranted.

During the staff's review of the applicant's discussion of the ETSZ, the staff noted that the applicant focused on whether the findings of the new studies should be integrated into the NUREG-2115 model's characterization of seismic sources, but did not evaluate the potential significance of recent studies on the site-specific seismic hazard analysis. The new information suggests that the ETSZ might represent a site-specific seismic source that is not appropriately represented by the regional model. Since the applicant did not evaluate the potential significance of recent studies on the site-specific seismic hazard analysis, in RAI 117, Question 02.05.02-53, the staff requested that the applicant assess and evaluate the effects of such recent ETSZ research results on the site-specific seismic hazard analysis at the WLS site and to provide a thorough description of the assessment and evaluation, and provide the basis to explain and justify the conclusions.

In response to RAI 117, Question 02.05.02-53, the applicant performed hazard sensitivity studies to assess and evaluate the effects of the recent research results on the site-specific seismic hazard analysis at the WLS site. The applicant evaluated the ETSZ as a hypothetical RLME source as defined in NUREG-2115, sources that represent zones of repeated (two or more) large magnitude earthquakes (**M**>6.5) in the CEUS region. The applicant used the ETSZ source zone boundary defined by the USGS National Seismic Hazard Mapping Program (Peterson et al., 2008), as shown in Figure 2.5.2-5 of this report, for its sensitivity studies. The applicant performed two hazard sensitivity studies: (1) evaluated the sensitivity of the maximum magnitude distribution of the ETSZ source; and (2) evaluated the magnitude-frequency distribution for ETSZ. The applicant provided comparisons of its sensitivity studies to FSAR hazard results at the WLS site.

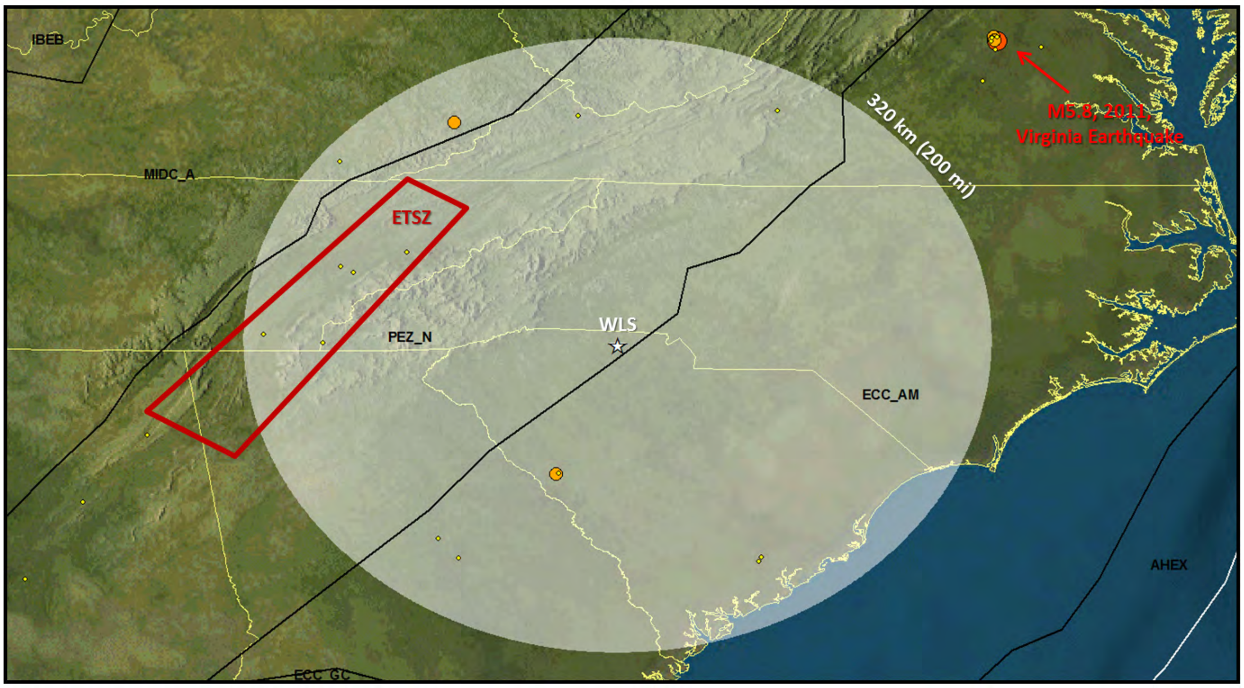


Figure 2.5.2-5 Approximate Representation of the USGS National Seismic Hazard Mapping Program (Peterson et al., 2008) ETSZ (Red Polygon) and The Black Lines and Labels Identify The NUREG-2115 CEUS-SSC Seismotectonic Zones (Model A)

To evaluate the sensitivity of the maximum magnitude distribution of ETSZ seismic sources, the applicant first assumed that the ETSZ had a maximum magnitude distribution equal to that of the PEZ-N seismic source, the host source for the WLS site. The applicant then assumed that a **M**6.5 earthquake occurred during the historical period within the ETSZ and re-assessed its maximum magnitude distribution, as shown in Table 2.5.2-2 of this report. The applicant then applied these updated, or sensitivity, maximum magnitude values to its ETSZ source and calculated the hazard for the ETSZ using both the PEZ-N maximum magnitude distributions and the updated maximum magnitude distributions. The applicant's calculations showed that the percent difference in total hazard using the two different ETSZ at spectral frequencies 1 and 10 Hz increased by 0.3 percent or less at the WLS site, as shown in Figure 2.5.2-6 of this report. The staff considers the differences calculated in this sensitivity study to be within the uncertainty in the overall PSHA calculations.

Table 2.5.2-2 Comparison Between Original (PEZ-N) and Updated Sensitivity Maximum Magnitude Distributions for The WLS ETSZ and Associated Weights (Ref. RAI 117, Question 02.05.02-53 Response Table 1)

CEUS SSC HID (M)	5.9	6.4	6.8	7.2	7.9
Sensitivity (M)	6.6	6.8	7.1	7.6	8.1
Weights	0.101	0.244	0.309	0.244	0.101

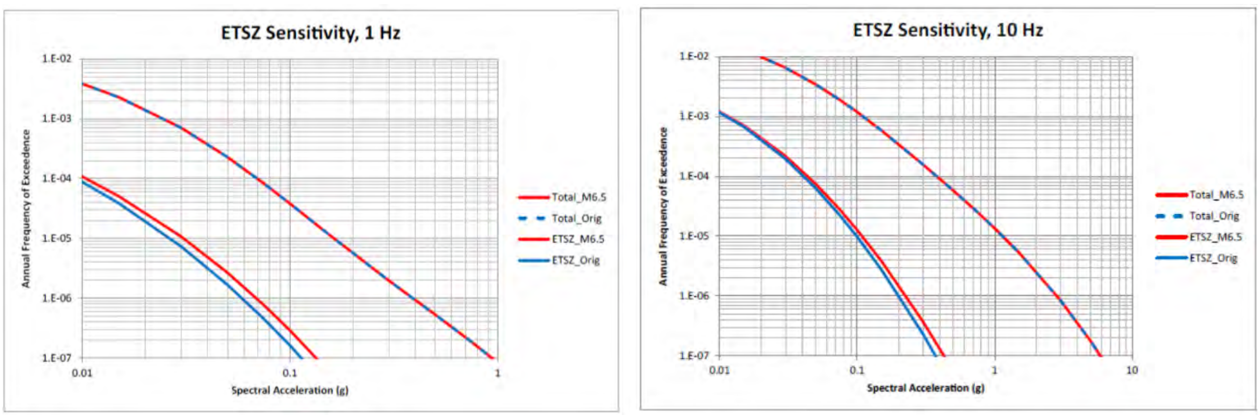


Figure 2.5.2-6 Comparison of Total Hazard and Hazard From The WLS ETSZ at 1 Hz (Left) and 10 Hz (Right). Blue Lines Show Hazard Using The PEZ-N Maximum Magnitude Distribution For The WLS ETSZ and The Red Lines Use The Updated Maximum Magnitude Distribution. (Ref. RAI 117, Question 02.05.02-53 Response Figure 2 and 3)

To evaluate the magnitude frequency distribution, or recurrence parameters, for the ETSZ, the applicant calculated magnitude frequency distributions for cell centers associated with the WLS ETSZ. Specifically, the applicant calculated magnitude frequency distributions for cell centers of the PEZ-N zone that lie within the boundary of the WLS ETSZ. The applicant stated that its magnitude frequency distribution calculations used the NUREG-2115 seismicity rates (a-values), b-values, and extracted parameters of rate per cell area, cell-area, and beta values for each source. The applicant developed the final weighted magnitude frequency distribution by applying the NUREG-2115 global logic tree weights to the calculated magnitude frequency distributions from each source. The applicant's calculated mean recurrence intervals for earthquakes occurring in the ETSZ with magnitudes ranging from 6.0 to 7.5 are tabulated in Table 2.5.2-3 of this report. The applicant stated that results show that the NUREG-2115 model indicates a mean recurrence interval for large earthquakes (**M**>6.5) in the ETSZ of 14,900 years. The applicant noted that this mean recurrence interval is consistent with the hypothetical occurrence of several such earthquakes in the late Pleistocene and Holocene (the last 130,000 years), as hypothesized by Hatcher (2012). The applicant concludes that these sensitivity results indicate that the NUREG-2115 model generates moderate to large events (**M** > 6.5) with sufficient frequency in the ETSZ area to explain the interpretation presented in Hatcher (2012) of field observations that imply the occurrence of two events of **M** > 6.5 or larger in the late Quaternary Period. The applicant stated that, without any modification, the NUREG-2115 model generates about seven **M**6.5 events and one **M**7.0 event in the ETSZ about every 100,000 years.

Table 2.5.2-3 Applicant's Estimated Mean Recurrence Intervals For Hypothetical ETSZ Large-Magnitude Events (Ref. RAI 117, Question 02.05.02-53 Response Table 4)

Magnitude (M)	Recurrence Interval (yrs)
≥6.0	3,200
≥6.5	14,900
≥7.0	103,000
≥7.5	1,000,000

The staff evaluated the applicant's response to RAI 117, Question 02.05.02-53 and its sensitivity studies. Because the applicant's sensitivity calculations showed that incorporating a **M**6.5 earthquake into the ETSZ magnitude distribution resulted in increases in total hazard of 0.3 percent or less at spectral frequencies 1 and 10 Hz at the WLS site and because the staff considers differences in total hazard of 0.3 percent as within the uncertainty in the overall PSHA calculations, the staff concludes that the applicant adequately assessed and evaluated the effects of recent research results on maximum magnitude distributions for the site-specific seismic hazard analysis at the WLS site. Additionally, because the applicant evaluated the magnitude-frequency distribution for the ETSZ and showed that the NUREG-2115 model generates **M**>6.5 events with sufficient frequency in the ETSZ area to explain the interpretation presented in recent research, the staff concludes that the applicant adequately assessed and evaluated the effects of recent research results on magnitude-frequency distributions for the site-specific seismic hazard analysis at the WLS site. Accordingly, the staff considers RAI 117, Question 02.05.02-53 resolved.

Investigations of the **M**5.8, 2011, Virginia Earthquake

The **M**5.8, 2011, Virginia earthquake occurred approximately 450 km (280 mi) northeast of the WLS site (Figure 2.5.2-5 of this report). The applicant did not include the **M**5.8, 2011, earthquake in its seismicity catalog or modify the NUREG-2115 model to incorporate the event in the WLS site PSHA. The staff reviewed updating the NUREG-2115 model to include the **M**5.8, 2011, Virginia earthquake in the WLS site PSHA. The staff previously reviewed the effect of the earthquake on site-specific PSHA parameters for a nuclear power plant application site located 180 km (112 mi) closer to the earthquake than the WLS site. Specifically, the staff reviewed the effect of the **M**5.8, 2011, Virginia earthquake on the PSEG site, which is located approximately 270 km (170 mi) from the earthquake. The staff's conclusion from review of the effect of the **M**5.8, 2011, Virginia earthquake on the PSEG site was that the effect of the **M**5.8, 2011, Virginia earthquake on the mean background hazard and the total mean site hazard at the PSEG site was negligible and that the PSEG applicant's use of the original NUREG-2115 model earthquake recurrence parameters was acceptable. Since the WLS site is located 180 km (112 mi) farther from the earthquake than the PSEG, the effect of the earthquake at the WLS site would be less than that assessed at the PSEG site. Therefore, the staff concludes that the applicant's use of the NUREG-2115 model earthquake recurrence parameters is acceptable for use in the WLS site PSHA.

2.5.2.4.2.2 *Staff Conclusions on Geologic and Tectonic Characteristics of the Site and Region*

Based upon its review of WLS COL FSAR Sections 2.5.2.2.5 and 2.5.2.4 and the applicant's response to RAI 117, Question 02.05.02-53, the staff concludes that the applicant adequately assessed the NUREG-2115 seismic sources as the input to its PSHA for the WLS site. In addition, the staff concludes that the applicant adequately considered modifications to the NUREG-2115 seismic sources for the WLS site. The staff concludes that the applicant's use of NUREG-2115 seismic source models as described by the applicant in WLS COL FSAR Sections 2.5.2.2.5 and 2.5.2.4 forms an adequate basis for the seismic hazard characterization of the site and meets the requirements of 10 CFR 52.79 and 10 CFR 100.23.

2.5.2.4.3 Correlation of Earthquake Activity with Seismic Sources

WLS COL FSAR Section 2.5.2.3 describes the correlation of seismicity in the region with the seismic source model used in the WLS PSHA study. The applicant discussed the correlation of the NUREG-2115 earthquake catalog to the specific NUREG-2115 seismic sources used by the applicant. The staff previously reviewed the NUREG-2115 seismic source model and approved its use as a starting regional model for nuclear power plant applications. Since the applicant did not update the NUREG-2115 earthquake catalog or seismic sources, the applicant chose to rely on the correlation of earthquake activity with seismic sources made in NUREG-2115. Based on the applicant's assessment in FSAR Section 2.5.2.3, the applicant's response to RAI 117, Question 02.05.02-53, and the staff's confirmatory analysis described in Section 2.5.2.4.1 of this report, the staff concludes that the applicant's characterization of the correlation of earthquake activity is adequate. Since the applicant adequately correlated the seismic activity in the WLS site region with the appropriate seismic source zones, the staff concludes that the applicant's analysis meets the requirements of 10 CFR 52.79 and 10 CFR 100.23.

2.5.2.4.4 Probabilistic Seismic Hazard Analysis and Controlling Earthquakes

In WLS COL FSAR Section 2.5.2.4, the applicant stated that it used the NUREG-2115 seismic model in the probabilistic seismic hazard calculations at the WLS site and the procedures outlined therein. Using the NUREG-2115 CEUS-SSC model sources and the EPRI (2013) GMM, the applicant calculated generic hard rock seismic hazard curves at the seven frequencies defined by the EPRI (2013) GMM. Using the hard rock seismic hazard curves, the applicant obtained UHRS at the annual frequency of exceedances of 10^{-4}, 10^{-5}, and 10^{-6}. Using the procedures outlined in RG 1.208, the applicant also developed the controlling earthquakes' magnitudes and distances. The following describes the staff's assessment of the applicant's PSHA calculations and the determination of the controlling earthquakes and their parameters.

2.5.2.4.4.1 *PSHA Inputs*

As described in Section 2.5.2.2.4 of this report, the applicant implemented the entire NUREG-2115 model with no modifications. The applicant stated that it included all Mmax zones defined by the NUREG-2115 model in its hazard calculation for the WLS site, truncated at a distance from the site of approximately 520 km (325 mi). The seismotectonic zones that the applicant included in the hazard calculation for the WLS site are the Atlantic Highly Extended Crust (AHEX), Extended Continental Crust-Atlantic Margin (ECC-AM), Extended Continental Crust-Gulf Coast (ECC-GC), Illinois Basin Extended Basement (IBEB), Paleozoic Extended Crust (PEZ), Midcontinent-Craton (MidC) seismotectonic zones (MidC-A through MidC-D), and Reelfoot Rift zone-Rough Creek graben (RR-RCG) zones. The applicant truncated each seismotectonic zone at a distance of 520 km (325 mi) from the site. The applicant determined that the RLME sources that contribute significantly to hazard at the WLS site are Charleston and New Madrid RLME sources, so the applicant included the Charleston and New Madrid RLME sources in its hazard calculations. The staff previously reviewed the NUREG-2115 seismic source model and approved its use as a starting regional model for nuclear power plant applications. Therefore, the applicant's PSHA inputs are consistent with RG 1.208 and the staff concludes that the applicant's PSHA inputs are adequate.

2.5.2.4.4.2 *PSHA Software Audit and Confirmatory Analysis*

During the development of the applicant's response to RAI 105, Question 01.05-1, the staff conducted software audits to distinct seismic hazard calculation software being used by the industry to respond to the NTTF Recommendation 2.1 seismic RAIs submitted to all COL and ESP applicants. The purpose of these audits was to review seismic hazard software and examine the implementation of the new seismic source models described in NUREG-2115. The objective was to gain in-depth understanding of the seismic software being used and review the implementation of the new seismic source model into the existing codes. The applicant contracted with ENERCON, who contracted with Lettis Consultants International, Inc. (LCI), formerly known as William Lettis & Associates, Inc., to perform its seismic hazard calculations. The LCI software audit took place on August 20 and 21, 2012. During the software audit, the staff reviewed software runs and reviewed several quality assurance documents related to LCI seismic hazard code.

As part of its confirmatory analysis, the staff used the NUREG-2115 CEUS-SSC model Mmax, seismotectonic, and RLME sources and independently calculated the seismic hazard curves at the WLS site for all seven ground motion frequencies defined in the EPRI (2013) ground motion prediction models. From the NUREG-2115 seismic source model, the staff included all Mmax and seismotectonic sources zones defined by the NUREG-2115 model in its hazard calculation for the WLS site and truncated the zones' hazard contributions at a distance from the site of 500 km (310 mi). The staff also included all RLME source zones from NUREG-2115 and truncated the RLMEs' hazard contributions at a distance from the WLS site of 1000 km (621 mi). Figure 2.5.2-7 of this report shows the staff's results as compared to the applicant's for the generic hard rock UHRS at the annual frequency of exceedances of 10^{-4} and 10^{-5}.

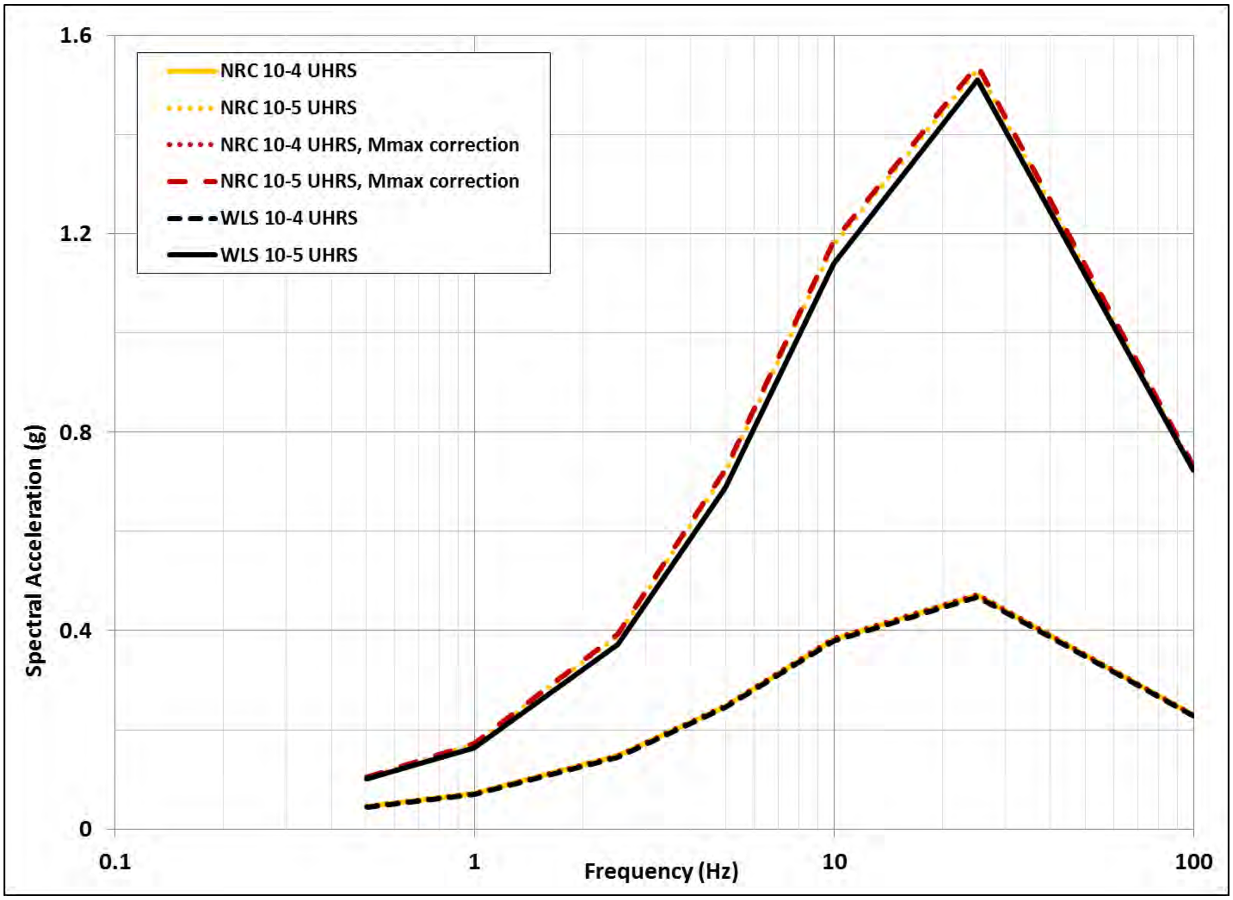


Figure 2.5.2-7 Staff Confirmatory Analysis of Hard Rock UHRS Calculations for Seven Spectral Frequencies at the Annual Frequency of Exceedances of 10^{-4} and 10^{-5}. The Black Lines Are the WLS UHRS, the Yellow Lines Are the Staff's Confirmatory UHRS, and the Red Lines Are the Staff's Confirmatory, Sensitivity Study UHRS Performed to Test and Possible Error in Maximum Magnitude Distributions in the NUREG-2115 Documentation

On July 28, 2014, EPRI wrote a letter (Agencywide Documents Access Management System (ADAMS) Accession No. ML14260A280), to the NRC notifying the agency of errors that EPRI identified in the NUREG-2115 documentation. The error is that maximum magnitude distributions for three of the seismotectonic model zones are incorrect. Table 2.5.2-4 of this report shows the published and corrected maximum magnitude values for the affected seismotectonic zones (PEZ-N, PEZ-W, and IBEB). As shown in Figure 2.5.2-4 of this report, the WLS Site is located with the PEZ-N seismotectonic zone in the NUREG-2115 seismotectonic zone model A. Alternative models in NUREG-2115 designate PEZ-W as the host seismotectonic zone for the WLS Site. Seismotectonic zone IBEB, however, is located approximately 534 km (332 mi) from the WLS Site (Figure 2.5.2-4 of this report) and is not included in the applicant's WLS Site PSHA. The staff performed confirmatory analysis to test the effect of these errors in maximum magnitudes on the seismic hazard analysis at the WLS site. The staff performed separate PSHA calculations using the corrected maximum magnitude distributions for PEZ-N, PEZ-W, and IBEB, as shown in Table 2.5.2-4 of this report. The results are shown in Figure 2.5.2-7 of this report and are labeled in that figure as "NRC…UHRS, Mmax correction." When the staff compared its original confirmatory UHRS at seven frequencies (yellow lines in Figure 2.5.2-7 of this report) to the "NRC…UHRS, Mmax correction" in Figure 2.5.2-7 of this report, the percent difference between the spectra range from 0.4 to 1.0 percent change. These percent differences are similar to those calculated by EPRI in its July 28, 2014, letter. In that letter, EPRI showed preliminary sensitivity study results for the NUREG-2115 Chattanooga test site. Like the WLS Site, the Chattanooga test site is located in the PEZ-N and PEZ-W seismotectonic zones. EPRI's preliminary sensitivity study results showed that at the NUREG-2115 Chattanooga test site the effect of the errors in maximum magnitude result in percent differences ranging from 0.6 to 1.5 percent for the generic hard rock UHRS at the annual frequency of exceedances of 10^{-4} and 10^{-5}. Both the staff's and EPRI's results show that the errors in maximum magnitude distributions in the NUREG-2115 PEZ-N, PEZ-W, and IBEB seismotectonic zones do not significantly affect the PSHA results at the WLS Site.

Table 2.5.2-4 Published And Corrected Maximum Magnitude Values For Seismotectonic Zones PEZ-N, PEZ-W, and IBEB. (Ref. Table in July 28, 2014, EPRI Letter)

PEZ-N		PEZ-W		IBEB	
CEUS SSC Report	Corrected Value	CEUS SSC Report	Corrected Value	CEUS SSC Report	Corrected Value
5.9	6.0	5.9	6.0	6.5	6.4
6.4	6.5	6.4	6.4	6.9	6.7
6.8	6.9	6.8	6.9	7.4	7.0
7.2	7.4	7.2	7.4	7.8	7.3
7.9	8.0	7.9	8.0	8.1	7.9

Based on review of WLS COL FSAR Section 2.5.2.4 and the staff's confirmatory analysis, the staff concludes that the applicant adequately characterized the mean seismic hazard at the WLS Site.

2.5.2.4.4.3 *Controlling earthquakes*

To determine the low- and high-frequency controlling earthquakes' magnitudes and distances, the applicant used a procedure called deaggregation of the seismic hazard. The applicant followed the deaggregation procedures outlined in RG 1.208, Appendix D. The deaggregation results showed that local seismic sources within 35 km (21 mi) of the WLS Site are the primary contributors to the high-frequency seismic hazard at the site, while the RLME sources as well as regional sources were contributors to the low-frequency seismic hazard at the WLS Site. Table 2.5.2-1 of this report shows the applicant's deaggregation results for the mean 10^{-4}, 10^{-5}, and 10^{-6} PSHA results. The applicant calculated the controlling earthquakes for two different cases: hazard from earthquakes located less than 100 km (62 mi) away and hazard from earthquakes located beyond 100 km (62 mi). As shown in the deaggregation, Table 2.5.2-1 of this report, for the high-frequency hazard, the controlling earthquakes are those with magnitudes about **M**6 occurring at short distances. For the low frequency hazard, the controlling earthquakes are several hundred kilometers away with magnitudes greater than **M**7.

Since the applicant used the guidance outlined in RG 1.208 to determine the controlling earthquakes and their magnitudes and distances, the staff concludes that the procedures used by the applicant are adequate and the resultant controlling earthquake parameters are representative of the controlling earthquakes in this region.

2.5.2.4.4.4 *Staff Conclusions Regarding PSHA and Controlling Earthquakes*

After its review of the applicant's PSHA and controlling earthquake determination, the staff's confirmatory calculations, and the staff's review of the code used by WLS during the software audit, the staff concludes that the applicant's PSHA adequately characterizes the seismic hazard for the WLS Site and that the controlling and deaggregation earthquakes determined by the applicant are representative of earthquakes that would be expected to contribute the most to the hazard. The staff concludes that the applicant's PSHA and controlling earthquake analysis meets the requirements of 10 CFR 52.79 and 10 CFR 100.23.

2.5.2.4.5 Seismic Wave Transmission Characteristics of the Site

WLS CIK FSAR Section 2.5.2.5 describes the WLS Site as a hard rock site located on igneous and metamorphic rocks of Paleozoic age with the majority of V_S measurements exceeding 2.8 km/s (9,200 fps) and variation in V_S measurements of several hundred feet per second centered at the hard rock condition considered to be a negligible variation in site response calculations. Therefore, the EPRI (2013) ground motion prediction equations are used directly, without calculation of site response and the UHRS reflects this hard rock condition. Based on the applicant's V_S analysis and the staff's review, the staff concludes that the applicant's use of the hard rock UHRS adequately characterizes the WLS Site. The applicant appropriately followed the guidance in RG 1.208, which meets the requirements of 10 CFR 52.79 and 10 CFR 100.23.

2.5.2.4.6 Ground Motion Response Spectra

WLS COL FSAR Section 2.5.2.6 describes the method used by the applicant to develop the horizontal and vertical, site-specific, GMRS. To obtain the horizontal GMRS, the applicant used the performance based approach described in RG 1.208 and American Society of Civil Engineers/Structural Engineering Institute (ASCE/SEI) Standard 43-05, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities." WLS COL FSAR Section 2.5.2.6, states that the horizontal GMRS (for each spectral frequency), is obtained by scaling the 10^{-4} soil UHRS by the design factor specified in RG 1.208. The final GMRS is shown in Figure 2.5.2-3 of this report.

In WLS COL FSAR Section 2.5.2.6., the applicant stated that to accommodate the change in source distance with both annual exceedance probability and structural frequency, the applicant's computed V/H ratios at a suite of distances are given relative weights, as shown in WLS COL FSAR Table 2.5.2-223. The probabilistic method, Approach 3, used by the applicant is described in NUREG/CR-6728. Since the applicant used an accepted methodology presented in NUREG/CR-6728, the staff concludes that the applicant's V/H ratios are adequate for the use of the WLS Site vertical GMRS. The applicant's horizontal and vertical GMRS are shown in Figure 2.5.2-3 of this report. Since the applicant used the standard procedure outlined in RG 1.208 and NUREG/CR-6728 to develop both the horizontal and vertical GMRS, the staff concludes that the applicant's GMRS adequately represents the WLS Site ground motion. This information addresses WLS COL 2.5-2 to resolve COL Information Item 2.5-2 and WLS COL 2.5-3 to resolve COL Information Item 2.5-3. In conclusion, the applicant provided sufficient information for satisfying 10 CFR 52.79 and 10 CFR 100.23.

2.5.2.5 *Post Combined License Activities*

There are no post COL activities related to this section.

2.5.2.6 *NRC Conclusions Regarding Vibratory Ground Motion*

The staff reviewed the application and checked the referenced DCD. The staff confirmed that the applicant has addressed the required information relating to vibratory ground motion, and that there is no outstanding information expected to be addressed in the WLS COL FSAR related to this subsection. Accordingly, the staff considers RAI 105, Question 01.05-1, which is the RAI issued after the NTTF recommendation following the Fukushima accident in Japan in March 2011, resolved.

As set forth above, the staff reviewed the seismic information submitted by the applicant in WLS COL FSAR Section 2.5.2. On the basis of its review of WLS COL FSAR Section 2.5.2, the staff finds that the applicant provided a thorough characterization of the seismic sources surrounding the site, as required by 10 CFR 100.23. In addition, the staff finds that the applicant adequately addressed the uncertainties inherent in the characterization of these seismic sources through a PSHA, and that this PSHA follows the guidance provided in RG 1.208. The staff concludes that the controlling earthquakes and associated ground motion derived from the applicant's PSHA are consistent with the seismogenic region surrounding the COL site. In addition, the staff finds that the applicant's GMRS, which was developed using the performance-based approach, adequately represents the regional and local seismic hazards and accurately includes the effects of the local site subsurface properties. The staff concludes that the proposed WLS COL site is acceptable from a geologic and seismologic standpoint and meets the requirements of 10 CFR 100.23.

2.5.3 Surface Faulting

2.5.3.1 *Introduction*

WLS COL FSAR Section 2.5.3 discusses information related to the potential for tectonic (i.e., due to faulting) and non-tectonic surface and near-surface deformation collected by the applicant during site characterization investigations. This information provided by the applicant addresses the following specific topics: geologic, seismic, and geophysical investigations; geologic evidence, or absence of evidence, for surface deformation; correlation of earthquakes with capable tectonic sources; ages of most recent deformations; relationships between tectonic structures in the site area and regional tectonic structures; characterization of capable tectonic sources; designation of zones of Quaternary (2.6 million years (Ma) to present) deformation in the site region; and the potential for surface tectonic and non-tectonic deformation at the site.

2.5.3.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 2.5 incorporates by reference AP1000 DCD, Revision 19, Section 2.5.3. In addition, in WLS COL FSAR Section 2.5.3, the applicant provided site-specific supplemental information to address the following:

AP1000 COL Information Item

- WLS COL 2.5-4

The applicant provided additional information in WLS COL 2.5-4 to address COL Information Item 2.5-4 (COL Action Item 2.5.3-1). WLS COL 2.5-4 addresses the evaluation of site-specific surface and subsurface geologic, seismic, and geophysical information related to the potential for surface and near-surface faulting at the site.

The applicant developed FSAR Section 2.5.3 for WLS based on information derived from review of published literature and geologic maps; interpretation of aerial photographs and satellite imagery; discussions with experts familiar with the geology, seismology, and tectonic history of the site region, site vicinity, site area, and site location; review of historical and recorded seismicity data; field geologic and aerial reconnaissance investigations performed specifically for the WLS application; analysis of test pits, trenches, and boreholes located at WLS; geologic mapping of bedrock exposures in the excavations for former CNS Unit 3 (WLS Unit 2) and CNS Unit 2; and the central and eastern U.S. seismic source characterization (CEUS SSC) report, provided as NUREG-2115, that contains the most recent seismic source models for the CEUS. The applicant also used information presented in the PSAR for the CNS site to develop FSAR Section 2.5.3 for WLS. In addition, as part of the site characterization studies performed for the WLS application to assess results of previous geologic mapping of foundation bedrock exposures in excavations at the CNS site, the applicant performed confirmatory geologic mapping of foundation grade level bedrock exposed in a small portion of the CNS Unit 2 excavation and compared that geologic map to archived scanned records for the CNS site.

As discussed in more detail in Section 2.5.1.2 of this report, to support the confirmatory geologic mapping effort described above and document lithologies and geologic features (including tectonic faults, shear and breccia zones, and fractures) that occur in foundation grade level bedrock underlying CNS Unit 1(WLS Unit 1), the applicant compiled a detailed geologic map of the foundation grade level bedrock surface in the CNS Unit 1 excavation from original field notes and geologic maps prepared in the 1970s to characterize site location geology in that excavation. The applicant provided a report containing the compiled geologic map and associated data, which documented the similarity of rock types and orientations and ages of tectonic structures in the WLS Unit 1/CNS Unit 1, CNS Unit 2, and WLS Unit 2/CNS Unit 3 excavations.

Based on information derived from the sources defined above, the applicant concluded in FSAR Section 2.5.3 that no evidence exists for Quaternary faults or capable tectonic sources within the site vicinity or at the site location, and that the potential for non-tectonic surface or near-surface deformation at the site location is negligible. With regard to assessment of evidence for capable tectonic sources, based on the definition in RG 1.208, the applicant stated that a capable tectonic source is a tectonic structure that can generate both tectonic surface deformation (e.g., faulting or folding) and vibratory ground motion in the present seismotectonic setting. Sections 2.5.3.2.1 through 2.5.3.2.8 of this report summarize the information provided by the applicant in FSAR Section 2.5.3 related to surface and near-surface tectonic deformation due to faulting, as well as surface and near-surface non-tectonic deformation.

2.5.3.2.1 Geologic, Seismic, and Geophysical Investigations

FSAR Section 2.5.3.1 outlines the geologic, seismic, and geophysical investigations performed by the applicant to assess the potential for tectonic and non-tectonic surface and near-surface deformation in the WLS vicinity and site area and at the site location. Sections 2.5.3.2.1.1 through 2.5.3.2.1.6 of this report summarize these investigations and the conclusions made by the applicant based on them in regard to tectonic and non-tectonic surface and near-surface deformation at WLS.

2.5.3.2.1.1 *Previous Site Investigations*

FSAR Section 2.5.3.1.1 discusses previous investigations conducted at WLS in connection with preparation of the PSAR for the former co-located CNS site. The applicant reported that detailed excavation mapping at the CNS site did not reveal any evidence for active or geologically recent (i.e., Quaternary) tectonic faulting in the WLS area. The applicant indicated that minor shear zones mapped in the CNS excavations commonly occurred near the contacts of metamorphosed mafic dikes that intrude metamorphosed plutonic intrusive body Zto, which comprises the foundation unit at WLS. The applicant cross-referenced FSAR Sections 2.5.1.2 and 2.5.4.1 for a detailed discussion of these minor geologic features that occur in foundation bedrock at WLS.

2.5.3.2.1.2 *Published Geologic Mapping*

FSAR Section 2.5.3.1.2 discusses the results of geologic mapping conducted in the site area by the USGS, the South Carolina Geological Survey, and other researchers. The applicant reported that these published geologic maps do not show any evidence for geologically recent or active faulting in the site area. The applicant noted that the geologic maps developed for the CNS site in the 1970s are the most detailed maps documenting geologic characteristics of WLS.

2.5.3.2.1.3 *Current Geologic Mapping*

FSAR Section 2.5.3.1.3 discusses current geologic mapping at WLS. The applicant indicated that the published geologic maps discussed in FSAR Section 2.5.3.1.2 formed the basis for the geologic maps presented in FSAR Section 2.5.1. The applicant conducted field reconnaissance of the site location, site area, and site vicinity for the WLS application to refine these geologic maps as necessary. In addition, the applicant analyzed the contact between the western margin of metamorphosed plutonic intrusive rock mass Zto, the foundation unit at WLS, and the adjacent metavolcanic country rock that the plutonic body intrudes to determine if any evidence for faulting exists along this contact. The applicant investigated the contact using geologic mapping, boreholes, test pits, and trenches and concluded that the irregular map pattern and intrusive character of the contact precluded a fault along the western margin of foundation unit Zto.

2.5.3.2.1.4 *Previous Seismicity Data*

FSAR Section 2.5.3.1.4 discusses previous seismicity data within and beyond the site vicinity. The applicant stated that, based on the earthquake catalog included in the CEUS SSC report (NUREG-2115), the largest earthquake in the site vicinity was an 1886 event with an expected moment magnitude (E[M]) of 4.13 (i.e., estimated given the uncertainty in earthquake magnitude) located south-southwest of WLS as shown in Figure 2.5.3-1 of this report. The E[M] 4.54 January 1913 Union County, South Carolina earthquake, also located south-southwest of WLS just outside the site vicinity (Figure 2.5.3-1 of this report), had an estimated Rossi-Forel shaking intensity of VI at WLS. A Rossi-Forel VI is approximately equivalent to a shaking intensity of VI in the more commonly used Modified Mercalli Intensity (MMI) scale, indicating an earthquake felt by all but causing only slight damage. The applicant stated that the 1886 Charleston earthquake, which occurred in the site region with an epicenter likely more than 240 km (150 mi) from WLS, produced a shaking intensity at the site of MMI VI. The applicant noted that the Union County and Charleston earthquakes have not been related to known causative faults.

2.5.3.2.1.5 *Current Seismicity Data*

WLS COL FSAR Section 2.5.3.1.5 discusses current seismicity data within and beyond WLS vicinity, including data from the updated CEUS SSC earthquake catalog (NUREG-2115) covering earthquakes that occurred within the CEUS between 1568 and 2008. The applicant cross-referenced WLS COL FSAR Section 2.5.2.1 for a more detailed description of this earthquake catalog.

The applicant reported that four minor earthquakes not included in the CEUS SSC catalog occurred in northeastern South Carolina in 2006. Two of these earthquakes occurred on January 24 and 25, 2006, approximately 32 km (20 mi) southwest of the site location with body-wave magnitudes (m_b) of 2.5 and 1.5, respectively. The applicant noted that the epicenters of these two small earthquakes have not been accurately located due to their small magnitudes. The other two 2006 earthquakes occurred in September more than 121 km (75 mi) east-southeast of WLS, as shown in Figure 2.5.3-1, with m_b values of 3.5 (September 22) and 3.7 (September 25). The applicant stated that the two September earthquakes appear to be spatially related to a small Mesozoic (25 to 145.5 Ma) extensional basin lying beneath the Coastal Plain as mapped by Benson (1992). Talwani (2006) suggested that the earthquakes may be spatially related to the regional Eastern Piedmont fault system (EPFS) of Hatcher and others (1977), which lies beneath the Coastal Plain at the estimated epicentral locations of the two earthquakes. The applicant reported that there is no definitive correlation of the September 2006 earthquakes with the EPFS because of the uncertainty in their epicentral locations.

2.5.3.2.1.6 *Current Aerial and Field Reconnaissance*

WLS COL FSAR Section 2.5.3.1.6 discusses current aerial and field reconnaissance studies performed using aerial photographs, satellite imagery, and topographic maps. The applicant indicated that no information acquired from the reconnaissance studies showed any evidence of active tectonic surface deformation (i.e., surface fault rupture, warping, or offset of geomorphic features) within the site.

The applicant reported that a northeast-trending topographic lineament, which occurs about 3 km (2 mi) northwest of the WLS location (i.e., Lineament 1), resulted from localized erosion along a tributary of the Broad River that flows parallel to an erosion-resistant, northeast-trending, quartzite-capped ridge. Although this lineament has the same trend as the regional northeastern structural grain of the Appalachian orogenic belt, it does not extend across the Broad River and the applicant concluded that it is non-tectonic in origin.

2.5.3.2.2 Geologic Evidence, or Absence of Evidence, for Surface Deformation

WLS COL FSAR Section 2.5.3.2 addresses the presence or absence of evidence for surface deformation within the WLS vicinity and at the site location. The applicant reviewed existing literature, performed aerial and field reconnaissance studies, and examined aerial photographs and satellite imagery for evidence of Quaternary (2.6 Ma to present) surface deformation along six mapped fault traces, interpreted to be Paleozoic (> 251 Ma) in age, which occur in the site vicinity. These structures include the Kings Mountain shear zone; the southwestern extension of the Boogertown shear zone; an unnamed fault north of Gaffney; and the Tinsley Bridge, Brindle Creek, and Reedy River faults. Figure 2.5.3-1 of this report shows the locations of these six faults relative to the WLS location. The applicant documented a Paleozoic age for these six tectonic features based on geologic field relationships, and concluded that no evidence exists for Quaternary surface deformation associated with these faults.

Based on geologic field relationships, the applicant also stated that strain markers comprising two elongated, north-striking quartzite units in the western part of the site area and the northerly-striking western margin of the metamorphosed pluton comprising foundation unit Zto do not show any displacement along northeast-striking faults. A northeastern orientation is the predominant trend of regional tectonic features in the WLS vicinity and site region. The applicant stated that the quartzite units and the metamorphosed pluton are Precambrian (> 542 Ma) in age and provide reliable strain markers for determining that no major faults project into or across the WLS location.

2.5.3.2.3 Correlation of Earthquakes with Capable Tectonic Sources

WLS COL FSAR Section 2.5.3.3 discusses correlation of earthquakes with capable tectonic sources within the WLS vicinity. Based on the epicentral data shown in Figure 2.5.3-1 of this report, the applicant concluded that there is no spatial correlation of earthquake epicenters with known or postulated faults or other tectonic features within the site vicinity. The applicant stated that no historical earthquakes have been associated with bedrock faults in the WLS vicinity, and concluded that none of the faults in the site vicinity are capable tectonic structures.

2.5.3.2.4 Ages of Most Recent Deformation

WLS COL FSAR Section 2.5.3.4 discusses ages of most recent deformations within the WLS vicinity. Based on geologic field relationships as discussed above in Section 2.5.3.2.2 of this report, the applicant documented a Paleozoic age for the six faults mapped in the site vicinity (i.e., the Kings Mountain shear zone; the southwestern extension of the Boogertown shear zone; an unnamed fault north of Gaffney; and the Tinsley Bridge, Brindle Creek, and Reedy River faults). The applicant reported that Garihan and others (1993) suggested the Kings Mountain shear zone may have experienced localized reactivation during Mesozoic (251 to 65.5 Ma) time. However, based on information presented in WLS COL FSAR Section 2.5.3.2 and summarized above in Section 2.5.3.2.2 of this report, no evidence exists for Quaternary (2.6 Ma to present) deformation associated with any of these structures.

2.5.3.2.5 Relationships of Site Area Tectonic Structures to Regional Tectonic Structures

WLS COL FSAR Section 2.5.3.5 addresses the relationships of the six faults identified in the site vicinity with regional tectonic structures. The applicant indicated that four of these faults (i.e., the Kings Mountain shear zone, the Tinsley Bridge fault, the southwest extension of the Boogertown shear zone, and the Reedy River thrust fault) are part of the regional Central Piedmont shear zone (CPSZ) of Paleozoic (> 251 Ma) age, which separates the Western Piedmont Zone from the Charlotte terrane of the Carolina Zone in which WLS lies. The applicant stated that the relationship of the unnamed fault north of Gaffney and the CPSZ is not clear, and concluded that none of the six faults represents a capable tectonic structure.

2.5.3.2.6 Characterization of Capable Tectonic Sources

WLS COL FSAR Section 2.5.3.6 discusses characterization of capable tectonic sources within the WLS vicinity. Based on review of published literature, interviews with technical experts knowledgeable about the geology and seismology of the site region and site vicinity, and investigations performed by the applicant for the WLS application, the applicant concluded that no evidence exists for the existence of capable tectonic sources within the WLS vicinity. The applicant noted that this interpretation is fully consistent with data derived from previous investigations performed for the CNS site.

2.5.3.2.7 Designation of Quaternary Deformation Zones in the Site Region

WLS COL FSAR Section 2.5.3.7 addresses zones of Quaternary deformation in the site region. Based on review of published literature, interviews with technical experts knowledgeable about the geology and seismology of the site region and site vicinity, and investigations performed for the WLS application, the applicant concluded that no evidence exists for zones of Quaternary deformation within the WLS region or site vicinity.

2.5.3.2.8 Potential for Surface Tectonic Deformation at the Site

WLS COL FSAR Section 2.5.3.8 addresses the potential for surface tectonic and non-tectonic deformation at the site. The applicant stated that detailed geologic mapping of the former CNS site revealed no evidence of geologically recent or active (i.e., Quaternary) faulting at WLS. Based on review of published literature, interviews with technical experts knowledgeable about the geology and seismology of the site region and site vicinity, and investigations performed for the WLS application, the applicant concluded that no Quaternary faults or capable tectonic sources occur within the WLS vicinity and the potential for surface tectonic deformation at the site is negligible.

Since lithologies in the site vicinity and site area and at the site location are metamorphosed igneous, volcanic, and sedimentary rock units, the applicant also stated that none of these rock units are susceptible to dissolution collapse or subsidence due to fluid withdrawal. The applicant indicated that no evidence exists to suggest there is any potential for non-tectonic surface deformation within the WLS area, and concluded that the potential for non-tectonic surface deformation within the site area and at the site location is negligible. The applicant evaluated the potential for RIS in WLS COL FSAR Section 2.5.2.1.3 and concluded in WLS COL FSAR Section 2.5.3.8 that the potential for non-tectonic surface deformation due to RIS is also negligible.

2.5.3.3 *Regulatory Basis*

The regulatory basis for the information incorporated by reference is addressed in NUREG-1793 and its supplements. The applicable regulatory requirements for surface faulting are as follows:

- 10 CFR 52.79(a)(1)(iii), as it relates to identifying geologic site characteristics with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity and period of time in which the historical data have been accumulated

- 10 CFR 100.23, as it relates to determining the potential for surface tectonic and non-tectonic deformation at and in the region surrounding the site

In addition, the related acceptance criteria associated with relevant requirements of NRC regulations for surface deformation are given in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” Section 2.5.3 as follows:

- Geologic, Seismic, and Geophysical Investigations: Requirements of GDC 2 in Appendix A of 10 CFR Part 50, 10 CFR 52.17(a)(1)(vi),or 10 CFR 52.79(a)(1)(iii) and 10 CFR 100.23(c) and 10 CFR 100.23(d)(2), are met and guidance in RG 1.208, and RG 4.7 followed for this area of review if discussions of Quaternary tectonics, structural geology, stratigraphy, geochronologic methods used for age dating, paleoseismology, and geologic history of the site vicinity, site area, and site location are complete, compare reasonably with studies conducted by others in the same area, and are supported by detailed investigations performed by the applicant. Site vicinity, site area, and site location-specific geologic maps and cross-sections constructed at scales adequate to clearly illustrate surficial and bedrock geology, structural geology, topography, and relationship of power plant foundations and site boundaries to these features should be included in the application. For sites located near bodies of water, the application should address how investigations have been conducted to detect possible surface deformation features that might be located beneath water.

- Geologic Evidence for Surface Deformation: Requirements of GDC 2 in Appendix A of 10 CFR Part 50, 10 CFR 52.17(a)(1)(vi) or 10 CFR 52.79(a)(1)(iii), and 10 CFR 100.23(c) and 10 CFR 100.23(d)(2), are met and guidance in RG 1.208, and RG 4.7 followed for this area of review if the applicant provides sufficient surface and subsurface information for the site vicinity, area, and location to confirm and characterize presence or absence of surface deformation (e.g., faulting, growth faulting, subsidence or collapse related to dissolution of limestone, salt or gypsum deposits, or salt diapirism and paleoliquefaction) features. The applicant should also take into account the potential for blind faults.

- Timing of Deformation: Requirements of GDC 2 in Appendix A of 10 CFR Part 50, 10 CFR 52.17(a)(1)(vi) or 10 CFR 52.79(a)(1)(iii), and 10 CFR 100.23(c) and 10 CFR 100.23(d)(2), are met for this area of review if recognized surface deformation features (e.g., tectonic faults and non-tectonic features including growth faults) and features associated with a blind fault, are investigated in sufficient detail to constrain the age of the most recent surface deformation event, and, if applicable, the ages of preceding deformation events. The application shall also provide an acceptable evaluation of sensitivity and resolution of the exploratory geologic and geophysical techniques used to determine whether or not appropriate techniques were applied to assess the age of the most recent displacement.

- Correlation of Earthquakes with Tectonic Feature: Requirements of GDC 2 in Appendix A of 10 CFR Part 50, 10 CFR 52.17(a)(1)(vi or 10 CFR 52.79(a)(1)(iii)), and 10 CFR 100.23(c) and 10 CFR 100.23(d)(2), are met for this area of review if the applicant evaluates all reported historical earthquakes within the site vicinity with respect to accuracy of hypocenter location and source of origin, and with respect to correlation to tectonic features. The applicant shall evaluate the potential for historical activity on tectonic features in the site vicinity. The application should include a plot of earthquake epicenters superimposed on a map showing tectonic features in the site vicinity.

- Relationship of Geologic Features in the Site Vicinity to Regional Geologic Features: Requirements of GDC 2 in Appendix A of 10 CFR Part 50, 10 CFR 52.17(a)(1)(vi) or 10 CFR 52.79(a)(1)(iii), and 10 CFR 100.23(c) and 10 CFR 100.23(d)(2), are satisfied for this area of review if the applicant evaluates the relationships between faults or other deformation features in the site vicinity and the regional framework. The application should provide an acceptable evaluation of the relationships between the regional (tectonic and non-tectonic) framework and deformation features in the site vicinity, including growths faults and growth fault systems. The applicant should show how this information is used in the evaluation of potential for future surface deformation at the site.

- Potential for Surface Deformation at the Site: To meet requirements of GDC 2 in Appendix A of 10 CFR Part 50, 10 CFR 52.17(a)(1)(vi) or 10 CFR 52.79(a)(1)(iii), and 10 CFR 100.23(c) and 10 CFR 100.23(d)(2), for this area of review, the applicant shall assess the potential future tectonic and nontectonic surface deformation at the site. The applicant should provide sufficient geological, seismological, and geophysical information to clearly establish whether there is a potential for future surface deformation at the site. If the potential for future surface deformation exists at the site, the application must provide information that demonstrates the potential effects of surface deformation are within the design basis of the proposed facility. NRC regulations do not restrict building in an area with surface faulting potential, but if that potential exists, the regulations require that surface deformation must be taken into account in the design

> and operation of the proposed nuclear power plant. It is questionable whether it might be feasible to design for surface deformation with any degree of confidence that safety-related structures, systems, and components would maintain their safety functions if surface displacements occur in the future. Consequently, it is NRC policy (e.g. RG 1.208) to recommend that any site located on a surface or near-surface feature with a potential for future displacement be re-located to an alternate site.

Geologic characteristics should also be consistent with the related guidance from appropriate sections of RG 1.132, Revision 2, RG 1.198, RG 1.206, and RG 1.208.

2.5.3.4 *Technical Evaluation*

The staff reviewed WLS COL FSAR Section 2.5.3 and checked the referenced DCD to ensure that the combination of information presented in the WLS COL FSAR and the AP1000 DCD completely represents the required information related to tectonic (i.e., faulting) and non-tectonic surface and near-surface deformation. The staff's review confirmed that information contained in the application or incorporated by reference addresses the information required for this review topic. NUREG-1793 and associated supplements document the results of the staff's evaluation of information incorporated by reference into the WLS application. The staff reviewed the following information in the WLS COL FSAR.

AP1000 COL Information Item

- WLS COL 2.5-4

The staff reviewed WLS COL 2.5-4 included in WLS COL FSAR Section 2.5.3. WLS COL FSAR Section 2.5.3 addresses the potential for surface or near-surface tectonic and non-tectonic deformation within the site vicinity and site area and at the site location. The COL information item from the AP1000 DCD, Section 2.5.3, states:

> Combined License applicants referencing the AP1000 certified design will address the following surface and subsurface geological, seismological, and geophysical information related to the potential for surface or near-surface faulting affecting the site: (1) geological, seismological, and geophysical investigations; (2) geological evidence, or absence of evidence, for surface deformation; (3) correlation of earthquakes with capable tectonic sources; (4) ages of most recent deformations; (5) relationship of tectonic structures in the site area to regional tectonic sources; (6) characterization of capable tectonic sources; (7) designation of zones of Quaternary deformation in the site region; and (8) potential for surface tectonic deformation at the site.

Based on the discussion of the potential for tectonic and non-tectonic surface deformation at the site presented in WLS COL FSAR Section 2.5.3, the staff concludes that the applicant provided the information required to satisfy WLS COL 2.5-4.

Section 2.5.3.2 above specifies the data sources used by the applicant to develop WLS COL FSAR Section 2.5.3, which contains information related to the potential for tectonic and non-tectonic surface and near-surface deformation within the site vicinity and site area and at the site location. Through the review of WLS COL FSAR Section 2.5.3, the staff determined whether the applicant had complied with the applicable regulations and conducted the investigations at an appropriate level of detail in accordance with RG 1.208.

As part of the technical evaluation of WLS COL FSAR Section 2.5.3, the staff visited WLS on April 27 to May 2, 2008 (ML120320233); January 27 and 28, 2009 (ML120320258); July 12 through 14, 2011 (ML11216A256); and February 10, 2014 (ML14126A584), to interact with the applicant and its consultants in regard to the geologic, seismic, geophysical, and geotechnical investigations being conducted to characterize the site. Technical experts from the USGS accompanied the staff during the January 2009 site visit to assist with evaluation of geologic and seismic data collected by the applicant. On February 23 and July 20, 2009 (ML120270235), the staff audited information used by the applicant to constrain timing of development of minor ductile and brittle shear zones mapped in the excavations at the original CNS site. The staff also audited Revisions 0 and 1 of the report discussing the compiled geologic map for the CNS Unit 1 (i.e., WLS Unit 1) excavation on June 6, 9, and 10, 2011 and October 25 through 27, 2011 (ML120270246). The Duke report documented geologic characteristics of foundation grade level bedrock at WLS Unit 1 based on previous geologic mapping of foundation grade bedrock conducted in the former co-located CNS Unit 1 excavation, which lies under concrete poured during initial construction activities at the CNS site. Section 2.5.1.4 of this report presents details about the site visits and data documentation audits that enabled staff to assess conclusions made by the applicant regarding the potential for surface and near-surface tectonic and non-tectonic deformation at WLS and confirm that no geologic features occurring in foundation bedrock represent capable tectonic structures.

Sections 2.5.3.4.1 through 2.5.3.4.8 of this report present the staff's evaluation of information provided by the applicant in WLS COL FSAR Section 2.5.3 and the applicant's responses to RAIs for WLS COL FSAR Section 2.5.3. In addition to RAIs addressing specific technical issues related to tectonic and non-tectonic surface and near-surface deformation at the site, discussed in detail below, the staff also prepared editorial RAIs to further clarify certain descriptive statements made by the applicant in the WLS COL FSAR and to qualify geologic features illustrated in WLS COL FSAR figures. This detailed technical evaluation does not discuss these editorial RAIs.

2.5.3.4.1 Geologic, Seismic, and Geophysical Investigations

WLS COL FSAR Section 2.5.3.1 describes geologic, seismic, and geophysical investigations performed by the applicant to assess the potential for tectonic (i.e., due to faulting) and non-tectonic surface and near-surface deformation within the site vicinity and site area, as well as the potential for surface fault rupture at the WLS location. Based on these investigations, the applicant concluded that no evidence exists for active or geologically recent (i.e., Quaternary age, 2.6 Ma to present) faulting within the site vicinity and site area or at the site location. Sections 2.5.3.4.1.1 through 2.5.3.4.1.4 of this report present the staff's evaluations of the geologic, seismic, and geophysical investigations performed by the applicant for WLS.

2.5.3.4.1.1 *Previous Site Investigations and Published Geologic Mapping*

WLS COL FSAR Sections 2.5.3.1.1 and 2.5.3.1.2 discuss, respectively, previous WLS area and site location investigations and published geologic maps for the site vicinity and site area. The applicant concluded that no evidence exists for active or geologically recent (i.e., Quaternary, 2.6 Ma to present) faulting or shearing in the site area, but did not summarize information in the WLS COL FSAR to qualify this conclusion.

The staff focused the review of WLS COL FSAR Sections 2.5.3.1.1 and 2.5.3.1.2 on documentation of the sources used by the applicant to conclude that previous detailed geologic mapping at the CNS site revealed no evidence for active or geologically recent faulting, and on clarifying the timing of deformation along the brittle shear zones that occur at the margins of mafic dikes that intrude foundation unit Zto at the site location. In RAI 26, Question 02.05.03-1, the staff requested that the applicant summarize data derived from geologic investigations performed at the CNS site used to conclude that no evidence exists for active or geologically recent faulting in the site area. In RAI 85, Question 02.05.03-11, the staff requested that the applicant provide the information considered for constraining time of development of the brittle shear zones observed along the margins of mafic intrusive dikes at the site location and documenting that the zones are much older than Quaternary.

In response to RAI 26, Question 02.05.03-1, the applicant summarized the information acquired at the CNS site and re-evaluated during geologic characterization of WLS. The applicant stated that radiometric age dates on deformed and undeformed mineral samples, collected adjacent to or within minor faults and shear zones at the CNS site, confirmed that no fault movement or shearing younger than Mesozoic (251 to 65.5 Ma) had occurred at the site. The applicant reported that geologic investigations performed for the WLS application, including the confirmatory mapping of lithologies and geologic features in the northern portion of the CNS Unit 2 foundation excavation and comparison of those mapping results with the original geologic map prepared for CNS Unit 2, supported the interpretation of a pre-Quaternary (> 2.6 Ma) age for tectonic deformation features at WLS. Therefore, the applicant concluded that the historical CNS geologic map data can be used for characterization of foundation grade level bedrock at WLS to establish a pre-Quaternary age for the latest shearing and faulting observed at the site.

In response to RAI 85, Question 02.05.03-11, the applicant cross-referenced WLS COL FSAR Section 2.5.1.2.5.4, which presents radiometric age dates that constrain timing of development of the shear zones to be no younger than Mesozoic (251 to 65.6 Ma) based on the presence of undeformed potassium feldspar minerals in veins that cross-cut the shear zones. The applicant also indicated that the shear zones may be older than 300 Ma based on the presence of undeformed post-tectonic metamorphic minerals. Therefore, the applicant concluded that the shear zones are much older than Quaternary and do not reflect active or recent faulting at WLS.

The staff concludes that the shear zones are minor and much older than Quaternary, and that no evidence exists for active or geologically recent faulting or shearing in the site area or at the site location. The staff makes these conclusions based on information derived from the following activities: review of WLS COL FSAR Sections 2.5.3.1.1 and 2.5.3.1.2 and the applicant’s responses to RAI 26, Question 02.05.03-1 and RAI 85, Question 02.05.03-11; direct field observation of minor shear zones in the CNS excavations during site visits in April-May 2008 (ML120320233), January 2009, and July 2011 (ML11216A256) , including the southeastern end of a prominent, northwest-trending shear zone in the excavation for CNS Unit 2 (originally mapped as Shear Zone 6 at the CNS site) that extends beneath the concrete covering the CNS Unit 1 (i.e., WLS Unit 1) excavation; direct examination of core from the relocated footprints of WLS Units 1 and 2 in February 2014; and review of radiometric age dates and the compiled geologic map prepared by Duke for CNS/WLS Unit 1, including data specifically related to Shear Zone 6, during data documentation audits in June and October 2011. Accordingly, the staff considers RAI 26, Question 02.05.03-1 and RAI 85, Question 02.05.03-11 resolved.

2.5.3.4.1.2 *Current Geologic Mapping*

WLS COL FSAR Section 2.5.3.1.3 describes current geologic mapping in the site vicinity and site area and at the site location. The applicant stated that field investigations conducted for the WLS application enabled refinement of previous geologic maps and provided support for the conclusions that foundation grade level bedrock in CNS Unit 1 (WLS Unit 1) and CNS Unit 2 does not contain capable tectonic features or exhibit any evidence to suggest a potential for tectonic or non-tectonic surface or near-surface deformation. The applicant did not summarize the map refinements or present the refined geologic maps in the WLS COL FSAR.

The staff focused the review of WLS COL FSAR Section 2.5.3.1.3 on clarifying the refinements made by the applicant to previous geologic maps based on field investigations conducted for the WLS application. In RAI 26, Question 02.05.03-4, the staff requested that the applicant summarize the geologic map refinements and present them in the WLS COL FSAR. In response to RAI 26, Question 02.05.03-4, the applicant explained the refinements made to geologic maps previously presented in the WLS COL FSAR, which contain the western boundary of the metamorphosed pluton that comprises foundation unit Zto. The applicant documented that this boundary is an intrusive contact separating the plutonic rock body from the surrounding metamorphosed country rock based on field data derived from borings, test pits, and trenches located to investigate the geologic characteristics of the boundary. The applicant used these data to conclude that the contact boundary is more irregular than initially shown on earlier geologic maps and does not exhibit any characteristics indicative of faulting and revised WLS COL FSAR Figure 2.5.1-226 to more clearly illustrate locations of the field data sources.

Based on review of WLS COL FSAR Section 2.5.3.1.3 and the applicant’s response to RAI 26, Question 02.05.03-4, particularly the definitive field evidence derived from borings, test pits, and trenches used to investigate geologic characteristics of the western boundary of the pluton comprising foundation unit Zto, the staff concludes that the western boundary of the pluton is an intrusive contact with the surrounding county rock and not a fault. The staff notes that the applicant’s revisions to WLS COL FSAR Figure 2.5.1-226 reinforce the interpretation that the western margin of the pluton is irregular, rather than planar, and not a fault. The staff draws this conclusion based on the field data that show no evidence for faulting along the western boundary of the pluton that forms foundation unit Zto. Accordingly, the staff considers RAI 26, Question 02.05.03-4 resolved.

2.5.3.4.1.3 *Previous and Current Seismicity Data*

WLS COL FSAR Sections 2.5.3.1.4 and 2.5.3.1.5 discuss previous and current seismicity within the WLS vicinity, respectively. The staff focused the review of WLS COL FSAR Section 2.5.3.1.5 on understanding the age of last deformation along the EPFS and on statements made by the applicant that two September 2006 earthquakes with epicenters about 121 km (75 mi) east-southeast of WLS may have been related to either the EPFS or a Mesozoic extensional basin proposed by Benson to underlie Coastal Plain sediments. In RAI 26, Question 02.05.03-5 and RAI 85, Question 02.05.03-12, the staff requested that the applicant locate the EPFS on an appropriate map in the WLS COL FSAR; to document the age of last movement on this proposed fault system, including assessment of the interpretation by Nystrom that post-Miocene (< 5.3 Ma) displacement may have occurred along it; and to discuss the possible spatial correlation of the two earthquakes with the EPFS, as suggested by Talwani, and with the buried Mesozoic basin defined by Benson.

In response to RAI 26, Question 02.05.03-5 and RAI 85, Question 02.05.03-12, the applicant modified WLS COL FSAR Figures 2.5.1-209 and 2.5.1-210 to clearly show the location of the multiple segments comprising the EPFS. The applicant stated that field evidence generally constrains timing of last movement on the EPFZ to Paleozoic (i.e., > 251 Ma), since a granitic intrusive body, located in Georgia and dated at 269 Ma, and Mesozoic dikes (suggesting an age > 65.5 Ma) in South Carolina cross-cut different segments of the EPFS without any deformation. The applicant reported that Nystrom (2006) proposed local reactivation of some segments of the EPFS during the Cenozoic (65.5 Ma to present) based on inferred offset of mapped geologic units. The applicant noted that Nystrom also identified silicified breccias along some parts of the EPFS, a field relationship generally characteristic of Mesozoic (251 to 65.5 Ma) faulting in the Piedmont. The applicant summarized the existing field data, which documents that some of the offsets interpreted by Nystrom (2006) to suggest possible post-Miocene reactivation of the EPFS have not been observed along all segments of the ECFS and that the outcrop patterns of some geologic units mapped by Nystrom (2006) actually preclude Cenozoic movement along the entire length of the EPFS. The applicant stated that no definitive evidence exists to demonstrate that the two September 2006 earthquakes show any causative relationship with the EPFS or the Mesozoic basin proposed by Benson (1992) in light of the lack of focal mechanism data for these seismic events and the large uncertainty associated with their epicentral locations. Based on these collective data, the applicant concluded that the EPFS is most likely not younger than Paleozoic in age, and that no definitive evidence exists for any relationship between the two September 2006 earthquakes and either the EPFS or the buried Mesozoic extensional basin.

Based on the review of WLS COL FSAR Sections 2.5.3.1.4 and 2.5.3.1.5, the applicant's responses to RAI 26, Question 02.05.03-5 and RAI 85, Question 02.05.03-12, and an independent examination of the information sources discussed by the applicant in the WLS COL FSAR and the RAI responses, the staff concludes that no existing data suggest the two small September 2006 earthquakes resulted from displacement along either the EPFS or a fault bounding the proposed buried Mesozoic basin. The staff draws this conclusion because of the large uncertainty in the epicentral locations of these two earthquakes, making even a spatial correlation between them and either of the structures difficult to establish. The staff also concludes that existing field data do not indicate Cenozoic (65.5 Ma to present), including post-Miocene (< 5.3 Ma), deformation characterizes the entire length of the EPFS, but rather show that this fault system is generally not younger than Mesozoic (> 65.5 Ma), because the granite body in Georgia dated at 269 Ma and Mesozoic dikes in South Carolina dated at > 65.5 Ma cross-cut the EPFS without any deformation. Accordingly, the staff considers RAI 26, Question 02.05.03-5 and RAI 85, Question 02.05.03-12 resolved.

2.5.3.4.1.4 *Current Aerial and Field Reconnaissance*

WLS COL FSAR Section 2.5.3.1.6 discusses current aerial and field reconnaissance conducted by the applicant to assess the presence of active tectonic surface deformation in WLS area. The applicant concluded that no geomorphic evidence exists for active faulting in the site area, including along Lineament 1, the northeast-trending topographic feature that lies 3.2 km (2 mi) northwest of the site location.

The staff focused the review of WLS COL FSAR Section 2.5.3.1.6 on documenting a non-tectonic origin for Lineament 1. In RAI 26, Question 02.05.03-6, the staff requested that the applicant summarize the information used to conclude that Lineament 1 is the result of erosion. In response to RAI 26, Question 02.05.03-6, the applicant stated that previous geologic maps show no faults associated with Lineament 1; field reconnaissance conducted for WLS did not reveal any evidence for faulting along the lineament; erosion-resistant metasedimentary rocks cap the ridges lying adjacent to the lineament; streams flowing along the base of the ridges result in strong topographic enhancement of the lineament; and the lineament does not extend across the Broad River to the northeast. Based on these field observations, the applicant concluded that Lineament 1 is non-tectonic in character and the result of localized stream erosion controlled by erosion-resistant, ridge-forming metasedimentary rock units.

Based on review of the applicant’s response to RAI 26, Question 02.05.03-6 and field observations made during site visits in April-May 2008 (ML120320233) and January 2009 (ML120320258), the staff concludes that Lineament 1 is the result of erosion rather than faulting. The staff draws this conclusion because field evidence demonstrates that Lineament 1 reflects localized stream erosion controlled by a resistant ridge-forming quartzite and does not extend across the Broad River to the northeast. Accordingly, the staff considers RAI 26, Question 02.05.03-6 resolved.

Based on the review of WLS COL FSAR Sections 2.5.3.1.1 through 2.5.3.1.6 and the applicant’s responses to RAI 26, Questions 02.05.03-1, 02.05.03-4 through 02.05.03-6, RAI 85, Question 02.05.03-11 and Question 02.05.03-12, as well as field observations made during the site visits, the staff finds that the applicant presented thorough and accurate descriptions of the geologic, seismic, and geophysical investigations performed to assess the potential for tectonic and non-tectonic surface and near-surface deformation within the site vicinity and site area and at the site location in support of the WLS application.

2.5.3.4.2 Geologic Evidence, or Absence of Evidence, for Surface Deformation

WLS COL FSAR Section 2.5.3.2 discusses geologic evidence, or lack of evidence, for surface deformation within the WLS vicinity and at the site location. Based on field relationships and radiometric age dates, the applicant concluded that the six bedrock faults reported to occur in the site vicinity, shown in Figure 2.5.3-1 of this report, are Paleozoic (> 251 Ma) in age, and that no faults oriented parallel to the northeastern trend of regional tectonic structures extend through the WLS location because good lithologic strain markers show no deformation related to faulting.

The staff focused the review of WLS COL FSAR Section 2.5.3.2 on the evidence presented by the applicant to document that the six faults mapped in the vicinity of WLS, including those that strike toward the site location, do not exhibit any evidence for Quaternary surface or near-surface tectonic deformation. Based on review of WLS COL FSAR Section 2.5.3.2, independent examination of references cited by the applicant, and direct observations made during site field audits conducted in April-May 2008 (ML120320233), January 2009 (ML120320258), July 2011 (ML11216A256), and February 2014 (ML14126A584), the staff concludes that these faults are all pre-Quaternary in age and do not exhibit any characteristics suggesting a potential for Quaternary surface or near-surface tectonic deformation at WLS. The staff draws this conclusion based on the field evidence and radiometric age dates presented by the applicant that limit the timing of displacements on these tectonic features to pre-Quaternary (> 2.6 Ma).

Based on the review of WLS COL FSAR Section 2.5.3.2, independent examination of references cited by the applicant, and direct observations made during site field audits, the staff finds that the applicant presented thorough and accurate descriptions of information related to geologic evidence, or lack of evidence, for surface deformation due to faulting within the site vicinity and at the site location in support of the WLS application.

2.5.3.4.3 Correlation of Earthquakes with Capable Tectonic Sources

WLS COL FSAR Section 2.5.3.3 addresses the correlation of earthquakes with capable tectonic sources in the WLS vicinity. The applicant concluded that there is no correlation of earthquakes with capable tectonic sources in the WLS vicinity.

The staff focused the review of WLS COL FSAR Section 2.5.3.3 on information used by the applicant to document that there is no correlation of earthquakes with capable tectonic sources in the WLS vicinity. Based on review of WLS COL FSAR Section 2.5.3.3 and an independent examination of data sources referenced in WLS COL FSAR Section 2.5.2, the staff concludes that no correlation exists between earthquakes and capable tectonic sources in the WLS vicinity because no data demonstrate any correlation.

Based on the review of WLS COL FSAR Section 2.5.3.3 and an independent examination of references cited by the applicant in WLS COL FSAR Section 2.5.2, the staff finds that the applicant presented a thorough and accurate description of information related to correlation of earthquakes with capable tectonic sources in support of the WLS application.

2.5.3.4.4 Ages of Most Recent Deformations

WLS COL FSAR Section 2.5.3.4 discusses ages of most recent deformations in the vicinity of WLS. The applicant concluded that the six faults mapped within the site vicinity, shown in Figure 2.5.3-1 of this report, only exhibit evidence for pre-Quaternary (> 2.6 Ma) deformation.

The staff focused the review of WLS COL FSAR Section 2.5.3.4 on information used by the applicant to document that all faults mapped within the WLS vicinity are pre-Quaternary in age. Based on review of WLS COL FSAR Section 2.5.3.4 and related information presented in WLS COL FSAR Section 2.5.3.2, as well as independent examination of references cited by the applicant and site visits conducted in April-May 2008 (ML120320233), January 2009 (ML120320258), July 2011 (ML11216A256), and February 2014 (ML14126A584), the staff concludes that tectonic deformation in the site vicinity is pre-Quaternary in age. The staff draws this conclusion because field data presented in the WLS COL FSAR and field observations made by staff during the site visits document a lack of Quaternary deformation in the WLS vicinity and site area and at the site location.

Based on the review of WLS COL FSAR Section 2.5.3.4, independent examination of references cited by the applicant, and field observations made during site visits, the staff finds that the applicant presented a thorough and accurate description of ages of most recent deformations in support of the WLS application.

2.5.3.4.5 Relationships of Site Area Tectonic Structures to Regional Tectonic Structures

WLS COL FSAR Section 2.5.3.5 discusses the relationships of tectonic structures in the site area to regional tectonic structures. The applicant stated that four of the six faults mapped in the site vicinity (specifically the Kings Mountain shear zone, the southwestern extension of the Boogertown shear zone, and the Tinsley Bridge and Reedy River faults shown in Figure 2.5.3-1 of this report) are segments of the regional CPSZ. Based on constraining age dates and observed field relationships summarized in WLS COL FSAR Section 2.5.3.2, which document that the CPSZ and these four tectonic structures are Paleozoic (> 521 Ma) in age, the applicant concluded that none of the structures are capable tectonic features.

The staff focused the review of WLS COL FSAR Section 2.5.3.5 on the information used by the applicant to document that four of the six faults mapped within the WLS vicinity and site area are segments of the regional CPSZ, which is Paleozoic in age. The staff concludes that these faults are not capable tectonic features based on the Paleozoic age constraint for the faults and the CPSZ as summarized in WLS COL FSAR Section 2.5.3.2.

Based on the review of WLS COL FSAR Section 2.5.3.5, the staff finds that the applicant presented a thorough and accurate description of the relationship of tectonic structures in the site area to regional tectonic structures in support of the WLS application.

2.5.3.4.6 Characterization of Capable Tectonic Sources

WLS COL FSAR Section 2.5.3.6 addresses the characterization of capable tectonic sources within the WLS vicinity. Based on constraining age dates and observed field relationships summarized in WLS COL FSAR Section 2.5.3.2, the applicant concluded that no evidence exists for any capable tectonic sources in the vicinity of WLS.

The staff focused the review of WLS COL FSAR Section 2.5.3.6 on the information used by the applicant to document that no capable tectonic sources exist in the site vicinity. Based on the constraining age dates and observed field relationships summarized in WLS COL FSAR Section 2.5.3.2, as well as field observations made during site visits in April-May 2008 and January 2009, the staff concludes that no capable tectonic sources exist in WLS vicinity. The staff draws this conclusion because the age dates and observed field relationships document a lack of capable tectonic sources in the site vicinity. The staff notes the suggestion of Garihan and others (1993) that the Kings Mountain shear zone may have experienced localized Mesozoic (251 to 65.5 Ma) reactivation does not alter the conclusion about the absence of capable tectonic sources in the site vicinity.

Based on the review of WLS COL FSAR Section 2.5.3.6 and field observations made during the site visits, the staff finds that the applicant presented a thorough and accurate description of information related to characterization of capable tectonic sources in support of the WLS application.

2.5.3.4.7 Designation of Zones of Quaternary Deformation in the Site Region

WLS COL FSAR Section 2.5.3.7 discusses the designation of Quaternary (2.6 Ma to present) deformation zones in the WLS region. Based on information presented in WLS COL FSAR Section 2.5.1 and cross-referenced in WLS COL FSAR Section 2.5.3.7, the applicant concluded that no zones of Quaternary deformation requiring further investigation occur in the site region.

The staff focused the review of WLS COL FSAR Section 2.5.3.7 on the data presented by the applicant, mainly in WLS COL FSAR Section 2.5.1, to document that no zones of Quaternary deformation occur in the site region. The staff concludes that no evidence exists for zones of Quaternary deformation in the site region based on independent review of the data sources presented by the applicant in the WLS COL FSAR, which document a lack of Quaternary deformation in the site region.

Based on the review of WLS COL FSAR Section 2.5.3.7 and an independent review of references cited by the applicant, the staff finds that the applicant presented a thorough and accurate description of information related to designation of zones of Quaternary deformation in the site region in support of the WLS application.

2.5.3.4.8 Potential for Surface Tectonic Deformation at the Site

WLS COL FSAR Section 2.5.3.8 discusses the potential for surface and near-surface tectonic and non-tectonic deformation at WLS. The applicant documented that no Quaternary (2.6 Ma to present) faults or capable tectonic structures occur in the site vicinity or site area, and that rock units in the site area and at the site location are not susceptible to dissolution collapse or subsidence related to fluid withdrawals. Therefore, the applicant concluded that the potential for tectonic and non-tectonic surface and near-surface deformation is negligible at the site location.

The staff focused the review of WLS COL FSAR Section 2.5.3.8 on information presented by the applicant to document that no evidence exits to indicate any potential for surface or near-surface tectonic or non-tectonic deformation at WLS. The staff concludes that the potential for surface or near-surface tectonic or non-tectonic deformation at WLS is negligible based on results of the following activities: Review of WLS COL FSAR Section 2.5.3.8; examination of lithologic units and tectonic features in the CNS excavations during site visits in April-May 2008, January 2009, July 2011, and February 2014; data documentation audits conducted in February and July 2009 and June and October 2011; and an independent review of published sources cited by the applicant. The staff draws this conclusion because direct field observations made by staff during site visits, the data documentation audits, and the independent review of the information sources cited by the applicant strongly support it.

Based on the review of WLS COL FSAR Section 2.5.3.8, independent review of published sources cited by the applicant, results of the data documentation audits, and field observations made during the site visits, the staff finds that the applicant presented a thorough and accurate description of the potential for surface tectonic deformation at the site in support of the WLS application.

With regard to evaluating the potential for surface and near-surface tectonic and non-tectonic deformation in foundation bedrock units at WLS Unit 1, (i.e., former CNS Unit 1, the excavation for which lies under concrete poured during initial CNS construction activities), the staff deemed it necessary to document lithologies and geologic features that occur in foundation grade level bedrock in WLS Unit 1/CNS Unit 1 excavation. To accomplish this task, the staff performed the June and October 2011 data documentation audits described in Section 2.5.1.4 of this report specifically to review the report containing the compiled geologic map and associated data for CNS Unit 1. The staff also conducted the site visit in July 2011, as described in Section 2.5.1.4 of this report, to directly examine existing exposures in CNS Unit 2 and CNS Unit 3 (WLS Unit 2) for determining whether lithologies and geologic features mapped in the CNS Unit 1 (WLS Unit 1) excavation are similar to the rock units and geologic features that occur in the existing adjacent CNS site excavations. Based on the results of these activities, specifically focused on assessment of the compiled geologic map data presented in the Duke report and with due consideration for results of the data documentation audits performed in February and July 2009 and the site visits conducted in April-May 2008 and January 2009, the staff concludes that the applicant essentially satisfied the requirements for a geologic mapping License Condition for WLS Unit 1,. The staff makes this conclusion because field relationships observed during all site visits and field data reviewed during all data documentation audits fully support the premise of similarity of lithologies and geologic features in excavations for CNS Unit 1, Unit 2, and Unit 3, and, consequently, for WLS Unit 1,. Also, in support of the staff's conclusion regarding how the applicant satisfied the requirements for a geologic mapping License Condition for WLS Unit 1, the site visits and data documentation audits confirm that no capable tectonic structures or other potentially detrimental geologic features occur in the foundation rocks of WLS Unit 1, and that the potential for surface or near-surface tectonic and non-tectonic deformation at WLS Unit 1, is negligible.

Although the staff concludes that the applicant essentially satisfied the requirements for a geologic mapping License Condition for WLS Unit 1, based on all geologic data examined and the direct field observations made, it is the responsibility of the applicant to perform detailed geologic mapping of the WLS Unit 2 excavation for nuclear island structures. Section 2.5.3.5 of this report defines this responsibility as License Condition (2-2).

2.5.3.5 *Post-Combined License Activities*

The staff identified the following geologic mapping License Condition in Section 2.5.3.4.8 of this report as the responsibility of the COL licensee:

- License Condition (2-2) – The licensee shall perform detailed geologic mapping of the excavation for WLS Unit 2 nuclear island structures; examine and evaluate geologic features discovered in excavations for safety-related structures other than those for the Unit 2 nuclear island; and notify the Director of the Office of New Reactors, or the Director's designee, once excavations for Unit 2 safety-related structures are open for examination by the staff.

2.5.3.6 *Conclusions*

The staff reviewed the WLS COL FSAR and the referenced DCD. Based on these reviews, the staff confirmed that the applicant addressed the required information related to surface and near-surface tectonic (i.e., due to faulting) and non-tectonic deformation, and there is no additional outstanding information related to these types of deformation that must be discussed in the WLS COL FSAR. NUREG-1793 and associated supplements document the results of the staff's technical evaluation of information incorporated by reference into the WLS application.

As set forth above, the staff has reviewed the information in COL Information Item WLS COL 2.5-4 and concludes that the applicant thoroughly characterized the potential for surface and near-surface tectonic and non-tectonic deformation at WLS, as required by 10 CFR 10.23 and 10 CFR 52.79(a)(1)(iii). Based on the applicant's geologic investigations performed for the site vicinity, site area, and site location, the staff also concludes that the applicant properly addressed information related to the following specific topics for WLS: Geologic, seismic, and geophysical investigations; geologic evidence, or absence of evidence, for surface deformation; correlation of earthquakes with capable tectonic sources; ages of most recent deformations; relationships between tectonic structures in the site area and regional tectonic structures; characterization of capable tectonic sources; designation of zones of Quaternary (2.6 Ma to present) deformation in the site region; and the potential for surface tectonic and non-tectonic deformation at the site. In addition, the staff concludes that the applicant performed all investigations in accordance with 10 CFR 100.23 and 10 CFR 52.79(a)(1)(iii) and followed guidance provided in RG 1.208. Finally, the staff concludes that the applicant established an adequate basis to state that no known capable tectonic sources exist which could cause surface or near-surface tectonic deformation at WLS and no known sources exist for non-tectonic deformation at the site. Therefore, the staff finds that the proposed WLS site is acceptable in regard to surface and near-surface tectonic and non-tectonic deformation and meets the requirements of 10 CFR 10.23 and 10 CFR 52.79(a)(1)(iii).

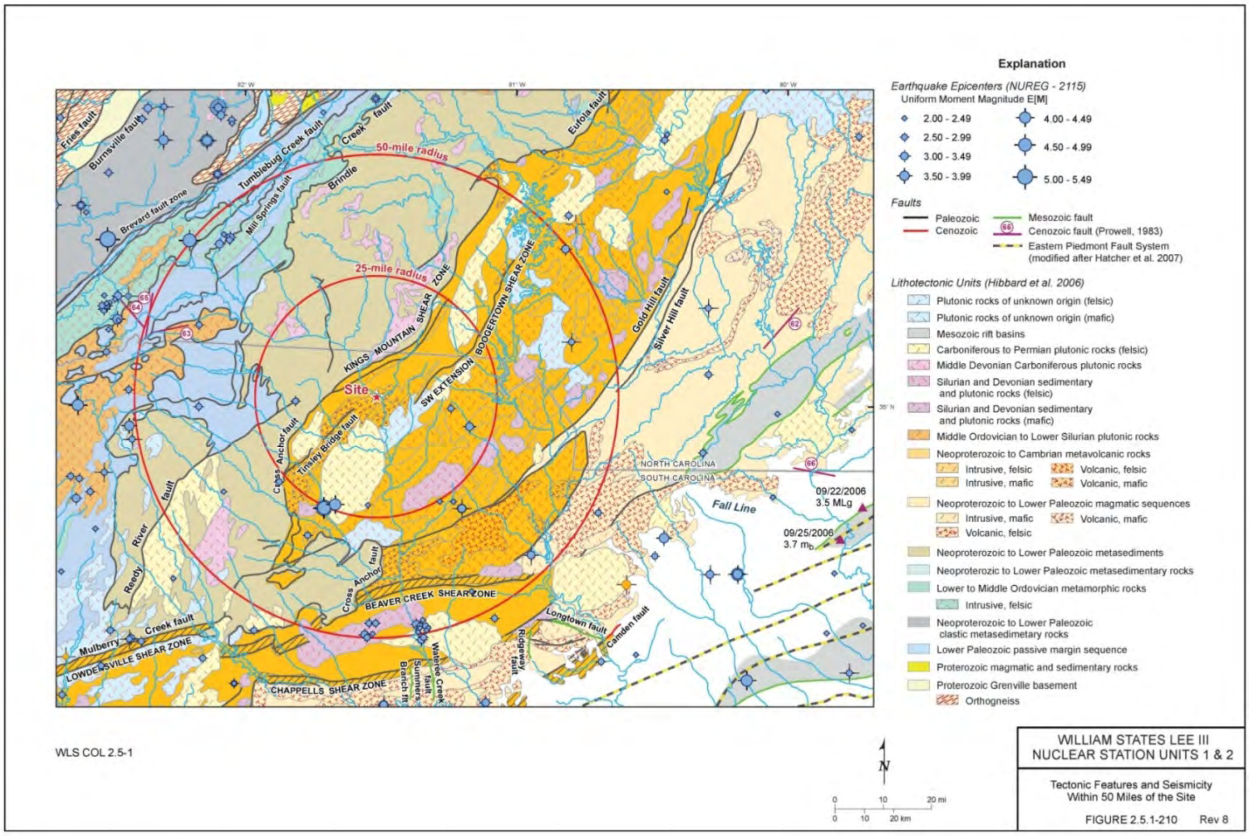


Figure 2.5.3-1 Tectonic Features And Seismicity Within 80.5 km (50 mi) of WLS (Reproduced from WLS COL FSAR Revision 8, Figure 2.5.1-210)

2.5.4 Stability of Subsurface Materials and Foundations

2.5.4.1 *Introduction*

WLS COL FSAR Section 2.5.4 presents the applicant's evaluation of the stability of subsurface materials and foundations that relate to the WLS site. The properties and stability of the soil and rock underlying the site are important to the safe design and siting of the plant. The information provided by the applicant in WLS COL FSAR Section 2.5.4 addresses: (1) geologic features in the site vicinity; (2) static and dynamic engineering properties of soil and rock strata underlying the site; (3) the interface between the foundations for safety-related facilities and the engineering properties of underlying materials; (4) results of the geophysical surveys performed to provide the velocity profiles underneath foundation footprints; (5) excavation and backfill plans and engineered earthwork analysis and criteria for safety-related structure foundations; (6) groundwater conditions at the site and associated testing monitoring and analyses; (7) responses of site soils or rocks to dynamic loading; (8) soil liquefaction potential analysis and evaluation; (9) earthquake design bases; (10) results of investigations and analyses conducted to determine foundation material stability, deformation, and settlement under static and dynamic loading conditions; (11) criteria, references, and design methods used in static and dynamic analyses of subsurface materials and foundation stability; (12) techniques and specifications to improve subsurface conditions, which are to be used at the site to provide suitable foundation conditions; and any additional information deemed necessary in accordance with 10 CFR Part 52.

2.5.4.2 *Summary of Application*

WLS COL FSAR Section 2.5.4 incorporates by reference AP1000 DCD, Revision 19, Section 2.5.4.

In addition, in WLS COL FSAR Section 2.5.4, the applicant provided site-specific information to address the following:

AP1000 COL Information Items

- WLS COL 2.5-1

The applicant provided additional information in WLS COL 2.5-1, to resolve COL Information Item 2.5-1. WLS COL 2.5-1 addresses the provision of regional and site-specific geological, seismological, and geophysical information, as well as conditions caused by human activities; structural geology; seismicity of the site; geological history; evidence of paleoseismicity; site stratigraphy and lithology; engineering significance of geological features; site groundwater conditions; dynamic behavior during prior earthquakes; zones of alteration, irregular weathering, or structural weakness; unrelieved residual stresses in bedrock; materials that could be unstable because of mineralogy or unstable physical properties; and effect of human activities in the area. This COL information item is also addressed in WLS COL FSAR Sections 2.5.1, 2.5.4.1, 2.5.4.3, 2.5.4.3.3, 2.5.4.3.5 and 2.5.4.8, Appendix 2AA, Appendix 2BB.

- WLS COL 2.5-2

The applicant provided additional information in WLS COL 2.5-2, to resolve COL Information Item 2.5-2. WLS COL 2.5-2 addresses the provision for the collection of site-specific geological, seismological, and geotechnical data and demonstration that the proposed site meets the specified requirements. This COL information item is addressed in WLS COL FSAR Sections 2.5, 2.5.2, and 2.5.4.3.3.

- WLS COL 2.5-3

The applicant provided additional information in WLS COL 2.5-2, to resolve COL Information Item 2.5-3. WLS COL 2.5-3 addresses the provision for the performance of site-specific geological, seismological, and geotechnical evaluation; if the site-specific spectra at foundation level exceed the response spectra in AP1000 DCD Figures 3.7.1-1 and 3.7.1-2 at any frequency, or if soil conditions are outside the range evaluated for AP1000 design certification. This COL information item is addressed in WLS COL FSAR Section 2.5.4.3.3.

- WLS COL 2.5-5

The applicant provided additional information in WLS COL 2.5-5, to resolve COL Information Item 2.5-5. WLS COL 2.5-5 addresses the provision of site-specific information underlying site conditions and geologic features, including site topographic features and locations of Seismic Category I structures. This COL information item is addressed in WLS COL FSAR Sections 2.5.4.1, 2.5.4.3.5, and 2.5.4.5.

- WLS COL 2.5-6

The applicant provided additional information in COL Information Item 2.5-6, resolve WLS COL 2.5-6. WLS COL 2.5-6 addresses the provision that the properties of the foundation materials are within the range of the properties considered for the design of the nuclear island basemat. This COL information item is addressed in WLS COL FSAR Sections 2.5.4.2, 2.5.4.2.1, 2.5.4.3.6, 2.5.4.4, 2.5.4.5.1, 2.5.4.5.2, 2.5.4.5.3, 2.5.4.7, 2.5.4.9, and 2.5.4.10.

- WLS COL 2.5-7

The applicant provided additional information in WLS COL 2.5-7, to resolve COL Information Item 2.5-7. WLS COL 2.5-7 addresses the provision of data concerning the extent (horizontally and vertically) of all Seismic Category (SC) I excavations, fills, and slopes, conditions and geologic features. This COL information item is addressed in WLS COL FSAR Section 2.5.4.3.6, 2.5.4.5.1, 2.5.4.5.2, and 2.5.4.5.3.

- WLS COL 2.5-8

The applicant provided additional information in WLS COL 2.5-8, to resolve COL Information Item 2.5-8. WLS COL 2.5-8 addresses the provision of addressing groundwater conditions relative to the foundation stability of the safety-related structures at the site. This COL information item is addressed in WLS COL FSAR Section 2.5.4.5.4 and 2.5.4.6.4.

- WLS COL 2.5-9

The applicant provided additional information in WLS COL 2.5-9, to resolve COL Information Item 2.5-9. WLS COL 2.5-9 addresses the provision of demonstration that the potential for liquefaction is negligible. This COL information item is addressed in WLS COL FSAR Section 2.5.4.8.

- WLS COL 2.5-10

The applicant provided additional information in WLS COL 2.5-10, to resolve COL Information Item 2.5-10. WLS COL 2.5-10 addresses the verification that the site-specific allowable soil bearing capacities are equal to or greater than the values documented in the standard design, or will provide a site-specific evaluation as described in AP1000 DCD Section 2.5.4.2 under all combined loads, including the safe shutdown earthquake, for static and dynamic loads. This COL information item is addressed in WLS COL FSAR Section 2.5.4.10.1.

- WLS COL 2.5-11

The applicant provided additional information in WLS COL 2.5-11, to resolve COL Information Item 2.5-11. WLS COL 2.5-11 addresses the design of static and dynamic lateral earth pressures and hydrostatic pressures acting on safety-related facilities using site-specific seismic loading and soil parameters. This COL information item is addressed in WLS COL FSAR Section 2.5.4.10.3

- WLS COL 2.5-12

The applicant provided additional information in WLS COL 2.5-12, to resolve COL Information Item 2.5-12. WLS COL 2.5-12 addresses soil characteristics affecting the stability of the nuclear island including foundation rebound, settlement and differential settlement. This COL information item is addressed in WLS COL FSAR Section 2.5.4.10.2.

- WLS COL 2.5-13

The applicant provided additional information in WLS COL 2.5-13, to resolve COL Information Item 2.5-13. WLS COL 2.5-13 addresses the provision of data on instrumentation, if any, proposed for monitoring the performance of the foundations of the nuclear island; including the type, location, and purpose of each instrument, as well as significant details of installation methods, along with the location and installation procedures for permanent benchmarks and markers for monitoring settlement. This COL information item is addressed in WLS COL FSAR Section 2.5.4.10.

- WLS COL 2.5-16

The applicant provided additional information in WLS COL 2.5-16, to address COL Information Item 2.5-16. WLS COL 2.5-16 addresses the provision of data on short-term (elastic) and long-term (heave and consolidation) settlement for soil sites for the history of loads imposed on the foundation consistent with the construction sequence. The resulting time-history of settlements includes construction activities such as dewatering, excavation, bearing surface preparation, placement of the basemat, and construction of the superstructure. This COL information item is addressed to the extent that the applicant stated that this is a rock site.

In addition, this WLS COL FSAR section addresses AP1000 DCD Interface Item 2.12 related to the seismic parameters peak ground acceleration, response spectra and shear wave velocity and Interface Item 2.13 related to the required bearing capacity of foundation materials.

AP1000 COL Supplements

- COL SUP 2.5-2

The applicant provided supplemental information in COL SUP 2.5-2 regarding the backfilling of exploratory boreholes for monitoring wells. This supplement is addressed in WLS COL FSAR Section 2.5.4.2.1.2.

- COL SUP 2.5-3

The applicant provided supplemental information in WLS COL Supplement 2.5-2 regarding the grouting of polyvinyl chloride (PVC) monitoring well locations that were left open for continued monitoring. This supplement is addressed in WLS COL FSAR Section 2.5.4.2.1.2.

Interface Items

- Interface Item 2.12

The WLS COL FSAR addresses Interface Item 2.12 of the AP1000 DCD related to the seismic parameters peak ground acceleration, response spectra and shear wave velocity.

- Interface Item 2.13

The WLS COL FSAR addresses Interface Item 2.13 related to the required bearing capacity of foundation materials.

2.5.4.2.1 Geologic Features

WLS COL FSAR Section 2.5.4.1 evaluates the non-tectonic processes and features that may cause permanent ground deformation or foundation instability at the WLS site. In particular, the applicant focused on areas of actual or potential surface or subsurface subsidence, uplift or collapse and the causes of these potential conditions. The applicant also considered potential zones of alteration, irregular weathering profiles, zones of structural weakness, geologic history, unrelieved residual stresses in the bedrock at the site, and rocks or soils that may be unstable due to physical or chemical properties. The applicant concluded that that there are no geologic processes or features that would be detrimental to the stability of the proposed WLS site.

2.5.4.2.2 Properties of Subsurface Materials

WLS COL FSAR Section 2.5.4.2 details the various field and laboratory methods used to sample and test soil and rock to determine material properties needed for the site safety evaluation. The applicant stated that the field investigations for determining the engineering properties of soil and rock materials conform to RG 1.132. WLS COL FSAR Sections 2.5.4.2.1 and 2.5.4.2.2 provide more complete details of the soil and rock sampling programs, respectively. The applicant used the guidelines of RG 1.138 to plan and conduct the laboratory testing program. The applicant also summarized the static and dynamic laboratory tests performed, and reported the results from tests carried out on disturbed samples, undisturbed samples and remolded samples.

2.5.4.2.2.1 *Site Explorations*

WLS COL FSAR Section 2.5.4.2.1 describes the methods and equipment used to perform the site exploration. The applicant described soil and rock borings, groundwater monitoring wells, cone penetration test (CPT) soundings, surface geophysical surveys, geotechnical test pits, geologic trench excavations, and downhole deformation testing and seismic and non-seismic logging of the site soils and rock performed for the WLS investigations.

The site investigation plan was developed based on the evaluations of historic field explorations performed in the 1970’s for the CNS that is within the site boundary of the WLS. The site investigation for this application was first conducted between early 2006 and mid-2007. Since the relocation of the power plant footprint, additional field work was performed in 2012 to obtain additional geotechnical data at the revised nuclear island locations to confirm the applicability of the 2006-2007 data. The applicant provided CNS exploratory work as a supplement in WLS COL FSAR Appendix 2BB. Figure 2.5.4-01 of this report shows the exploration locations for the WLS power blocks and adjacent areas, as well as the soil and rock borings from the previous investigations for the CNS.

2.5.4.2.2.1.1 Soil and Rock Borings

The applicant obtained the soil and rock core borings using conventional drilling equipment and procedures. WLS COL FSAR Appendix 2AA presents the results of the WLS field exploration. For selected borings, the applicant presented summary borehole sheets. Figure 2.5.4-2 of this report shows the summary sheet for boring B-1004 as an example and includes the depths from which samples were taken for the shear wave velocity (V_s), compression wave velocity (V_p), Young's modulus and unconfined compression tests. In addition, the figure shows the percent recovery and the Rock Quality Designation (RQD), an index that reflects the soundness of the rock mass, in histogram fashion along the length of the boring length. The rock type is designated in the lithology column and Goodman Jack test results. The petrographic sample locations are also shown in the figure.

2.5.4.2.2.1.2 Groundwater Monitoring Wells

The applicant installed groundwater monitoring wells to obtain hydrologic data and performed a pump test to determine mass permeability to compliment the borehole slug tests and packer tests to obtain permeability within different material types within the boreholes. Groundwater flow at the WLS site occurs mainly in the saprolite, partially weathered rock and joints within the bedrock. The applicant left some groundwater monitoring wells open for continued monitoring. WLS COL FSAR Section 2.4.12 provides additional information regarding the groundwater monitoring wells and a detailed summary and evaluation of those wells.

2.5.4.2.2.1.3 Surface Geophysical Testing

The applicant performed surface geophysical testing at the site using the Spectral Analysis of Surface Waves (SASW) technique at 15 locations to compliment the downhole geophysical testing and to obtain geophysical data to depths that ranged from several meters (tens of feet) to approximately 45.7 m (150 ft) below the ground surface. The applicant stated the SASW surface geophysical testing compared favorably to down-hole geophysical testing methods, however, the SASW methodology was limited in its penetration depth and useful information was typically limited to several meters (tens of feet) below the ground surface.

2.5.4.2.2.1.4 In-Situ Testing

The applicant performed several different methods of in-situ testing in boreholes to estimate soil and rock properties. The applicant measured the deformation properties of the rock using the Goodman Jack and the borehole pressuremeter tests. Additional borehole geophysical tests in soil and rock using downhole geophysical techniques included optical televiewer, acoustic televiewer, P-S (compression and shear waves) suspension logging, downhole seismic velocity logging and seismic cone penetration test.

Goodman Jack Testing

The applicant performed 14 Goodman Jack tests in two boreholes at multiple depth intervals to measure the rock elastic modulus using the procedures described in American Society for Testing and Materials (ASTM) Standard D4971-02, "Standard Test Method for Determining in Situ Modules of Deformation of Rock Using Diametrically Loaded 76-mm (3 in.) Borehole Jack."

Borehole Pressuremeter Testing

The applicant performed 24 pressuremeter tests in two boreholes at multiple depth intervals following the procedures described in ASTM D4719-00, "Standard Test Methods for Prebored Pressuremeter Testing in Soils," to measure the elastic modulus of soil and rock. WLS COL FSAR Table 2.5.4-209 provides the results of the pressuremeter tests.

Borehole Geophysical Testing

The applicant performed borehole geophysical testing in 19 boreholes, including 16 P-S suspension logging tests, four downhole seismic velocity logging tests performed in P-S suspension logging borings for comparison purposes, and 17 borehole televiewer logging tests using optical and/or acoustic methods to develop the rock seismic profiles for the determination of the GMRS and FIRS in WLS COL FSAR Section 2.5.2. WLS COL FSAR Section 2.5.4.2.4 presents the results of the borehole geophysical testing in more details.

2.5.4.2.2.1.5 Petrographic Testing

The applicant collected 15 rock core samples from nine borings for petrographic testing. Using standard 27 by 46 mm (1.0 by 1.8 in.) covered thin sections, the applicant performed petrographic analyses and photomicrography at magnifications ranging from 25X to 500X and used the results to classify the bedrock.

2.5.4.2.2.2 *Soil and Rock Sampling*

The applicant performed Standard Penetration Tests (SPT) to determine the properties of soils that would be supporting non-safety-related structures. The SPT thick-walled sampler was used to collect disturbed samples for classification, and measurements of the resistance of soil to penetration, or N-value, was recorded.

The applicant performed rock coring when soil drilling methods met refusal to obtain intact rock cores for laboratory strength testing. Each rock core was photographed in the field and core logs prepared noting joints and fractures and other pertinent data. The percent recovery and RQD were recorded on the boring logs.

The applicant determined that rock having a RQD of 65 percent or greater is suitable foundation material and designated this material as "continuous rock," but noted that the continuous rock surface is uneven due to differential depths of weathering. Excavation to continuous rock is required at WLS Unit 2. The applicant's interpretation of the surface elevation of continuous rock across the site is depicted in Figure 2.5.4-3 of this report. The contour map shown in

Figure 2.5.4-3 of this report also reveals the uneven continuous rock surface across the site. The applicant stated that it will take down the high points and backfill the low areas with concrete under the foundation support zone of the nuclear island.

2.5.4.2.2.2.1 Static Properties of Geotechnical Materials

Table 2.5.4-1 of this report summarizes the static properties of the continuous rock based on laboratory testing of intact rock specimens and shows that rock core unconfined compressive strengths range from 68.9 to 193 Meganewtons/square meter (MPa) (10.0 to 28 kilopounds/square inch (ksi)) for the various metamorphic rocks identified at the site. Young's modulus determined from stress-strain measurements made during unconfined compressive strength (UCS) tests on instrumented samples ranged from 42,000 to 55,800 MPa (6,100 to 8,100 ksi). The applicant used data from the CNS Preliminary Safety Analysis Report (PSAR) to supplement the WLS geotechnical data for soils, such as the lightly weathered rock, where limited data was available.

The applicant provided the engineering properties of the pre-existing CNS concrete and underlying rock in a supplemental pre-demolition concrete basemat evaluation report. This report included a description of the field and laboratory activities, test results, and an evaluation of the concrete-rock interface. The UCS of the existing concrete generally exceeded 27.5 MPa (4,000 psi), and the V_s was typically greater than 2,286 m/s (7,500 fps). The applicant concluded that the field and laboratory V_s are consistent with the values used in the WLS Unit 1 FIRS.

Table 2.5.4-1 Summary of Laboratory Test Results for Intact Rock Cores (WLS COL FSAR Table 2.5.4-213)

	Meta Granodiroite	Meta Quartz Diorite	Meta Diorite	All Rock Types
Unit Weight, pcf	168 +/- 1 [19]	169 +/- 3 [14]	177 +/- 2 [8]	170 +/- 4 [41]
Unconfined Compressive Strength, ksi	23 +/- 5 [19]	17 +/-7 [14]	22 +/- 6 [8]	21 +/- 7 [41]
Young's Modulus, x10^6 psi	7.8 +/- 0.3 [3]	7.1 +/- 1.0 [7]	8.1 [1]	7.4 +/- 0.9 [11]
Poisson's Ratio	0.29 +/- 0.06 [3]	0.27 +/- 0.05 [7]	0.23 [1]	0.27 +/- 0.05 [11]

The number in brackets is the number of tests performed.

2.5.4.2.3 Foundation Interfaces

WLS COL FSAR Section 2.5.4.3 compiles the information gathered during the WLS site exploration into a geological representation of the foundation conditions at the site.

2.5.4.2.3.1 *Geotechnical Profiles*

The WLS COL FSAR includes geotechnical profiles along seven geologic cross-sections through the WLS nuclear islands and adjacent areas. Figure 2.5.4-1 of this report shows the locations where these cross sections were cut and from which is it possible to depict the former and existing ground surface, plant and yard grade, nuclear island and non-safety-related structures locations, geologic features, and relevant boring and geophysical test data. Figure 2.5.4-4 of this report shows the cross-section BB-BB', a West-East profile passes through the centerlines of the WLS. Figure 2.5.4-5 of this report shows the cross-section UU-UU', a West-East profile through the north end of the Units 1 and 2 nuclear island. The cross-sections show the extent of the existing concrete from the CNS construction at Unit 1 and the geologic interpretation of the foundation conditions underlying the structures.

2.5.4.2.4 Geophysical Surveys

WLS COL FSAR Section 2.5.4.4 describes the exploratory geophysical surveys that the applicant performed to determine the soil and rock low strain seismic characteristics within the power block areas and pre-existing concrete and soil fills. The applicant stated that it conducted the investigations using the methods described in RG 1.132. The applicant also used a combination of surface and borehole geophysical surveys to characterize the subsurface conditions. The geophysical tools used to characterize the existing concrete, soil and rock including SASW, seismic cone penetration tests (SCPT), P-S (compressional wave - shear wave) suspension logging, downhole seismic velocity logging and acoustic and optical televiewer logging. The applicant noted that the redundancy of state-of-the-art methods provides confidence in the results.

2.5.4.2.4.1 *Spectral Analysis of Surface Waves Surveys*

SASW surveys were conducted at 15 locations with survey depths ranging from several meters (tens of feet) to approximately 30.5 m (100 ft) below ground surface. The applicant concluded that the SASW technique yielded average V_s that compare favorably with borehole P-S suspension and downhole seismic velocity logs.

2.5.4.2.4.2 *Seismic Cone Penetration Tests*

The applicant performed SCPTs at 10 locations in the site soils and saprolite and used the results for the design of non-safety-related structures.

2.5.4.2.4.3 *Suspension and Downhole Seismic Velocity Logging*

The applicant performed suspension velocity logging in a total of 16 boreholes and companion downhole seismic velocity tests in 4 of the 16 boreholes for comparison. The applicant concluded that downhole velocity measurements compared favorably with the P-S suspension logging measurements and developed velocity profiles from the results of the seismic testing. Figures 2.5.4-06 through 2.5.4-07 of this report show a representative boring summary sheet for the WLS powerblock areas. These boring logs show consistent hard rock V_s of greater than 2,438 m/s (8,000 fps), particularly at depths below an El. of 162 m (533 ft).

2.5.4.2.4.4 *Acoustic and Optical Televiewer Logging*

The applicant conducted acoustic and optical televiewer logging in 17 boreholes to correct soil, rock and geophysical log depths to true depths, and provide acoustic images of the borehole wall to determine the dip and azimuth of fractures. The applicant used the fracture orientation data to estimate the Rock Mass Rating and evaluate discontinuity characteristics.

2.5.4.2.5 Excavation and Backfill

WLS COL FSAR Section 2.5.4.5 describes the excavation and backfilling required to reach foundation grade and plant grade for WLS. The applicant proposed using a combination of excavation slopes and temporary retaining structures to excavate to the bottom of unsuitable materials. The purpose of the retaining structures is to minimize disturbance of the slopes.

The applicant described the foundation for the nuclear islands as consisting of continuous rock at variable elevations below the basemat foundation level at an El. of 168.7 m (553.5 ft). The applicant noted that all of the WLS Unit 1 foundation support zone is underlain by pre-existing CNS concrete overlying continuous rock from the former construction activities associated with the CNS. The applicant performed pre- and post-demolition studies of the CNS structural and fill concrete to determine its static and dynamic properties. The existing rock and concrete surface at WLS Unit 1 is below the basemat founding elevation. In order to bring the site to basemat foundation grade, the applicant plans to use concrete fill. The applicant proposed to use a fill concrete mix design based on American Concrete Institute (ACI) Standard 318-02 with minimum design UCS of 17.2 MPa (2,500 psi). Figure 2.5.4-9 of this report, cut along cross-section BB-BB', shows the relationship between the excavation, concrete fill, engineered granular fill, Units1 and 2 Annex Building, Unit 1 and 2 nuclear island, existing CNS concrete fill and CNS structural fill (basemat).

2.5.4.2.5.1 *Extent of Excavation*

The applicant noted that it will enlarge the extent of the excavation an additional 3.05 m (10 ft) laterally into the fill and natural soil materials removing softened, sloughed, or other loose soil and rock materials in the slopes of the existing CNS excavation. Under the nuclear islands, the excavations will be taken down as necessary to reach continuous rock, defined as having a RQD of 65 percent. The applicant determined the limits of the excavation based on the core borings distributed across the site.

The WLS Unit 1 excavation is made to a foundation subgrade elevation of approximately 164.5 to 165.1 m (540 to 545 ft), corresponding to the elevation of the pre-existing CNS structural elements where they exist. The applicant stated that the foundation support zone for the WLS Unit 1 nuclear island is entirely underlain by the existing concrete of CNS Unit 1, which is underlain by continuous rock. Figure 2.5.4-8 of this report conceptually shows the WLS excavation depths to reach continuous rock and concrete fill to be placed underneath and surrounding the nuclear island. The applicant also stated that the existing CNS concrete foundation has several local pits with horizontal and vertical waterproofing membranes.
In those locations, excavation will be performed to remove those membranes and then be continued to reach the continuous rock, or the top of the fill concrete layer that is resting on the continuous rock. The new fill concrete will be placed underneath and surrounding planned Seismic Category I structures.

The WLS Unit 2 excavation will be made mostly in continuous rock with the exception of the east side of the Reactor Building. The applicant noted that boring B-1014 indicates that the continuous rock surface is approximately 4.8 m (16 ft) below the basemat foundation elevation. The applicant proposed the use of fill concrete in this and any other areas requiring backfill.

2.5.4.2.5.1.1 Concrete Fill

The applicant noted that variable depths of concrete fill and dental concrete are needed to construct a uniform and smooth subgrade for the placement of the basemat foundation at El. 168.7 m (553.5 ft). Several meters to 7.6 m (several feet to 25 ft) of concrete backfill are needed at WLS Unit 1 and approximately 4.8 m (16 ft) of concrete fill are needed at the east side of WLS Unit 2. The applicant stated that the concrete fill mix design will be in accordance with ACI 318-02 and will have a minimum unconfined compressive strength of 17.2 MPa (2,500 psi) and V_s of 2,286 m/s (7,500 fps). The applicant assumed V_s equal to 2,286 m/s (7,500 fps) in the dynamic analyses to correspond with the average V_s determined for the existing CNS concrete fill and structural concrete.

The applicant stated that it will prepare the existing concrete left in place from the CNS construction by cleaning and scoring the surface 0.64 cm (0.25 in.) deep to provide for adhesion between the existing concrete and new concrete fill. As described in Section 2.5.4.4.5.1.2 the applicant stated these properties will be ensured through testing of the mix design.

2.5.4.2.5.2 *Soil Backfill Sources and Quantities*

WLS COL FSAR Section 2.5.4.5.1 describes the use of both on- and off-site granular material sources for backfilling the area around the below-grade nuclear island walls out to the limits of the excavation slope. The offsite granular backfill will be used to support the non-safety-related and Seismic Category II structures surrounding the nuclear island. The applicant described steps for verification of foundation conditions and placement of proper backfill materials. The applicant proposed using South Carolina Department of Transportation (SCDOT) standard gravel or sand gradations readily available from state-approved borrow sources. The applicant did not identify the source for the granular fill but stated that the granular fill material would likely be obtained from an off-site source such as an operating rock quarry. The granular fill to be placed adjacent to Seismic Category I structures or beneath other important adjacent facilities would be verified to have the properties used in site response analyses and compatible with the material properties described in WLS COL FSAR Section 3.7.2.8.4.

2.5.4.2.5.3 *Specifications and Control*

WLS COL FSAR Section 2.5.4.5.3 describes the methods and procedures used for verification and quality control of the nuclear island foundation materials and the fill concrete beneath the nuclear island foundation limits. The applicant noted that these methods and procedures will verify that all unsuitable material has been removed down to continuous rock and will be verified by geologic mapping of the final exposed excavation surface prior to placement of concrete. The staff discusses the requirement for a geologic mapping and proposed a license condition in Section 2.5.3 of this report.

In pre- and post- demolition reports, the applicant verified that the concrete fill materials and reactor building basemat slab remaining from the CNS are suitable for support of the WLS nuclear island foundation. Fill concrete beneath the nuclear island foundation limits will be designed in accordance with ACI 318-02 to attain a V_s of 2,286 m/s (7,500 fps). The applicant stated surfaces to receive concrete fill will be intentionally roughened according to the guidance in ACI 349.

The applicant plans to prepare the foundation materials beyond the limits of the nuclear island footprint for support of engineered granular backfill. The applicant also stated that it will verify proper foundation conditions and any depressions or cavities in the surface of the foundation soil or rock will be backfilled with fill concrete or properly compacted soil fill materials. Engineered backfill to support Seismic Category II and non-safety-related buildings will be granular fill conforming to SCDOT gradation limits obtained from approved quarries. The applicant further stated that it will compact the backfill to a minimum of 96 percent of the maximum density determined by the modified Proctor compaction method as described in ASTM D1556, “Standard Test Method for Density and Unit Weight of Soil in Place by Sand-Cone Method.” Finally, the applicant stated that in areas where backfill is placed over saprolite, the saprolite will have N_{60} values of at least 15 blows per foot.

2.5.4.2.5.4 *Groundwater Control*

The applicant plans to accomplish temporary groundwater control during excavation and backfill by deep dewatering wells placed outside of the excavation limits, combined with sump pumps within low areas in the excavation. The applicant proposed a dewatering scheme similar to that used for the CNS.

2.5.4.2.6 Groundwater Conditions

The applicant noted that the groundwater conditions at the WLS nuclear island consist of a permanent ground water table elevation which ranges from 175 to 178 m (574 to 584 ft) MSL. The standard design plant maximum water table is at El. 180 m (591 ft), and the basemat foundation is at El. 168.7 m (553.5 ft) MSL. The applicant noted that, although the structure is below the long term static groundwater elevation, permanent dewatering systems are not required as the foundation basemat and below grade walls are waterproofed and designed for hydrostatic pressure as part of the AP1000 standard design.

2.5.4.2.7 Response of Soil and Rock to Dynamic Loading

WLS COL FSAR Section 2.5.4.7 presents the response of the soil and rock to dynamic loading. The applicant determined the dynamic material properties in rock and existing site soils by various means including geophysical borehole logging surveys, and by estimating the dynamic properties of borrow materials for engineered fill based on engineering experience with compacted fill and dynamic property research conducted on granular soils by Menq (2003).

2.5.4.2.7.1 *Field Dynamic Measurements*

The applicant established representative dynamic velocity profiles based on the seismic survey results obtained from downhole and surface seismic surveys performed across the site. Figure 2.5.4-9 of this report shows the locations of the dynamic profiles and the individual seismic surveys used to develop the dynamic profiles. Representative velocity profiles from the footprints of WLS were previously presented in the summary boring data sheets in Section 2.5.4.2.4 and Figures 2.5.4-6 and -7 of this report.

2.5.4.2.7.2 *Dynamic Profiles*

The applicant used the seismic velocity profiles to develop three velocity profiles representing the WLS Unit 1 centerline (smoothed dynamic Profile A), the WLS Unit 2 centerline (smoothed dynamic Profile C) and generic engineered granular fill (best estimate layer velocity Profile G). The applicant used Profile A to develop the site GMRS, and Profile C to evaluate possible differences in site response between WLS as a result of the spatial separation and possible lateral variability in the rock properties. The applicant developed Profile G to represent engineered granular fill placed over the bedrock and around the plant nuclear islands to develop the plant grade. Coinciding with the location of Profile A is the dynamic profile Base Case A1, which includes the layered CNS fill and structural concrete, and WLS concrete fill, which the applicant used for the Unit 1 FIRS analysis. The base Case A5 (CNS Unit 1 Pump Rooms) defines the GMRS and localized condition of the WLS Unit 1 nuclear island that will overlie legacy CNS pump rooms at approximately 161 m (527 ft) at the North American Vertical Datum of 1988 (NAVD). The site GMRS and WLS Unit 1 FIRS A1 dynamic analyses are summarized and evaluated in Sections 2.5.2 and 3.7.1 of this report, respectively.

2.5.4.2.7.2.1 WLS Unit 1 Nuclear Island

The applicant stated that the foundation rock consists primarily of slightly weathered to fresh rock that exhibits high average V_s, in the typical range of 2,743 to 3,048 m/s (9,000 to 10,000 fps). The applicant indicated that the Unit 1 nuclear island footprint initially proposed in its COL application was approximately 90 percent underlain by concrete, but the relocated plant footprint will be fully underlain by concrete placed on top of continuous rock during the CNS construction. The construction of WLS Unit 1 will require the placement of additional concrete fill ranging from several feet to as much as 7.6 m (25 ft) thick to bring the subgrade to basemat grade. For the development of the WLS dynamic velocity model, the applicant assumed that the Unit 1 concrete material properties were of similar composition, strength, quality and dynamic properties as the CNS concrete. Figure 2.5.4-10 of this report shows the WLS Unit 1 centerline dynamic Profile A, which the applicant used in determining the GMRS summarized and evaluated in Section 2.5.2 of this report.

2.5.4.2.7.2.2 WLS, Unit 2 Nuclear Island

The applicant stated that it will found the WLS Unit 2 nuclear island basemat on sound, massive metamorphic bedrock that fully complies with the criteria for a uniform site as described in AP1000 DCD Section 2.5.4.5. At locations where the bedrock is below the foundation grade, the applicant will use concrete fill. The maximum thickness of concrete fill is about 6.1 m (20 ft) beneath the east portion of the nuclear island. Figure 2.5.4-11 of this report shows the dynamic Profile C established under the centerline of Unit 2. The applicant compared the dynamic analyses using Profile C (Unit 2 centerline) to the dynamic response obtained using Profile A (Unit 1 centerline) to evaluate ground motion variability between Units 1 and 2. Figure 2.5.4-12 of this report shows the dynamic profile Base Case A5 and C4 that consider site variability within the foundation rock, the existing CNS concrete materials and WLS concrete fill, assuming a V_s of 2,286 m/s (7,500 fps) in the concrete materials. The WLS Unit 2 FIRS C4 analyses are described in Sections 2.5.2 and 3.7.1 of this report.

2.5.4.2.7.3 *Laboratory Dynamic Testing*

Although the final selection of the borrow area has not been made, the applicant stated that the laboratory dynamic testing program will consist of resonant column torsional shear (RCTS) testing of the selected borrow materials. The applicant estimated soil dynamic properties of the proposed granular fill types: well-graded gravel (GW), poorly-graded gravel (GP), or well-graded sand (SW) based on the relationships of Menq (2003) and developed generic fill dynamic profiles for the range of potential backfill types that may be used to support the Seismic

Category II and non-safety-related structures adjacent to the nuclear island. As an example, Figure 2.5.4-13 of this report presents the estimated V_s profile and the shear modulus degradation ratio and damping ratio as function of shear strain for the well-graded sand granular fill (SW). The applicant used the generic fill dynamic profiles and estimated dynamic soil properties in the soil structure interaction analyses and liquefaction potential evaluations.

2.5.4.2.8 Liquefaction Potential

WLS COL FSAR Section 2.5.4.8 states that all Seismic Category I safety-related plant foundations for WLS will bear on rock, or fill concrete over rock, neither of which are susceptible to liquefaction. The applicant further stated that it will place engineered fill under structures adjacent to Seismic Category I structures to 96 percent of maximum density in accordance with modified Proctor compaction specifications, and that these materials will have a very low potential for liquefaction. The applicant made this conclusion based on an evaluation of the liquefaction potential of granular soils (SW) placed in a very dense condition, where 96 percent modified Proctor density is equivalent to 80 percent relative density, according to Lee and Singh (1971). The estimated normalized SPT blow count number $(N_1)_{60}$ for densely compacted engineered fill is approximately 30 (Idriss and Boulanger, 2008), while gravel soils would have a $(N_1)_{60}$ of 45 (Rollins et al., 1998. The applicant conducted similar liquefaction analyses for the saprolite that underlies the engineered fill supporting non-safety-related structures and concluded that no liquefaction hazards exist for Seismic Category I safety-related structures, or the adjacent Seismic Category II and non-safety-related structures.

2.5.4.2.9 Earthquake Site Characteristics

WLS COL FSAR Section 2.5.4.9 summarizes the procedures used to determine the performance-based site-specific GMRS and FIRS developed for the WLS site. The applicant determined the methodology and the development of the Unit 2 location-specific GMRS and Unit 1 location-specific FIRS in accordance with RG 1.208. Site response analysis results are described in WLS COL FSAR Section 2.5.2.

2.5.4.2.10 Static Stability

WLS COL FSAR Section 2.5.4.10 evaluates the stability of the WLS for foundation bearing capacity, foundation settlement, and lateral pressures against below-grade walls. The applicant compared the bearing capacity and settlement estimates obtained for WLS with the design criteria in the AP1000 DCD. Additionally, the applicant estimated the bearing capacity and settlement of the non-safety-related structures surrounding the nuclear island to ensure that differential settlements between the nuclear island and the non-safety-related structures did not exceed the limit of 7.6 cm (3 in.) given in the AP1000 DCD. The applicant concluded that bearing capacity factors of safety and settlement of the Seismic Category I, Seismic Category II, and non-safety-related structures were within AP1000 DCD requirements.

2.5.4.2.10.1 *Bearing Capacity*

WLS COL FSAR Section 2.5.4.10.1 evaluates the bearing capacity of the WLS Units 1 and 2 nuclear island foundations separately. The applicant used the method of Peck, Hanson and Thornburn (1974) for allowable bearing pressure based on rock RQD, and determined the ultimate bearing capacity based on the Terzaghi (1943) bearing capacity equation and the strength of the rock mass from the Hoek-Brown (2002) criteria.

The applicant stated that the Peck, Hanson, and Thornburn (1974) method estimated a minimum allowable bearing pressure of 9,097 kPa (190,000 psf) at Unit 1 and 11,587 kPa (242,000 psf) at Unit 2. The Terzaghi method (1943) determined ultimate bearing capacities by utilizing Hoek-Brown parameters of the rock mass to establish the Mohr-Coulomb (1900) parameters of friction angle and cohesion for the rock, and the bearing capacity factors as developed in EM 1110-1-2908 (USACE, 1994) and by Sowers (1979), of at least 121,000 kPa (2,539,000 psf) under static conditions and 117,000 kPa (2,444,000 psf) under combined (static plus seismic) loading conditions. The applicant concluded that the ultimate static bearing capacity of the foundation materials at the WLS site exceeds the AP1000 DCD average static bearing capacity requirement of 425 kPa (8,900 psf) and the combined static and seismic bearing demand of 1,675 kPa (35,000 psf) by factors of safety of at least 3.0 for the static case and 1.5 for the dynamic case.

2.5.4.2.10.2 *Settlement*

WLS COL FSAR Section 2.5.4.10.2 estimates the post-construction settlement calculated separately for Unit 1 and Unit 2, and the surrounding Seismic Category II and non-safety-related structures. The applicant used three methods based on the theory of elasticity to calculate settlement: (1) the Steinbrenner equation (Bowles, 1988), (2) the United States Army Corps of Engineers equation (USACE, 1990), and (3) the Boussinesq equation (Li, 1995). The applicant also estimated settlement using an empirical approach described by Peck, Hanson, and Thornburn (1974), which is based on the RQD of the rock. The applicant assumed the nuclear island foundation bears directly on rock or on a depth of fill concrete resting on rock and, using the elastic methods, determined that the settlement is less than 0.25 cm (1/10 of an inch). The maximum estimated settlements beneath Units 1 and 2 are 0.119 and 0.121 cm (0.047 and 0.048 in.), respectively. Using the empirical RQD method, the applicant estimated a maximum settlement of 0.180 cm (0.071 in.) beneath Unit 1 and 0.140 cm (0.055 in.) beneath Unit 2. The applicant concluded that the calculated tilt settlements were within the limits allowed by the AP1000 DCD design parameter of 1.27 cm in 15.2 m (0.5 inches in 50 ft).

Additionally, the applicant computed the settlement of the adjacent Seismic Category II and non-safety-related structures to ensure that the differential settlement, settlement of the Seismic Category II and non-safety-related structures as compared to the settlement of the nuclear island, was within the AP1000 DCD criteria. The applicant calculated a total settlement of less than 5.1 cm (2 in.) for each of the Seismic Category II and non-safety-related structures; which, when compared to the nil settlement of the nuclear island, results in differential settlement of less than the allowable 7.6 cm (3 in.) between the adjacent Seismic Category II and non-safety-related structures and the nuclear island.

2.5.4.2.10.3 *Lateral Pressures*

WLS COL FSAR Section 2.5.4.10.3 addresses the lateral pressures that develop against the below grade nuclear island walls resulting from backfill placement. The applicant used the Rankine earth pressure theory to calculate earth pressure envelopes for the cases of active, at-rest and passive pressure conditions at both minimum and maximum groundwater elevations.

The applicant stated that because the walls are unyielding, the at-rest pressure case governs the design. The applicant also computed the additional lateral load induced by the compaction effort of the hand-operated equipment and heavy rollers operating within 1.5 m (5 ft) of the below ground walls. Finally, the applicant computed the dynamic lateral stresses on the walls using the method in American Society of Civil Engineers (ASCE) 4-98, "Seismic Analysis of Safety-Related Nuclear Structures and Commentary," assuming a peak ground acceleration of 0.3 g applied uniformly to the wall. Figure 2.5.4-14 of this report depicts the static at-rest lateral earth pressures, combined with the excess earth pressures created by compaction equipment operating within 1.5 m (5 ft) of the below ground nuclear island walls.

2.5.4.2.11 Design Criteria

WLS COL FSAR Section 2.5.4.11 summarizes the AP1000 DCD site parameter criteria and compares them to the site-specific parameters presented in the WLS COL FSAR. Table 2.5.4-2 of this report summarizes the critical parameters.

Table 2.5.4-2. Comparison of AP1000 Design Criteria to WLS Site Characteristics (Based on WLS COL FSAR Table 2.0-201)

	AP1000 DCD	WLS Site	WLS FSAR	WLS within Site Parameter
Static Bearing Capacity	425 kPa (8,900 psf)	9,097 to 11,587 kPa 190 to 242 ksf	2.5.4.10.1	Yes
Dynamic Bearing Capacity	1,675 kPa 35,000 psf	9,097 to 11,587 kPa 190 to 242 ksf	2.5.4.10.1	Yes
Shear Wave Velocity	305 m/s (1,000 fps)	2,743 to 3,048 m/s 9,000 to 10,000 fps	2.5.4.7	Yes
Lateral Variability	CASE 1, uniform layer thickness dips no more than 20 degrees, and less than 20% shear wave velocity variation in any layer.	CASE 1, non-dipping massive plutonic rock with less than 20% shear wave velocity variation.	2.5.4.7.4	Yes
Liquefaction Potential	No liquefaction considered beneath the SC I and SC II structures and immediate surrounding area.	None: nuclear Island founded on rock or concrete over rock; adjacent structures have negligible liquefaction potential.	2.5.4.8	Yes
Minimum Angle of Internal Friction of Foundation Soils	Greater than 35 degrees.	Founded on rock meeting the criteria	Not applicable	Yes

WLS COL FSAR Section 2.5.4.11 does not specify any departures from or supplements to the AP1000 DCD. However, because of the reevaluation of the site-specific seismic loading, the applicant addressed a departure of lateral earth pressure under one load combination in WLS COL FSAR Sections 2.5.4.10 and 3.8.4.4.4.

2.5.4.2.12 Techniques to Improve Subsurface Conditions

WLS COL FSAR Section 2.5.4.12 addresses the methods and means for preparing the rock surface to ensure that the WLS Units 1 and 2 nuclear island foundation mat is supported by continuous rock, or by fill concrete that is supported on continuous rock.

The applicant stated that it will remove the soil, weathered rock, and concrete within the nuclear island area and above the design foundation subgrade, down to continuous rock, described as rock having a RQD equal to or greater than 65 percent. The applicant noted that it will not remove some relatively minor zones of lower quality rock (RQD < 65 percent). Figure 2.5.4-3 of this report shows the contours of the continuous rock surface. The applicant planned to verify the foundation improvement by geologic mapping of the final excavation surface prior to foundation treatment or placement of any fill materials. The applicant also stated that the foundation preparation in advance of concrete placement involves removing loose soil and rock, or other materials from the foundation surface, removing protrusions and overhangs within the rock or concrete, washing the exposed rock or concrete surface with air and/or water, treating isolated depressions or cracks in the rock or concrete surface with fill concrete, and roughening exposed concrete surfaces. The applicant also planned to seal the CNS Unit 1 under slab drainage system with concrete to prevent the migration of fill materials.

Figure 2.5.4-15 of this report shows one example of how the applicant will backfill drainage channels in the existing CNS foundation, while Figure 2.5.4-16 of this report shows how the applicant will improve severely weathered localized zones of rock found in the continuous rock subgrade.

2.5.4.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in the FSER related to the AP1000 DCD and its supplements and in NUREG-1923.

In addition, the acceptance criteria associated with the relevant requirements of NRC regulations for stability of subsurface materials and foundations are given in NUREG-0800, Section 2.5.4.

The applicable regulatory requirements for reviewing the applicant's FSAR on the stability of subsurface materials and foundations are as follows:

- 10 CFR Part 50, Appendix A, GDC 1, “Quality Standards and Records,” as it relates to the requirement that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. It also requires that appropriate records of the design, fabrication, erection, and testing of SSCs important to safety be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

- 10 CFR Part 50, Appendix A, GDC 2, "Design Bases for Protection Against Natural Phenomena,” as it relates to 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.

- 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants," establishes quality assurance requirements for the design, construction, and operation of those SSCs of nuclear power plants that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public.

- 10 CFR Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," as it applies to the design of nuclear power plant structures, systems, and components important to safety to withstand the effects of earthquakes.

- 10 CFR 100 provides the criteria that guide the evaluation of the suitability of proposed sites for nuclear power and testing reactors.

- 10 CFR 100.23 provides the nature of the investigations required to obtain the geologic and seismic data necessary to determine site suitability and identify geologic and seismic factors required to be taken into account in the siting and design of nuclear power plants.

The related review acceptance criteria from NUREG-0800, Section 2.5.4 are as follows:

- Geologic Features: To meet the requirements of 10 CFR Part 50 and 10 CFR Part 100, the section defining geologic features is acceptable if the discussions, maps, and profiles of the site stratigraphy, lithology, structural geology, geologic history, and engineering geology are complete and are supported by site investigations sufficiently detailed to obtain an unambiguous representation of the geology.

- Properties of Subsurface Materials: To meet the requirements of 10 CFR Part 50 and 10 CFR Part 100, the description of properties of underlying materials is considered acceptable if state-of-the-art methods are used to determine the static and dynamic engineering properties of all foundation soils and rocks in the site area to sufficient depth that impact behavior during construction and over the life of the facility, including during postulated seismic events.

- Foundation Interfaces: To meet the requirements of 10 CFR Part 50 and 10 CFR Part 100, the discussion of the relationship of foundations and underlying materials is acceptable if it includes (1) a plot plan or plans showing the locations of all site explorations, such as borings, trenches, seismic lines, piezometers, geo logic profiles, and excavations with the locations of the safety-related facilities superimposed thereon; (2) profiles illustrating the detailed relationship of the foundations of all Seismic Category I and other safety-related facilities to the subsurface materials; (3) logs of core borings and test pits; and (4) logs and maps of exploratory trenches in the application for an early site permit or COL.

- Geophysical Surveys: To meet the requirements of 10 CFR 100.23, the presentation of the dynamic characteristics of soil or rock is acceptable if geophysical investigations have been performed at the site and the results obtained there from are presented in detail.

- Excavation and Backfill: To meet the requirements of 10 CFR Part 50, the presentation of the data concerning excavation, backfill, and earthwork analyses is acceptable if: (1) the sources and quantities of backfill and borrow are identified and are shown to have been adequately investigated by borings, pits, and laboratory property and strength testing (dynamic and static); long term solubility properties and dissolution behavior during the life of the facility have been determined; and these data are included, interpreted, and summarized; (2) the extent (horizontally and vertically) of all Seismic Category I excavations, fills, and slopes are clearly shown on plot plans and profiles; (3) compaction specifications and embankment and foundation designs are justified by field and laboratory tests and analyses to ensure stability and reliable performance over the life of the plant; (4) the impact of compaction methods are incorporated into the structural design of the plant facilities; (5) quality control methods are discussed and the quality assurance program described and referenced. If backfill is to be placed under safety-related structures, proper inspections, tests, analyses and acceptance criteria (ITAAC) should be specified in the applicant’s technical submittal to ensure that the static and dynamic properties of in-place backfill material will be the same as, or better than the design parameters. In case cementitious construction material is to be placed under safety -related structures, proper ITAAC should be specified in the applicant technical submittal to ensure that the cementitious backfill placed underneath any Seismic Category I structures to a thickness greater than five feet, meets the design, construction and testing of applicable ACI standards; (6) control of groundwater during excavation to preclude degradation of foundation materials and properties is described and referenced. In addition, the long-term behavior of the backfill subjected to any aggressive groundwater characteristics is evaluated; and (7) for sites where deeply embedded structures are involved, deep excavation techniques will likely utilize wall retaining systems rather than a sloped excavation of the soil. A description of the planned excavation technique(s) and design of the wall retention system with sufficient details is provided and it should be able to demonstrate that the excavation technique used will not significantly affect the surrounding soil properties that are relied upon in the analysis and design of the foundation and plant structures.

- Ground Water Conditions: To meet the requirements of 10 CFR Part 50 and 10 CFR Part 100, the analysis of groundwater conditions is acceptable if the following are included in this subsection or cross-referenced to the appropriate subsections in Section 2.4 of the applicant’s technical submittal: (1) discussion of critical cases of groundwater conditions relative to the foundation settlement and stability of the safety-related facilities of the nuclear power plant; (2) plans for dewatering during construction and the impact of the dewatering on temporary and permanent structures. This includes consideration of the potential for substantial head and volume of water due to the deep excavation for the plant structures; (3) analysis and interpretation of seepage and potential piping conditions during construction; (4) records of field and laboratory permeability tests as well as dewatering induced settlements; (5) history of groundwater fluctuations as determined by periodic monitoring of an adequate number of local wells and piezometers. Flood conditions should also be considered; and (6) evaluation of chemical properties of the groundwater that may impact long-term behavior of the rock/soil/fill materials as well as structural elements (concrete and steel materials).

- Response of Soil and Rock to Dynamic Loading: To meet the requirements of 10 CFR Part 50 and 10 CFR Part 100, descriptions of the response of soil and rock to dynamic loading are acceptable if: (1) an investigation has been conducted and discussed to determine the effects of prior earthquakes on the soils and rocks at the site. Evidence of liquefaction and sand cone formation should be include; (2) field seismic surveys (surface refraction and reflection and in-hole and cross-hole seismic explorations) have been accomplished and the data presented and interpreted to develop bounding P and S wave velocity profiles; and (3) dynamic tests have been performed in the laboratory on undisturbed samples of the foundation soil and rock sufficient to develop strain-dependent modulus-reduction and hysteretic damping properties of the soils and the results included. If generic soil degradation properties are used in the related preliminary analyses (e.g., site seismic response and soil structure interaction analyses), then reconciliation of the generic properties and laboratory testing results should be performed. The section should be cross-referenced with WLS COL FSAR Section 2.5.2.5.

- Liquefaction Potential: To meet the requirements of 10 CFR Part 50 and 10 CFR Part 100, if the foundation materials at the site adjacent to and under Seismic Category I structures and facilities are saturated soils and the water table is above bedrock, then an analysis of the liquefaction potential at the site is required.

- Static and Dynamic Stability: To meet the requirements of 10 CFR Part 50 and 10 CFR Part 100, the discussions of static and dynamic analyses are acceptable if the stability of all safety-related facilities has been analyzed from a static and dynamic stability standpoint including bearing capacity, rebound, settlement, and differential settlements under dead loads of fills and plant facilities, dynamic loads including "live" and seismic loads with consideration of loading sequences and combinations.

- Design Criteria: To meet the requirements of 10 CFR Part 50, the discussion of criteria and design methods is acceptable if the criteria used for the design, the design methods employed, and the factors of safety obtained in the design analyses are described and a list of references presented.

- Techniques to Improve Subsurface Conditions: To meet the requirements of 10 CFR Part 50, the discussion of techniques to improve subsurface conditions is acceptable if plans, summaries of specifications, and methods of quality control are described for all techniques to be used to improve foundation conditions (such as grouting, vibroflotation, bridging mats, dental work, rock bolting, or anchors).

In addition, the geotechnical engineering characteristics should be consistent with appropriate sections from: RG 1.27, "Ultimate Heat Sink for Nuclear Power Plants"; RG 1.28, "Quality Assurance Program Requirements (Design and Construction)"; RG1.132, "Site Investigations for Foundations of Nuclear Power Plants"; RG 1.138, "Laboratory Investigations of Soils And Rocks for Engineering Analysis and Design of Nuclear Power Plants"; RG 1.198, "Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites"; and RG 1.206, "Combined License Applications for Nuclear Power Plants."

2.5.4.4 *Technical Evaluation*

2.5.4.4.1 Description of Site Geologic Features

WLS COL FSAR Section 2.5.4.1 presents an overview of the site geologic features and refers to WLS COL FSAR Sections 2.5.1.1 and 2.5.1.2 for a more detailed description of the regional and site geology, respectively. The technical evaluation of site geologic features is presented in Section 2.5.1.4 of this report. The technical evaluation for vibratory ground motion is presented in Section 2.5.2.4 of this report, and the technical evaluation of the potential for surface faulting at the WLS site is located in Section 2.5.3.4 of this report. These sections of this report, respectively, present the WLS COL FSAR summaries and conclusions regarding the site geologic features. The applicant provided geotechnical and geological data in the form of boring logs from the WLS site investigation and the CNS site investigation in Appendices 2AA and 2BB to resolve WLS COL 2.5-1.

2.5.4.4.2 Properties of Subsurface Materials

WLS COL FSAR Section 2.5.4.2 describes the field investigation and subsurface material properties at the WLS site. Section 2.5.4.2.2 of this report provides a summary of the information presented in the WLS COL FSAR. The staff reviewed the geotechnical data presented and evaluated the WLS exploration and laboratory testing programs in accordance with the guidelines presented in RG 1.132 and RG 1.138. The main focus of the review was to ensure that the soil and rock profile was adequately characterized in order to evaluate the ability of the subsurface to support the nuclear island statically and dynamically.

2.5.4.4.2.1 *Site Explorations*

The applicant relied on 189 borings completed in support of the CNS exploration program to locate and drill a total of 131 geotechnical borings for the WLS site exploration. Figure 2.5.4-1 of this report shows the locations of the borings in the power block and adjacent areas. Additionally, the applicant made test pits and seismic CPT soundings in the soil profile at many locations, cored the existing CNS concrete and supporting rock, and provided the results of in-situ and laboratory engineering tests in pre- and post- demolition reports. In several boreholes and at various depths, the applicant performed downhole deformation testing with two testing devices in order to evaluate the engineering properties of the rock mass. The applicant continuously sampled additional holes for dynamic rock mass properties using P-S suspension logging and downhole seismic velocity logging methods. The applicant obtained acoustic and televiewer data to study the lithology and stratification of the rock. Samples collected for strength testing and petrographic analysis were handled in accordance with ASTM D4220-95, "Standard Practices for Preserving and Transporting Soil Samples."

In general and prior to the construction activities associated with the CNS, the site consists of residual soils overlying saprolite, overlying partially weathered rock, overlying bedrock. The staff noted that the depth of the overburden is variable due to differential weathering across the site and, particularly where the WLS will be located, the overburden has been partially or completely removed to expose the bedrock. The staff notes that the applicant estimated soil properties for the support of the non-safety-related structures. Since the estimated soil properties are not used in the design of safety-related structures, the staff concludes that the estimated soil data is adequate for the design of non-safety-related structure foundations and lateral stresses against the nuclear island below ground walls.

The staff noted that the WLS will be founded on continuous rock having a RQD of 65 percent or greater. The applicant cored the rock continuously to depths of up to approximately 77.7 m (255 ft) within the nuclear island footprints and between the WLS nuclear island locations. Selected core samples were tested in the laboratory for strength and deformation characteristics in unconfined compression and unconfined compression with stress-strain measurements. The applicant concluded that the materials were consistently high strength with high elastic modulus. The staff also observed that the applicant performed in-situ deformation tests in selected boreholes using the Goodman Jack test and pressuremeter. The results of those tests were lower than the results of tests on intact samples due to the discontinuities in the rock mass, but still indicated high strength and low deformation. The staff notes that with the exception of some local pockets of rock with lower strength and stiffness, this site is characterized by a rock mass having a high UCS and high elastic modulus, and consequently high bearing capacity and negligible deformation. Since the number of tests indicating lower strength was small and from few locations in a limited number of borings, the staff concludes that those areas with lower strength are not representative of the rock mass on the whole, and therefore it should not affect the static stability of the nuclear units. The staff further concludes that the exploratory program is sufficient to adequately characterize the subsurface profile and assign material properties to the geotechnical model.

2.5.4.4.2.2 *Existing Concrete at WLS Unit 1*

The applicant performed a pre-demolition exploration of the existing concrete to investigate the quality of the concrete and concrete/rock interface for concrete strength, V_s and adherence between the concrete and underlying bedrock. Documentation for the exploration was submitted as a supplement to the WLS COL FSAR.

The staff compared the results of the pre-demolition report to the concrete material property assumptions made in the WLS COL FSAR and concludes that the measured values of the pre-demolition exploration envelop the assumed UCS and V_s values of the concrete used in the design.

2.5.4.4.2.3 *Staff Conclusions Regarding the Properties of Subsurface Materials*

Based on the types of tests performed, the numbers and locations of tests, the redundancy of data collected by various methods, the consistency in strength and stiffness data, and the replication of data from previous exploration programs, the staff concludes that the exploratory program for the WLS site followed the guidance of RG 1.132 and RG 1.138, and collected sufficient information to provide the key geotechnical parameters required to perform the necessary engineering analyses. The staff also concludes that the applicant provided adequate geological, geotechnical and geophysical field and laboratory test results and engineering analyses to close COL Information Item 2.5-6. Furthermore, the staff concludes that the data from the geotechnical site investigation addressed the provision that the properties of the foundation materials are within the range of the properties considered for the certified design of the structures and foundations. The staff further concludes that the material properties are reasonable representatives of in-situ conditions at the WLS site. Accordingly, the staff concludes that the exploration program meets the requirements of 10 CFR Part 50 and 10 CFR Part 100.

2.5.4.4.3 Foundation Interfaces

WLS COL FSAR Section 2.5.4.3 describes the foundation conditions existing under the WLS nuclear islands and non-safety-related structures surrounding the nuclear islands.

The staff focused its review on the geologic interpretations made by the applicant by reviewing the individual boring logs, boring summary sheets, and geologic profiles relative to the foundation embedment depths of the nuclear island and the uniformity requirements of the AP1000 DCD.

The applicant stated that it will found the safety-related structures on continuous rock having an RQD of at least 65 percent representing a fair rock mass quality index. The staff reviewed the RQD values recorded on the CNS and WLS site borings, and concludes that the RQD values are consistently greater that 65 percent over most of the site area.

All or portions of the annex building, turbine building and radwaste building will be founded on engineered backfill resting on competent, non-liquefiable residual soils, saprolite, partially weathered rock, concrete or continuous rock depending on the category of the structure and its location. The staff reviewed the SPT borings penetrating the residual soil, saprolite and into the partially weathered rock and concludes the borings are adequate to determine the limits to which the excavation must be carried to ensure removal of all unstable soil and soil-like material. Figures 2.5.4-04 and 2.5.4-09 of this report illustrate the rock and soil boring information used to evaluate foundation conditions and excavation requirements under the footprints of the nuclear island and non-safety-related structures and also present SPT data for the residual soils and saprolite used in assessing the stability of the natural soils and the required depth of excavation.

Based on the review of WLS COL FSAR Section 2.5.4.3, the staff concludes that the applicant provided adequate information to illustrate the interface of foundation and subsurface materials. The cross-sections of the site in the power block and adjacent areas show the geological condition and excavation and backfill plan, and the soil profiles and boring summary sheets present geotechnical and site uniformity data. As the applicant also presented the detailed results of geotechnical, geological and geophysical field testing obtained during the WLS site investigation in Appendix 2AA, the staff further concludes that this information closes COL Information Items 2.5-2 and COL Information Item 2.5-3. The applicant provided cross-sections showing geologic profiles of WLS borings with structures superimposed at their locations and embedment depths. The staff concludes that the applicant provided sufficient information on the underlying site conditions, topographic features, and locations of Seismic Category I structures in WLS COL FSAR Section 2.5.4.3 to close COL Information item 2.5-5. Finally, the staff concludes that the applicant provided sufficient foundation interface data to satisfy the requirements of 10 CFR Part 50 and 10 CFR Part 100.

2.5.4.4.4 Geophysical Surveys

The applicant described surface and borehole geophysical surveys conducted on the WLS site in WLS COL FSAR Section 2.5.4.4. The staff focused its review on the methods used to determine the V_s, the level of agreement between the various surface and downhole testing methods used, and the site conformity with respect to the minimum acceptable V_s and uniformity requirements described in the AP1000 DCD.

The AP1000 DCD requires a minimum small strain V_s of 305 m/s (1,000 fps) within the footprint of the nuclear island at its excavation depth, and the V_s measured in any given layer must be within 20 percent of the average V_s of that layer, or a site-specific soil-structure interaction (SSI) analysis must be performed. The staff noted that of the three geophysical survey methods performed to measure the V_s, P-S suspension logging is the most accurate due to its near continuous record of discreet measurements made throughout the depth of the borehole.

The staff reviewed the V_s profiles and observed that the basemat bottom elevation of the nuclear island at both WLS will be at an elevation (El.) of 168.7 m (553.5 ft). The staff also observed that the borings typically originated below the basemat bottom elevation in the WLS Unit 1 powerblock area, and above the basemat bottom elevation at WLS Unit 2. Since the existing WLS Unit 1 subgrade is already below an El. of 168.7 m (553.5 ft), the applicant planned to add concrete fill with an assumed V_s of 2,286 m/s (7,500 fps) at WLS Unit 1 to raise the existing grade to the bottom of the basemat.

2.5.4.4.4.1 *WLS Unit 1 V_s Profiles*

At WLS Unit 1, V_s profiles taken at B-1002 and B-1004 indicate that the average V_s typically exceed 2,286 m/s (7,500 fps) below the existing ground surface, with the exception of the originally planned location for the northwest corner. At the original northwest corner location, V_s profiles shown in B-1074A and 1075A had an average V_s in the range of 1,219 to 1,828 m/s (4,000 to 6,000 fps) down to an El. of 155 m (509 ft), below which the V_s increased to 2,286 to 2,743 m/s (7,500 to 9,000 fps). However, the staff noted that the original northwest corner location for WLS Unit 1 represents a softer rock condition than the remainder of the site. Therefore, in RAI 44, Question 02.05.04-9, the staff requested that the applicant justify site uniformity given the much lower V_s in this area.

However, in Revision 7 of the application, dated May 9, 2013, as reflected in Revision 9, the applicant moved the footprint 15 m (50 ft) to the east for WLS Unit 1 and 20 m (66 ft) to the south for both units. Due to the relocation of the footprints, the planned location of WLS Unit 1 avoids the originally planned “northwest corner” where the uniformity of the subsurface material is less desirable than other areas within the footprint and, therefore, the concern of the subsurface material uniformity underneath the Seismic Category I structure no longer exists. Accordingly, the staff considers RAI 44, Question 02.05.04-9 resolved.

2.5.4.4.4.2 *Existing CNS Concrete*

The staff reviewed the supplemental pre-demolition CNS concrete report prepared and submitted by the applicant. The applicant investigated the existing CNS concrete that overlies the WLS Unit 1 footprint using SASW surveys and cross-hole seismic surveys performed in boreholes drilled in the existing concrete into the underlying bedrock. From the pre-demolition report results, the staff notes that V_s values from cross hole surveys and SASW testing indicated good agreement in the range of 2,286 m/s (7,500 fps) or greater. The staff concludes that the V_s of the existing CNS concrete are consistent with values assumed in the geotechnical model for the Unit 1 FIRS.

2.5.4.4.4.3 *WLS Unit 2 V_s Profiles*

At WLS Unit 2, the staff noted that the V_s profiles typically started above an El. of 168.7 m (553.5 ft), except for boring B-1014, where more extensive weathering created a depression under the East side of the Reactor building and no V_s profile was available. At this location, the continuous rock surface begins below about El. 165.6 m (543.5 ft). The staff noted that the V_s results in borings B-1012, B-1015 and B-1017 show V_s on the order of 2,438 to 3,048 m/s (8,000 to 10,000 fps). The depression underlying the Unit 2 Reactor, location of B-1014, will require placement of concrete fill to bring the subgrade to an El. of 168.7 m (553.5 ft). The staff concludes that the concrete fill will require at least V_s of 2,286 m/s (7,500 fps) to match the properties assumed in the applicant's dynamic analysis. In WLS COL FSAR Table 2.5.4-220, "Quality Control Recommendations for Nuclear Island Fill Concrete," the applicant stated that the compressive strength as determined from the preconstruction mix design and testing program will ensure that the fill concrete will exhibit an average shear wave velocity greater than or equal to 2,286 m/s (7,500 fps), which will meet design requirements.

2.5.4.4.4.4 *Conclusions Regarding Geophysical Surveys*

The staff concludes that WLS COL FSAR Section 2.5.4.4 was thorough and that the results obtained by various state-of-the-art methods were complementary and consistent. The staff further concludes that the geophysical surveys used to reliably determine dynamic properties of soil and rock were appropriate and the resulting data is sufficient for use in the determination of the GMRS and FIRS for structures, performance of SSI analyses, and determination of the liquefaction potential in the soil profiles at the site. Accordingly, the staff considers the information provided in WLS COL FSAR Section 2.5.4.4 acceptable to satisfy the criteria of 10 CFR Part 100.

2.5.4.4.5 Excavation and Backfill

The applicant presented information regarding excavations and backfill for the WLS site. Section 2.5.4.2.5 of this report summarizes this information. The staff focused its review in the applicant's description of excavations for safety-related structures, sources for backfill materials, properties of materials, temporary slopes, and quality control for backfill and excavations.

2.5.4.4.5.1 *Sources and Quantities*

Although the applicant has yet to select the offsite borrow source it will use to backfill the excavation, in WLS COL FSAR Section 2.5.4.5.3.5, committed to using a qualified quarry to obtain granular fill that meet the SCDOT gradation limits to support Seismic Category II and non-safety-related structures. Support of the nuclear island foundations will be on rock or concrete fill where rock elevations are below the basemat founding level.

2.5.4.4.5.1.1 Granular Backfill

Initially, the applicant planned to use on-site materials to backfill against the below ground nuclear island walls and as support for surrounding non-safety-related structures. However, the staff reviewed the engineering data presented for the on-site backfill and, in RAI 60, Question 02.05.04-10, requested that the applicant clarify how to ensure the static and dynamic properties of backfill materials are equal or exceed that used in design and analyses. In May 15, 2009, and October 30, 2009, responses to RAI 60, Question 02.05.04-10, the applicant stated that the original data from construction of the CNS plant backfill was no longer available, but that the backfill was not required for sliding stability of the nuclear island, and therefore is non-safety-related. However, to add defense in depth, the applicant decided to use an offsite backfill source for the support of the non-safety-related structures. The applicant also planned to use the on-site backfill, but only in areas where structural support is not required.

The gradations of the off-site borrow materials are approved by SCDOT and produced at South Carolina-approved borrow pits. The potential gradations of granular backfill proposed by the applicant consist of a well-graded gravel, poorly-graded gravel, or well-graded sand. The off-site granular backfill is planned to be placed and compacted to reach 96 percent modified Proctor density. The applicant estimated the static and dynamic properties of the three gradations from correlations, but planned to perform laboratory tests to confirm the estimated properties. The applicant also determined a “best estimate” seismic velocity profile for each of the proposed gradations for use in dynamic analyses. Shear modulus and damping ratio versus strain relationships estimated based on research by Menq on granular materials will be verified by the applicant using RCTS tests on the selected borrow materials.

The staff reviewed the proposed offsite backfill materials, the testing plans, placement requirements and quality control plans and concludes that the use of the uniform materials available from SCDOT-approved borrow sources placed to 96 percent modified Proctor density meets the stability requirements for support of the non-safety-related structures surrounding the nuclear island. The staff further concludes that the use of on-site borrow materials at non-critical locations placed to 95 percent standard Proctor density is also acceptable because there is no backfill requirement for sliding stability or overturning for nuclear islands founded on hard rock. Accordingly, the staff considers RAI 60, Question 02.05.04-10 resolved.

On April 9-11, 2012, the staff audited the site response and SSI calculations performed by Westinghouse Electric Corporation in support of the design of WLS. During the audit, the applicant indicated that the GW, GP, and SW type of soils, as defined by Unified Soil Classification system, were considered as backfill in its site response and SSI analyses. The applicant also planned to utilize a gabion/Mechanically Stabilized Earth (MSE) wall to support fill adjacent to the nuclear island. However, the staff noted that the applicant did not specify the soil classification of the backfill soil to be placed around Seismic Category I and adjacent Seismic Category II structures, and there is no mention of the MSE wall in WLS COL FSAR Section 2.5.4. Therefore, in RAI 106, Question 02.05.04-17, the staff requested that the applicant specify the type of material to be used as backfill with its classification and static and dynamic properties; discuss how to ensure that the as-placed backfill properties are comparable to those assumed in all related site-specific analyses; and provide details of MSE wall design and discuss its impact on side fill compaction and structural stability analyses.

In a June 18, 2012, response to RAI 106, Question 02.05.04-17, the applicant stated that two SCDOT granular products, namely Macadam Base Course (MBC) and Washed Screenings are candidates for backfill material. Two specific gradations within the range for MBC and a third specific gradation within the range for Washed Screenings are to be used to predict static and dynamic properties that are considered to be typical for these materials. Both of these materials will be considered for construction use pending selection of the quarry source and performance of a test program to confirm compatibility with the estimated properties. The selection criterion is that a particular granular quarry aggregate must have static and dynamic properties comparable to those used in design and analyses. The applicant also states that the classification symbol for a given specimen of Washed Screenings could be SW or GP.

The applicant revised WLS COL FSAR Table 2.5.4-222 to present a list of laboratory and field tests that will be conducted to confirm the properties of the chosen backfill material. The applicant also planned to construct a "test fill" pad on-site using the equipment and granular fill materials to be used in construction, and then perform the listed tests on the test fill. The applicant then specified that before the production backfill commences, an engineering report will be provided to the NRC to demonstrate that the equipment and methods used to construct the test fill are capable of producing acceptable and consistent results. A program of in-place measurements of V_s in the granular backfill will be performed to confirm that shear wave velocities of the as-placed granular backfill are comparable to those used in the site-specific analyses.

The applicant stated that for construction convenience, a system of MSE walls will be constructed to surround the nuclear islands. The applicant then described the MSE wall design, including materials to be used, the facing of the wall and construction procedure. This description emphasize that the presence of a MSE wall (with its facing of wire forms and geogrids with geotextile fabric) will have no adverse effect on the structural stability, foundation bearing capacity or settlement analyses of these structures. The applicant then proposed a revision to WLS COL FSAR Section 2.5.4.5.3.5 and WLS COL FSAR Table 2.5.4-222 with a statement of that the moisture content of the fill at the time of compaction should be within approximately 3 percent of the optimum moisture content determined in accordance with ASTM D1557, “Standard Test Methods for Laboratory Compaction Characteristics of Soil Using Modified Effort (56,000 ft-lbf/ft3 (2,700 kN-m/m3)).”

Based on the review of responses to RAI 106, Question 02.05.04-17, the staff concludes that the applicant provided sufficient information on the type of backfill materials to be used, the sources where the backfill soil will be obtained, and a suite of tests that will be performed during backfill construction, and details of the MSE wall design and construction. The staff further concludes that the backfill materials identified by the applicant have properties that can meet the design parameter requirements, and the planned field and laboratory tests are adequate to ensure that the as-placed backfill will have properties as designed. Since the designed MSE wall will have similar properties as the soil surrounding the wall, the staff concludes that the existence of the MSE wall will have no adverse effect on the structural stability, foundation bearing capacity or settlement analysis results of these structures. The staff concludes that the applicant properly addressed the staff’s concerns regarding the compaction of the fills, and confirmed that WLS COL FSAR Revision 11, incorporated proposed changes committed to in the response. Accordingly, the staff considers RAI 106, Question 02.05.04-17 resolved.

2.5.4.4.5.1.2 Concrete Fill

Concrete fill is required at both WLS. The applicant assumed for its dynamic analyses that the WLS concrete fill will have a V_s of 2,286 m/s (7,500 fps) with a design compressive strength of 17.2 MPa (2,500 psi). In RAI 95, Question 02.05.04-16, the staff requested that the applicant provide details on how it would ensure that the in-place concrete fill would have V_s of 2,286 m/s (7,500 fps), while theoretically concrete with compressive strength of 17.2 MPa (2,500 psi) is not normally capable of possessing such a high shear wave velocity.

In a March 17, 2011, response to RAI 95, Question 02.05.04-16, the applicant stated that the concrete mix design would be in accordance with ACI 318-02 and testing of the mix design would be performed to ensure that a V_s of 2,286 m/s (7,500 fps) is reached. The applicant acknowledged that this may be at a higher compressive strength than 17.2 MPa (2,500 psi) and proposed a revision to WLS COL FSAR Table 2.5.4-220 to indicate that an increase in unconfined compressive strength may be required to meet the V_s requirement of 2,286 m/s (7,500 fps).

The staff concludes that the required shear wave velocity for concrete backfill can be obtained by increasing the UCS using the mix design in accordance with ACI 318-02. Therefore, the staff finds the applicant’s response to RAI 95, Question 02.05.04-16 acceptable. The staff further confirms that the proposed changes were incorporated in Revision 9 of the WLS COL FSAR. Accordingly, the staff considers RAI 95, Question 02.05.04-16 resolved.

2.5.4.4.5.2 *Extent of Excavations*

WLS COL FSAR Section 2.5.4.5.2 states that the excavation for the CNS removed much of the surface material above the bedrock within the footprints of the proposed WLS.

At WLS Unit 1, previous excavation for the CNS was completed down to continuous rock under the WLS Unit 1 footprint. The applicant planned to conduct necessary excavation down to continuous rock, and prepare the existing rock and concrete surfaces to receive new concrete fill. Proposed temporary excavation slopes will vary from 1.5 horizontal to 1 vertical in the residual soil to as steep as 0.25 horizontal to 1 vertical in the partially weathered rock. Figure 2.5.4-8 of this report shows a typical excavation cross-section of WLS. The staff noted that the excavation at WLS Unit 2 requires the removal of hard rock down to the basemat grade at El. 168.7 m (553.5 ft) over most of the nuclear island footprint. However, an area underlying the east side of the WLS Unit 2 nuclear reactor is already below grade and contains rock with RQD values less than 65 percent as observed in boring B-1018. This will require removal of unsatisfactory material to reach continuous rock, and then subgrade preparation, prior to placing approximately 4.8 m (16 ft) of concrete fill to bring this area to grade. The concrete fill will have properties that meet the strength and V_s requirements as previously discussed. Prior to the nuclear island footprints being relocated, the staff issued RAI 44, Question 02.05.04-3, requesting that the applicant clarify how it would determine the limits of excavation in the northwest corner given the differences in RQD data between Borings B-1074 and B-1074A.

In a December 23, 2008, response to RAI 44, Question 02.05.04-3, the applicant stated that boring B-1074 encountered difficult drilling conditions and the borehole was abandoned at El. 152.8 m (501.5 ft) and replaced by boring B-1074A, drilled 1.8 m (6 ft) away. The applicant ascribed the differences in RQDs to poorer quality rock and/or mechanical breakage in boring B-1074 due to drilling. The applicant indicated that continuous rock had not been reached until El. 152.8 m (501.5 ft) in boring B-1074. The applicant concluded that this boring represents a local zone of lower quality rock that can be treated with dental concrete as shown in Figure 2.5.4-16 of this report and, therefore, can remain in place without affecting the overall static or dynamic bearing capacity of the northwest corner. In addition, because of the relocation of the WLS Unit 1 footprint, the whole foundation of the nuclear island will be founded on continuous rock, or concrete fill supported by continuous rock.

Figure 2.5.4-17 of this report shows the area in the vicinity of boring B-1018 that is already below foundation grade at Unit 2, and contains rock that does not meet the definition of continuous rock, RQD less than 65 percent. The staff determined that geologic mapping will ensure that rock not meeting the definition of continuous rock is removed down to continuous rock for the nuclear islands of WLS. As a result of the applicant's plan to found the nuclear islands on continuous rock, or concrete supported by continuous rock, the staff considers RAI 44, Question 02.05.04-3 resolved.

2.5.4.4.5.3 *Conclusions Regarding Excavations and Fills*

The applicant provided an excavation plan relative to geologic conditions and features showing the horizontal and vertical extent of excavations, fills, and slopes for Seismic Category I structures. The applicant identified areas underlying Seismic Category I structures that required fill to bring the site to basemat foundation grade and addressed fill quantities, backfill material types, estimated backfill engineering properties, and backfill compaction specifications. Based on the information provided in FSAR Section 2.5.4.5.2 and RAI responses, the staff concludes that the applicant adequately addressed and closed COL Information Item 2.5-7. Further, the dewatering plan the applicant provided for the planned excavation at WLS closes COL Information Item 2.5-8.

The staff concludes that the excavation plan considered the stability of temporary slopes and the details of subgrade preparation and placement of dental concrete for localized soft zones. The staff further concludes that engineered backfill supporting Seismic Category II and non-safety-related structures placed in a very dense condition of 96 percent of the modified Proctor density, and conforming to SCDOT specifications, and the estimated engineering properties verified by future static and dynamic laboratory and field testing, is sufficient to satisfy the requirements of 10 CFR Part 100. Finally, a license condition is required for geologic mapping of the WLS Unit 2 excavation as identified in Section 2.5.3 of this report, and additional information gained during the excavation and mapping operations will allow for modifications to the conceptual excavation plans at WLS Unit 2 to ensure that continuous rock is reached prior to the placement of concrete fill.

2.5.4.4.6 Groundwater Conditions

Details about historic and current groundwater conditions for the WLS site, as well as seepage and groundwater movement information, were provided in WLS COL FSAR Section 2.4.12. WLS COL FSAR Section 2.5.4.6 provides a summary description of the groundwater conditions at the site. The staff focused its review on the proposed plan for dewatering considering groundwater levels, groundwater flow within the saprolite and fractured rock, and proposed dewatering systems to lower the groundwater during excavation and backfill placement. The staff's review and evaluation of permanent groundwater levels, permeability and groundwater flow are provided in Section 2.4.12 of this report. The review in this section is limited to the dewatering of the excavation during construction.

The applicant justified using a range of groundwater elevations between 178 and 175 m (584 and 574 ft) based on the high water marks found on the CNS superstructure after 30 years of exposure. Since the ground water level is below the AP1000 design elevation of 180 m (591 ft) and will not require a permanent dewatering system, the applicant is only concerned with temporarily lowering the groundwater level during excavation and backfill placement.

Through field pumping tests, borehole slug tests and borehole hydraulic packer tests the applicant determined that the seepage into the excavation will occur mostly in the saprolite, partially weathered rock, and open fractures and joints in the bedrock. Based on staff experience in the region, the staff concludes that this is typical of the hydrologic characteristics of the Piedmont soil/rock profile.

The staff reviewed the packer tests and slug tests and found that similar permeability parameters were obtained by different methods. With the inflow occurring mainly in the saprolite and partially weathered rock, the staff concludes that the use of deep wells around the perimeter of the excavation and sumps at selected locations in the excavation will intercept and control seepage and uplift pressures. This plan capitalizes on the construction dewatering experience gained during the excavation for the CNS and takes into consideration the current topography and groundwater conditions. The staff concludes that this dewatering plan will be effective in minimizing lateral flow into the excavation and removing water from within the excavation.

The staff further concludes that because this is a hard rock site, there is no negative groundwater effect on the rock foundation. Erosion, settlement, bearing capacity, and bottom heave of the foundation cannot occur as a result of seepage or groundwater elevation. Although seepage into the excavation from the excavation side walls may occur, this should have minimal effect on the stability of temporary slopes due to the nature of the materials at the site. The staff notes that most of the flow should be cutoff by the exterior deep wells, and surface runoff and seepage entering the excavation can be gathered at the designated sumps and be pumped out. The staff concludes that the experience gained during the construction of the CNS eliminates many of the unknowns regarding potential excavation and dewatering problems.

Therefore, the staff concludes that the groundwater data collected, together with past experience on the site, is sufficient for the design of the temporary dewatering system to allow safe excavation and backfilling operations during the construction of WLS. Based on the review of WLS COL FSAR Section 2.5.4.6, the staff concludes that the applicant adequately described the groundwater conditions to design a temporary dewatering system to meet the construction dewatering demands, and this section meets the requirements of 10 CFR Part 50 and 10 CFR Part 100 and is acceptable.

2.5.4.4.7 Response of Soil, Granular Fill and Rock to Dynamic Loading

WLS COL FSAR Section 2.5.4.7 presents the dynamic properties for hard rock at the WLS site and refers to WLS COL FSAR Section 2.5.2 for the presentation of the development of the GMRS and the FIRS for WLS Unit 1. In this section, the staff reviewed the methods and results used to obtain dynamic profiles of the rock supporting the nuclear island, and the dynamic properties of engineered fill supporting the surrounding structures. The staff focused its review on the field dynamic measurements for the development of the rock dynamic profile underlying the Seismic Category I structures and the existing soil overburden, as well as the estimated soil dynamic material properties for the proposed engineered fill that will support the non-safety-related structures surrounding the WLS nuclear islands.

2.5.4.4.7.1 *Field Dynamic Measurements*

The staff reviewed the field seismic test program and results and the use of the state-of-the-art P-S suspension logging which provides continuous downhole measurements on 0.5 m (1.6 ft) intervals, and the confirmation of the P-S suspension logging results obtained in four of the same boreholes using downhole seismic logging equipment that averages V_s over 3.04 m (10 ft) intervals. The staff concludes that agreement between the two methods was satisfactory, and therefore reliable. Accordingly, the staff concludes that the number and distribution of velocity profiles performed for the nuclear island foundations are adequate to characterize the dynamic properties of the rock and existing concrete across the site.

The staff also considered the selection of borings B-1004, B-2000 and B-2002 for WLS Unit 1, and B-1074A and B-1075A for the previous location of the northwest corner of WLS Unit 1; and B-1015, B-1017 and B-2005 for WLS Unit 2, and concluded that these locations provided consistent results and were deemed representative of those areas for developing dynamic Profiles A and C, respectively. Based on the fact that the densely compacted fills would be supporting Seismic Category II and non-safety-related structures, the staff concludes that the use of the Menq’s relationships is an acceptable method to estimate the soil profiles and dynamic properties of the potential borrow materials until the dynamic properties can be determined by RCTS testing.

2.5.4.4.7.1.1 Foundation Conditions and Uniform

The staff noted that the velocity profiles for the previous location of the northwest corner have lower velocities than the remainder of the site down to approximately El. 155 m (509 ft). In RAI 44, Question 02.05.04-9, the staff requested that the applicant clarify how the WLS site meets the uniformity criteria, given rock not meeting the AP1000 DCD criteria would constitute a soft spot in the subgrade. However, as a result of the revision of application, the applicant moved the Unit 1 footprint about 25 m (82 ft) to the south-east from the originally planned location, which addressed the staff’s uniformity concern for the subsurface materials underneath the nuclear island. Furthermore, dynamic confirmatory sensitivity analyses performed and evaluated in Sections 2.5.2 and 3.7.1 of this report ensure uniform dynamic response. Accordingly, the staff considers RAI 44, Question 02.05.04-9 resolved.

2.5.4.4.7.2 *Conclusions Regarding Response to Dynamic Loading*

On the basis of the sufficiency of the field testing and similar data collected by various test methods, the staff concludes that the applicant adequately determined the dynamic properties of the subsurface rock. The staff also concludes that the applicant addressed Interface Item 2.12 that is related to the seismic parameters for peak ground acceleration, response spectra and shear wave velocity by providing geophysical field test data for input to dynamic analyses performed as presented in WLS COL FSAR Sections 2.5.2 and 3.7.1. Also, the staff concludes that the dynamic properties of the existing soils, proposed engineered fill, and existing concrete were reliably established either by field testing, or estimated by reliable correlations. Future laboratory testing planned by the applicant will confirm the estimated dynamic properties of the engineered fill. Therefore, the staff concludes that the requirements of 10 CFR Part 50 and 10 CFR Part 100 have been adequately addressed.

2.5.4.4.8 Liquefaction Potential

The AP1000 DCD requires that no liquefaction is allowed beneath the Seismic Category I and Seismic Category II structures and the immediate surrounding area. WLS COL FSAR Section 2.5.4.8 states that the WLS site has a liquefaction potential of low to none. The applicant based this conclusion on the fact that the Seismic Category I structures will be constructed on top of either bedrock or concrete fill, and that the non-safety-related structures, including the Seismic Category II annex building adjacent to the nuclear island, will be founded on engineered fill compacted to 96 percent modified Proctor density. The staff reviewed the AP1000 DCD requirements for liquefaction, WLS COL FSAR Section 2.5.4.8, the applicant’s response to liquefaction related RAIs, and considered the foundation conditions underlying the nuclear island and the adjacent non-safety-related and Seismic Category I structures to evaluate the liquefaction potential at the WLS site.

2.5.4.4.8.1 *Nuclear Islands*

The nuclear islands will be founded either on rock or concrete overlying rock and, therefore, the staff concludes that liquefaction is not possible for the materials beneath the nuclear islands of WLS.

2.5.4.4.8.2 *Granular Backfill*

The staff noted that the backfill surrounding the nuclear island will be a granular material with a low percentage of fines that meets the SCDOT gradation requirements and the fill material will be obtained from a SCDOT-approved quarry. The particular gradation has not yet been selected, but the applicant provided data on three fill types; well-graded gravel, poorly-graded gravel, and well graded sand. The engineered fill will be placed in the field with 96 percent of modified Proctor relative compaction, and the granular material will support the non-safety-related structures. In RAI 44, Questions 02.05.04-1 and 02.05.04-5, and RAI 61, Questions 02.05.04-11, 02.05.04-12, 02.05.04-13, and 02.05.04-14, the staff requested that the applicant provide the results of the liquefaction analyses performed for the Group 1 fill material originally cited as the on-site backfill source. However, the applicant subsequently revised its liquefaction analysis when it replaced the Group 1 fill with the SCDOT granular fill, therefore the concerns raised by the staff in those RAIs no longer apply and accordingly, each of those RAIs are closed.

The applicant submitted a second liquefaction analysis using offsite borrow material with a compaction of 96 percent of modified Proctor density (ASTM D1557). The staff reviewed the liquefaction analyses and performed a simplified confirmatory analysis to check the applicant's conclusion. The staff determined that 96 percent relative compaction based on the modified Proctor specification results in a relative density of 80 percent. This was determined by the Lee and Singh relationship given in Equation 2.5.4-1 of this report.

$$RC = 80 + .02D_r \qquad \text{Equation 2.5.4-1}$$

Where: RC is relative compaction with respect to ASTM D1557, and D_r is relative density

Considering that the SW gradation would be the most critical gradation from the standpoint of liquefaction, the staff determined the $(N_1)_{60}$ from Figure 2.5.4-18 of this report to be 38, corresponding to a relative density of 80 percent. The applicant had used a different correlation that gave a more conservative $(N_1)_{60}$ of 30. The relationship between the cyclic stress ratio and $(N_1)_{60}$ published in Youd et al., shown in Figure 2.5.4-19 of this report, indicates that the liquefaction potential of clean granular materials having a $(N_1)_{60}$ of 30 or greater is nil. This is concluded because data points to the right of the "SPT Clean Sand Base Curve" shown in the figure are in the "no liquefaction" portion of the plot. Therefore, a $(N_1)_{60}$ of 30 or greater always resides to the right of the liquefaction/no liquefaction boundary. Therefore, the staff concludes that the applicant made a conservative assumption regarding the strength of the compacted soils, $(N_1)_{60}$ equal to 30, and that engineered fill placed to 96 percent relative compaction will provide adequate resistance to liquefaction under the SSE loading.

2.5.4.4.8.3 *Saprolite*

The applicant also performed liquefaction analyses for the saprolite that will remain to support the engineered fill. The saprolite will be removed from under the safety-related portions of the Seismic Category II structures, but saprolite will be left in place at some locations under non-safety-related buildings or non-safety-related portions of Seismic Category II buildings, specifically the annex and turbine buildings. The applicant found that for corrected N-values, N_{60}, exceeding 15, factors of safety were greater than 2.0. The applicant stated that saprolite exhibiting SPT N_{60}-values less than 15 will be removed.

The staff performed a simplified confirmatory liquefaction analysis based on RG 1.198 and determined that for soil with a N_{60} value of 15, at a depth of roughly 5.6 m (18.5 ft) from the existing ground surface, and 15 percent fines, the cyclic resistance ratio (CRR) was approximately 0.25 from Figure 2.5.4-19 of this report. Since the relationship in Figure 2.5.4-19 of this report is based on a magnitude 7.5 earthquake, and the local earthquake is a magnitude (M_w) 5.1, a Magnitude Scaling Factor (MSF) of 2.68 was applied to yield a CRR of 0.67. The correction factor is a lower bound correction factor recommended in Youd, et al.

$$MSF = {10^{2.24}}\big/{M_w^{2.56}} \qquad \text{Equation 2.5.4-2}$$

The factor of safety is computed as:

$$FS = \frac{CRR}{CSR} \times MSF \qquad \text{Equation 2.5.4-3}$$

A conservative cyclic stress ratio (CSR) for the Magnitude M_w 7.5 earthquake was determined from Equation 2.5.4-4 of this report for the interface at depth of 5.6 m (18.5 ft), because this profile resulted in the greatest peak ground acceleration of 0.345g.

$$CSR = \frac{\tau_{av}}{\sigma'_{vo}} = 0.65\left(\frac{a_{max}}{g}\right)\left(\frac{\sigma_{vo}}{\sigma'_{vo}}\right) r_d \qquad \text{Equation 2.5.4-4}$$

Where:

τ_{av} = average cyclic shear stress

a_{max} = maximum ground acceleration at the ground surface

g = acceleration of gravity

σ_{vo} = total overburden stress

σ'_{vo} = effective overburden stress

r_d = reduction factor based on depth of soil

Based on the above equations, the staff calculated a cyclic stress ratio, CSR, of 0.36, which results in a factor of safety (FS) of 1.86. From this result, the staff concludes that saprolite with N_{60} of 15 or greater provides adequate resistance against liquefaction and may remain to support the engineered fill. The staff also reviewed the excavation plan and boring logs in areas that show saprolite supporting engineered fill below safety-related structures to observe the field N-values that will remain below the excavation line. To convert the field N-values to N_{60} values the staff consulted WLS COL FSAR Appendix 2AA and found that the average energy imparted was 78.1 percent in Boring B-1023. Thus, a N_{60} value of 15 is equivalent to a field N-value of 11.5. The staff determined that the field N values of saprolite that will remain below the planned excavation grade were all greater than 11.5. Based on the results of the confirmatory SPT liquefaction analysis, the staff concludes that the engineered fill placed to the compaction specifications of 96 percent modified Proctor maximum density will not liquefy, and saprolite below the planned excavation grade with N_{60} values of 15 or greater will not liquefy.

However, in Revision 7, submitted on May 9, 2013, WLS COL FSAR Section 2.5.4.8 indicates that the immediate underlying material for some non-seismic portions of the annex and turbine buildings will be saprolite soils, instead of compacted engineered granular fill over saprolite soils as originally planned. Since the SPT N_{60} values of the saprolite soil are between 11 and 30 as listed in WLS COL FSAR Table 2.5.4-211, and the possible maximum groundwater level will be above this layer of soil, this saprolite soil is not totally liquefaction potential free. Therefore, in RAI 112, Question 02.04.05-18, the staff requested that the applicant address the liquefaction potential of the saprolite and the impact of such materials underneath portions of the foundation on the differential settlement between the nuclear island and adjacent buildings.

In an April 24, 2014, response to RAI 112, Question 02.05.04-18, the applicant stated that because of the relocation of WLS, the southernmost area of the non-seismic annex building for Unit 1 lies over existing concrete, which in turn is underlain by continuous rock. Therefore, no saprolite will exist beneath the southernmost area of the non-seismic annex building for WLS Unit 1. The applicant further stated that the existing saprolite material will be removed underneath the non-seismic portion of the WLS Unit 2 turbine building. The only area where saprolite soil will remain beneath the AP1000 non-seismic buildings is beneath the southernmost end of the non-seismic part of the Unit 2 annex building. However, the base of the open excavation has $(N_1)_{60}$ values equal to 26-27, and thus may be considered as highly resistant to liquefaction. Since the liquefaction potential is negligible at the planned site based on the properties of the subsurface materials, the applicant stated that the foundation performance of buildings supported on the granular fill will meet the AP1000 DCD differential settlement criterion. The applicant then proposed changes to the WLS COL FSAR.

Based on the review of the response to RAI 112, Question 02.05.04-18, the staff concludes that the liquefaction potential at the planned site is negligible because the whole power block and non-seismic annex building of Unit 1 will be founded over existing concrete, which in turn is underlain by continuous rock; and the remaining saprolite soil underneath a portion of the Unit 2 annex building has properties that are highly resistant to liquefaction. The staff confirmed that WLS COL FSAR Revision 11, incorporated proposed changes to ensure the standard design requirement on liquefaction potential at the proposed site are met. Accordingly, the staff considers RAI 112, Question 02.05.04-18 resolved.

2.5.4.4.8.4 *Conclusion for Liquefaction Potential*

Based upon its review of WLS COL FSAR Section 2.5.4.8 and staff's independent confirmatory calculations, the staff concludes that no liquefaction can occur below the nuclear island as the bedrock formation and concrete fill are non-liquefiable, nor will liquefaction occur in either the engineered fill or the saprolite that support the Seismic Category II and non-safety-related structures. Since the applicant demonstrated that liquefaction was not possible in the rock and concrete foundation materials underlying WLS and demonstrated that liquefaction would not occur in the densely compacted backfill underlying the non-safety-related structures that border the nuclear islands, the staff concludes that the applicant adequately addressed and closed COL Information Item 2.5-9. The staff concludes that the liquefaction analysis described in WLS COL FSAR Section 2.5.4.8 forms an adequate basis for the assessment of the potential for liquefaction at the WLS site, and meets the requirements of 10 CFR Part 50, Appendix S; and 10 CFR 100.23.

2.5.4.4.9 Earthquake Site Characteristics

In WLS COL FSAR Section 2.5.4.9, the applicant summarized the performance-based site-specific GMRS and FIRS, which were completed in accordance with RG 1.208. The technical evaluation of the development of GMRS and FIRS is included in Section 2.5.2 of this report. The history of the surface faulting for the site and evaluation of surface deformation are evaluated in Section 2.5.3 of this report.

2.5.4.4.10 Static Stability

The staff focused its review of WLS COL FSAR Section 2.5.4.10 on the determination of the representativeness of the soil and rock properties used in the analyses, the applicability of the methods of analysis used, and the applicant's statement that the estimated performance was within the AP1000 bounding criteria. The applicant presented the assumptions, methodologies, and technical references used for its evaluation of the foundation conditions relative to the proposed demands created by excavation, structure and backfill loads, and lateral pressures exerted against the exterior walls of the nuclear island.

2.5.4.4.10.1 *Bearing Capacity*

AP1000 DCD, Tier 1, Chapter 5, states that a site is acceptable if its site characteristics fall within the AP1000 plant site design parameters in the AP1000 DCD Tier 1, Table 5.0-1. An extract of this table for bearing capacity is presented in Table 2.5.4-3 of this report. The applicant evaluated the ultimate bearing capacity of the rock foundation for WLS using two methods of analysis. An empirical method based on average RQD, cited in Peck, Hanson, and Thornburn, was used to determine that the allowable bearing pressure provided lower results than using Tezaghi's bearing capacity equation.

Table 2.5.4-3 AP1000 DCD Bearing Capacity Requirements

AP1000 DCD Table 5.0-1	
Average Allowable Static Soil Bearing Capacity	Greater than or equal to 8900 lb/ft^2 over the footprint of the nuclear island at its excavation depth.
Maximum Allowable Dynamic Bearing Capacity For Normal Plus Safe Shutdown Earthquake (SSE)	Greater than or equal to 35,000 lb/ft^2 at the edge of the nuclear island at its excavation depth.

Using the relationship presented in Peck et al., the applicant determined a minimum allowable bearing capacity of 9,097 kPa (190,000 psf) at Unit 1, and 11,587 kPa (242,000 psf) at Unit 2. In RAI 44, Question 02.05.04-6, the staff requested that the applicant clarify if it considered the lower RQD values obtained in the northwest corner of WLS Unit 1 in the determination of the allowable bearing capacity. In a December 23, 2008, response to RAI 44, Question 02.05.04-6, the applicant considered the lower RQD values and provided a summary of the pertinent RQD values. Table 2.5.4-4 of this report summarizes the RQD values the applicant used in the determination of the allowable bearing capacity provided in response to this RAI. In every instance, the allowable bearing capacity, which includes a FS of 3, was greater than the applied maximum bearing pressure.

Table 2.5.4-4 Summary of Minimum RQD and Allowable Bearing Capacity at WLS Unit 1

Boring	Rock Elevation Range, ft MSL		Foundation Width, ft	Minimum RQD_{avg}, %	Allowable Bearing Capacity On Rock, ksf	Comments $q_{allow} > q_{applied}$
	Top	Bottom				
B-1001	524.1	459	91	86	349	Yes
B-1001A	550.0	459	91	82	309	Yes
B-1002	535.5	459	91	65	190	Yes
B-1003	533.2	497.2	91	86	349	Yes
B-1004	544.0	459	91	86	349	Yes
B-1004A	528.5	459	91	79	279	Yes
B-1011	537.7	459	91	70	214	Yes
B-1074A	505.2	459	91	95	492	Yes
B-1075A	511.5	459	91	91	422	Yes

The staff performed a bearing capacity confirmatory analysis by first verifying the average RQDs as provided in the boring logs included in WLS COL FSAR Appendix 2AA. The staff noted that, based on the boring logs, the average RQD within the zone of influence (a depth approximately equal to the minimum width of the basemat) was equal to approximately 65 percent at WLS Unit 1 and 80 percent at WLS Unit 2. These values were assumed in the applicant's analysis as shown in Table 2.5.4-4 of this report. Using the Peck relationship shown graphically in Figure 2.5.4-20 of this report, the staff obtained approximate ultimate bearing capacity values of 9.125 and 14 MPa (190,600 and 292,400 psf) at WLS, respectively. These values closely match the values given by the applicant and provide factors of safety of greater than 3 in the static case and greater than 1.5 in the dynamic case. Due to this agreement between the applicant's estimations and the staff's calculations, the staff considers RAI 44, Question 02.05.04-6 resolved.

WLS COL FSAR, Revision 7, submitted on May 9, 2013, WLS COL FSAR Section 2.5.4.10.1.1, states, "[t]he applied seismic loading may exceed, by a relatively small amount, the AP1000 DCD value as a result of the site-specific seismic loading." Since the dynamic bearing capacity site parameter is a Tier 1 requirement, any exceedance will result in a departure. Therefore, in RAI 112, Question 02.05.04-19, the staff requested that the applicant provide details on how the calculated loading differs from the AP1000 DCD value, and properly address this departure in this application. In an April 11, 2014, response to RAI 112, Question 02.05.04-19, the applicant clarified that the site-specific maximum bearing pressure is approximately 1,103 kPa (23,030 psf) based on site-specific structural stability analyses, which is significantly less than the AP1000 DCD site acceptance characteristic of 1,675 kPa (35,000 psf). The site-specific analyses also confirmed that no foundation liftoff is expected when the nuclear islands are subjected to the nuclear island FIRS. The applicant then proposed a revision to the WLS COL FSAR to clearly state the dynamic bearing capacity at the WLS site. As the applicant clarified that the calculated site-specific maximum dynamic bearing pressure is less than that specified in the AP1000 DCD, the staff concludes that there is no departure related to this design parameter. The staff further confirmed that FSAR Revision 11 incorporated proposed changes to properly address the dynamic bearing capacity at the proposed site. Accordingly, the staff considers RAI 112, Question 02.04.05-19 resolved.

Based on the review of WLS COL FSAR Sections 2.5.4.10 and 3.8.5.5, and the applicant's responses to related RAIs, the staff concludes that the applicant showed that the allowable bearing capacity of the rock supporting the nuclear islands of WLS Units 1 and 2 is greater, with FS of 3, than the maximum bearing pressure determined from the analyses described in AP1000 DCD Appendix 3G (35,000 psf (1,675 kPa) under all combined loads including the safe shutdown earthquake). Therefore, the staff finds that COL Information Item 2.5-10 is adequately addressed and is closed. The staff also notes that the applicant used material properties obtained from field and laboratory tests in bearing capacity analyses to satisfy Interface Item 2.13, related to the required bearing capacity of foundation materials. The staff concludes that the allowable bearing capacity as determined from the empirical relationship presented in Peck et al. provides an adequate bearing capacity for both the static and dynamic loading conditions.

2.5.4.4.10.2 *Settlement*

The applicant calculated settlement using three methods proposed by: Steinbrenner (Bowles, 1988), the U.S. Army Corps of Engineers (USACE, 1990), and Boussinesq (Li, 1988). Each of these methods is based on the theory of elasticity. The applicant also used an empirical relationship obtained from Peck et al. (1974) that uses the RQD of the rock to predict settlement. The applicant computed settlements of approximately 0.127 cm (0.05 in.) at WLS Units 1 and 2 using the elastic theory derived methods. Using the empirical method of Peck et al. (1974), the applicant computed settlements of 0.058 cm (0.023 in.) at Unit 1 and 0.038 cm (0.015 in.) at Unit 2. The computed settlements from those methods are small and fall within the design settlement criteria established in the AP1000 DCD: 7.62 cm (3 in.) for total settlement, and 1.27 cm in 15.24 m (0.5 in. in 50 ft) for differential settlement.

The staff inspected the boring logs to verify input data. Based on the strengths of the rock and the computed bearing capacity factors of safety, the staff concludes that the settlements will be small and in the elastic range, which verifies the reasonableness of the applicant's approach. The staff also performed its own confirmatory analysis using elastic theory. The staff selected the four borings with V_s measurements at WLS Unit 1 to estimate possible maximum differential settlement. The average elastic modulus values to a depth of approximately 45.7 m (150 ft), the maximum depth of these borings, were conservatively corrected to account for strain using a correction factor of 0.5. Although settlements were only calculated to a depth of 45.7 m (150 ft) due to the limitation of the boring depths, the staff concludes that this was sufficiently deep because the stresses decrease as depth below the basemat increases, and the rock elastic modulus remains the same or increases as observed by the V_s measurements made in borings taken to depths of 76.2 m (250 ft). The calculated settlements between the four borings were compared: the largest settlement was 0.129 cm (0.051 in.) and the least settlement was 0.058 cm (0.023 in.). The AP1000 assigns an allowable total settlement of 7.62 cm (3.0 in.) and allowable differential settlement of 1.27 cm in 15.24 m (0.5 in. in 50 ft). Accordingly, the staff concludes that the estimated total settlement and differential settlement of the WLS are within the AP1000 DCD criteria.

The staff noted that the AP1000 DCD limits differential settlement between the nuclear island and the surrounding structures to 7.62 cm (3 in.); however, the applicant did not present differential settlement data between structures. Therefore, in RAI 66, Question 02.05.04-15, the staff requested that the applicant provide the differential settlements between the nuclear island and each of the surrounding structures. In June 2, 2009 and October 30, 2009, responses to RAI 66, Question 02.05.04-15, the applicant presented the differential settlements between the nuclear island and the surrounding structures. The applicant based the settlement calculations on the method of Peck et al. (1974). Table 2.5.4-5 of this report summarizes the results.

Table 2.5.4-5 Allowable Bearing Pressure Based on Limiting Settlement (WLS COL FSAR Table 2.5.4-229)

Structure	Subsurface	$q^{(a)}_{allow}$ (k/ft^2)	$q_{applied}$ (k/ft^2)	$q_{allow} > q_{applied}$	Anticipated Settlement (inches)
SW Sand Granular Backfill					
Annex Building	Granular Fill – SW	7.29	2.43	Yes	<2
Turbine Building	Granular Fill – SW	6.96	3.51	Yes	<2
Radwaste Building	Granular Fill – SW	7.24	1.31	Yes	<2
GP Gravel Granular Backfill					
Annex Building	Granular Fill – GP	10.93	2.43	Yes	<2
Turbine Building	Granular Fill – GP	10.44	3.51	Yes	<2
Radwaste Building	Granular Fill – GP	10.86	1.31	Yes	<2
GW Gravel Granular Backfill					
Annex Building	Granular Fill – GW	10.93	2.43	Yes	<2
Turbine Building	Granular Fill – GW	10.44	3.51	Yes	<2
Radwaste Building	Granular Fill – GW	10.86	1.31	Yes	<2

(a) For limiting settlement to 2 inches

The staff reviewed the applicant’s calculations and observed that total settlement is limited to less than 5.08 cm (2 in.) for each of the Seismic Category II or non-safety-related structures irrespective of the engineered fill type assumed in the calculations. The most critical case is associated with the Turbine building and its static bearing pressure of 98.6 kPa (3.51 ksf). The allowable bearing pressure (q_{allow}) for 5.08 cm (2 in.) settlement at the Turbine building is 340.4 kPa (7.11 ksf). Since a 340.4 kPa (7.11 ksf) of pressure is expected to produce 5.0 cm

(2 in.) total maximum settlement, for the actual bearing pressure of 98.6 kPa (2.06 ksf), the applied Turbine building load is expected to produce settlement of less than 5.08 cm (2 in.). This amount of total settlement compared to the negligible settlement of the nuclear island, falls within the 7.62 cm (3 in.) differential settlement criteria between structures imposed by the AP1000 DCD.

The staff concludes that the applicant resolved COL Information Item 2.5-12 by demonstrating through engineering analyses that the rebound, settlement and differential settlement of the Seismic Category I, Seismic Category II and non-safety-related structures were within the range of acceptable settlements required by the AP1000 DCD. Also, because the applicant provided settlement calculations to demonstrate that settlement of the nuclear islands on rock was negligible, the staff concludes that the applicant adequately addressed COL Information Item 2.5-16. The staff further concludes that based on this analysis, the fact that most of the settlement under the Seismic Category II and non-safety-related structures will occur during construction, and given the engineering properties of the granular fill, the differential settlement between the nuclear island and the surrounding structures will be negligible and is acceptable.

2.5.4.4.10.3 *Lateral Pressures*

The applicant calculated lateral earth pressures caused by the placement of backfill against the exterior walls of the nuclear island using Rankine's theory. Earth pressure calculations were made with conservatively assumed soil properties and the range of expected water table elevations. Dynamic lateral earth pressures had not been submitted; therefore, in RAI 44, Question 02.05.04-8, the staff requested that the applicant provide dynamic lateral earth pressures acting on below ground walls of the plant safety-related facilities.

In December 23, 2008, and October 30, 2009, responses to RAI 44, Question 02.05.04-8, the applicant provided the lateral stresses due to dynamic forces imposed by the SSE. The dynamic pressures were calculated in accordance with ASCE 4-98. The staff reviewed the ASCE 4-98 methodology and concludes that this method is a state-of-the-art and widely used method for non-yielding walls and, therefore, acceptable for estimating dynamic lateral earth pressures. The staff also concludes that the input parameters used by the applicant to calculate the dynamic lateral stresses are conservative. Accordingly, the staff considers RAI 44, Question 02.05.04-8 resolved.

However, although in revised WLS COL FSAR Chapter 2, Section 2.5.4.10.3, it states, "Westinghouse has evaluated the WLS site-specific lateral earth pressures and has determined that they are bounded by the standard AP1000 design pressures," there is no related table or figure to show that the site-specific lateral earth pressures are bounded by the standard AP1000 design. Therefore, in RAI 112, Question 02.05.04-20, the staff requested that the applicant provide necessary table(s) or figure(s) to confirm that the site-specific lateral earth pressures are bounded by the standard AP1000 design.

In an April 11, 2014, response to RAI 112, Question 02.05.04-20, the applicant stated that Westinghouse evaluated the WLS site-specific lateral earth pressures by comparing the site-specific pressures on the nuclear island below-grade walls for Load Combinations (LC) 1 through 9 in AP1000 DCD Table 3.8.4-2 to the corresponding pressures that were used in the AP1000 standard design. The analysis results showed that the site-specific lateral pressures on the nuclear island exterior walls below grade are bounded by the AP1000 design pressures for load combinations 1, 2, 3, 4, 5, 6, 8, and 9 in both the east-west (E-W) and north-south (N-S) directions, with the exception that the site-specific lateral pressure in Load Combination 7 (LC7) for the GW backfill material slightly exceeds the standard design lateral pressure. The applicant then indicated that the exceedance of lateral earth pressure under LC7 is mainly attributed to the lower groundwater level at the site and with the assumption of full passive lateral earth pressure. However, the maximum lateral displacement at the base of the nuclear island when subjected to the certified seismic design response spectra (CSDRS) is expected to be 3 to 5 mm (0.12 to 0.19 in.) and such small lateral displacements are not capable of developing the full passive earth pressure. Therefore, the applicant concluded that the site-specific nuclear island below-grade wall pressures resulting from the NI FIRS will be less than those used in the standard AP1000 design for this load combination.

To confirm its conclusion, the applicant presented the results of a sensitivity study that varied the groundwater levels with a conservative assumption of displacement of 5 mm (0.2 in.). This study results show that when varying groundwater levels from 2.44 to 5.49 m (8 to 18 ft) below ground, the maximum fraction of the fully mobilized passive earth pressure is 0.83, or the passive earth pressure on the nuclear island below-grade wall will not be fully mobilized. However, the site-specific pressures on below-grade walls for LC7 exceed those specified in the AP1000 standard design, therefore, this exceedance is identified as a departure from the AP1000 DCD. The applicant then proposed changes to the WLS COL FSAR accordingly and referred to WLS COL FSAR Section 3.8.4.4.4 for a detailed discussion of site-specific lateral earth pressure.

Based on the review of the RAI response and WLS COL FSAR Section 3.8.4.4.4, the staff concludes that the applicant adequately demonstrated that the passive earth pressure on the nuclear island below-grade wall will not be fully mobilized given the very small expected lateral displacement under design seismic loading conditions, and the sensitivity study results confirmed this conclusion. Therefore, the staff concludes that the site-specific maximum lateral earth pressure will be smaller than those calculated in the standard AP1000 design for all load combinations, and the applicant adequately addressed this issue. .

2.5.4.4.10.4 *Conclusions of Bearing Capacity, Settlement and Lateral Earth Pressures*

In WLS COL FSAR Section 2.5.4.10, the applicant considered the static and dynamic bearing capacities, settlement, and lateral stresses at the WLS site. Based on the analytical results, the staff concludes that the bearing capacity of the WLS metamorphic and igneous rock foundation is sufficient to meet both the static and dynamic loading demands imposed by the WLS nuclear islands with adequate factors of safety. The staff also concludes that the settlement of the foundation under the maximum static load imposed by the nuclear island on the bedrock was minimal for either unit, less than 0.129 cm (0.051 in.), and the least settlement was 0.058 cm (0.023 in.); therefore, both the differential and total settlements are smaller than those specified in the standard design. The staff further concludes that the applicant conducted a site-specific lateral earth pressure analysis that used a conservative approach and showed that the expected maximum lateral pressures applied to the nuclear island below grade walls at this site will be smaller than those specified in the standard design.

The staff reviewed WLS COL FSAR Section 2.5.4.10 and concludes that the applicant developed an accurate assessment of the static and dynamic stabilities at the WLS site that address COL Information Items 2.5-10 through 2.5-13, including the minimum static and dynamic bearing capacities, lateral earth pressures, and stability of facilities. The staff further concludes that the applicant resolved COL Information Item 2.5-13 by providing planned instrumentation programs proposed for monitoring the performance of the dewatering system during the excavation phase of the project. Since the nuclear islands are founded on rock, the staff concurs with the applicant that settlement monitoring is not required. The staff also concludes that the information provided with respect to the required bearing capacity of foundation materials is adequate to address Interface Item 2.13. Accordingly, the staff concludes that the applicant's information in WLS COL FSAR Section 2.5.4.10, with the close of all related confirmatory items, forms an adequate basis for the stability of subsurface materials and foundations of Seismic Category I structures at the site and meets the requirements of 10 CFR Part 50, Appendix A and Appendix S, and 10 CFR 100.23.

2.5.4.4.11 Design Criteria

WLS COL FSAR Section 2.5.4.11 describes design criteria considered in the design of the foundations for the WLS nuclear islands. The applicant referred to WLS COL FSAR Table 2.1-201 where the AP1000 DCD site parameter requirements are compared to the WLS site characteristics. The staff focused its review on design criteria used, design assumptions, design methods, calculated factors of safety, and conservatism in the analyses. The staff reviewed the cited references to ensure proper interpretation of the equations, assumptions and limitations on the theory and performed independent confirmatory analyses to check the applicant's results.

The staff concludes that the applicant utilized the AP1000 DCD design criteria in its evaluation of the WLS site, performed calculations for bearing capacity, settlement, static and dynamic lateral earth pressures in accordance with accepted industry standards, and evaluated liquefaction potential in accordance with RG 1.198 and state-of-the-art methodology. The staff further concludes that computed factors of safety for liquefaction of backfill, static and dynamic bearing capacity are acceptable. Accordingly, the staff concludes that the applicant addressed the AP1000 DCD Tier 1 requirements; that the applicant used state-of-the art engineering methodology in determining the engineering behavior of the foundations; and that the site exhibits factors of safety that are in accordance with acceptable nuclear industry standards. Therefore, the staff concludes that the information presented in WLS COL FSAR Section 2.5.4.11 is acceptable to satisfy the applicable criteria of 10 CFR Part 50.

2.5.4.4.12 Techniques to Improve Subsurface Conditions

The applicant stated that with minor exceptions the WLS Units 1 and 2 nuclear islands will be founded on continuous rock having RQD values greater than 65 percent and that rock with lower RQD value will be excavated and replaced with fill concrete. Details for using dental concrete to backfill narrow zones of deeper weathered rock were provided. The applicant also stated that a drainage network under the existing CNS concrete slab will require modification to prevent migration of backfill into the drains. The staff focused its review on the proposed bedrock surface preparation details, including the excavation of unsatisfactory rock, the backfilling of joints (dental concrete), and sealing of the open drainage ways under the CNS foundation.

The rock and existing concrete surfaces will undergo surface preparation in advance of concrete fill placement. This includes cleaning the bedrock surface of loose soil or weak rock, removal of rock protrusions and overhangs, backfilling of open joints with dental concrete, and roughening the surfaces to receive concrete. This procedure is standard practice for bedrock surface preparation for major structures placed on rock foundations. The staff concludes that those techniques are acceptable because they are consistent with the assumptions made in the static and dynamic foundation design. Geologic mapping of the rock surface will provide the documentation needed to ensure that the excavation was advanced to continuous rock, and that no geologic hazardous conditions exist that were not uncovered in the geotechnical exploration. Geologic mapping requirement is evaluated in Section 2.5.3 of this report.

Under the existing CNS foundation basemat, a network of drains exists and the applicant proposed to seal off the drainage system with concrete to prevent migration of backfill into the drains. The staff concludes that the placement of concrete as a plug to block the migration of backfill materials into the drainage system is acceptable. The staff considers the applicant’s plan for improving the subsurface conditions adequate and, therefore, concludes that the information is acceptable to satisfy the applicable criteria of 10 CFR Part 50.

2.5.4.5 *Post Combined License Activities*

There are no post COL activities related to this section.

2.5.4.6 *Conclusions*

Based on its review of WLS COL FSAR Section 2.5.4, the referenced design certification of AP1000 and the applicant’s responses to related RAIs, the staff concludes that the applicant adequately determined the engineering properties of the soil and rock underlying the WLS site through its field and laboratory investigations. The staff concludes that the applicant used the state-of-the-art field and laboratory testing methods, in accordance with RG 1.132, RG 1.138, and RG 1.198, to determine the required site-specific engineering properties for the WLS site and to ensure that these properties met the design criteria specified in the AP1000 DCD.

Based on the information provided in the WLS COL FSAR and related RAI responses from the applicant, the staff concludes that the subsurface profile underlying the WLS site was properly characterized, that state-of-the-art analytical methods were used with conservative input values to determine factors of safety and limits of deformation in subsurface material and foundation stability analyses, and that the applicant considered all aspects of the foundation design that could impact the safety-related SSCs. Specifically, the staff concludes that the applicant adequately determined: (1) the soil and rock static and dynamic properties through its field investigations and laboratory tests; (2) the response of the soil and rock to dynamic loading; (3) the liquefaction potential of the engineered fill; and (4) the static and dynamic stabilities, including the bearing capacity, settlement, and lateral earth pressures of the nuclear island and surrounding non-safety-related structures under static and seismic loading conditions.

The staff concludes that the applicant provided sufficient information in WLS COL 2.5-1 through WLS COL 2.5-13, WLS COL 2.5-16 and WLS COL 2.5-17 to adequately address COL information items pertaining to WLS COL FSAR Section 2.5.4, which met the requirements specified in the AP1000 DCD. Finally, as discussed above, the staff concludes that WLS COL FSAR Section 2.5.4 is acceptable and meets the applicable requirements of 10 CFR Part 50, Appendix A (GDC 2) and Appendix S; and 10 CFR 100.23.

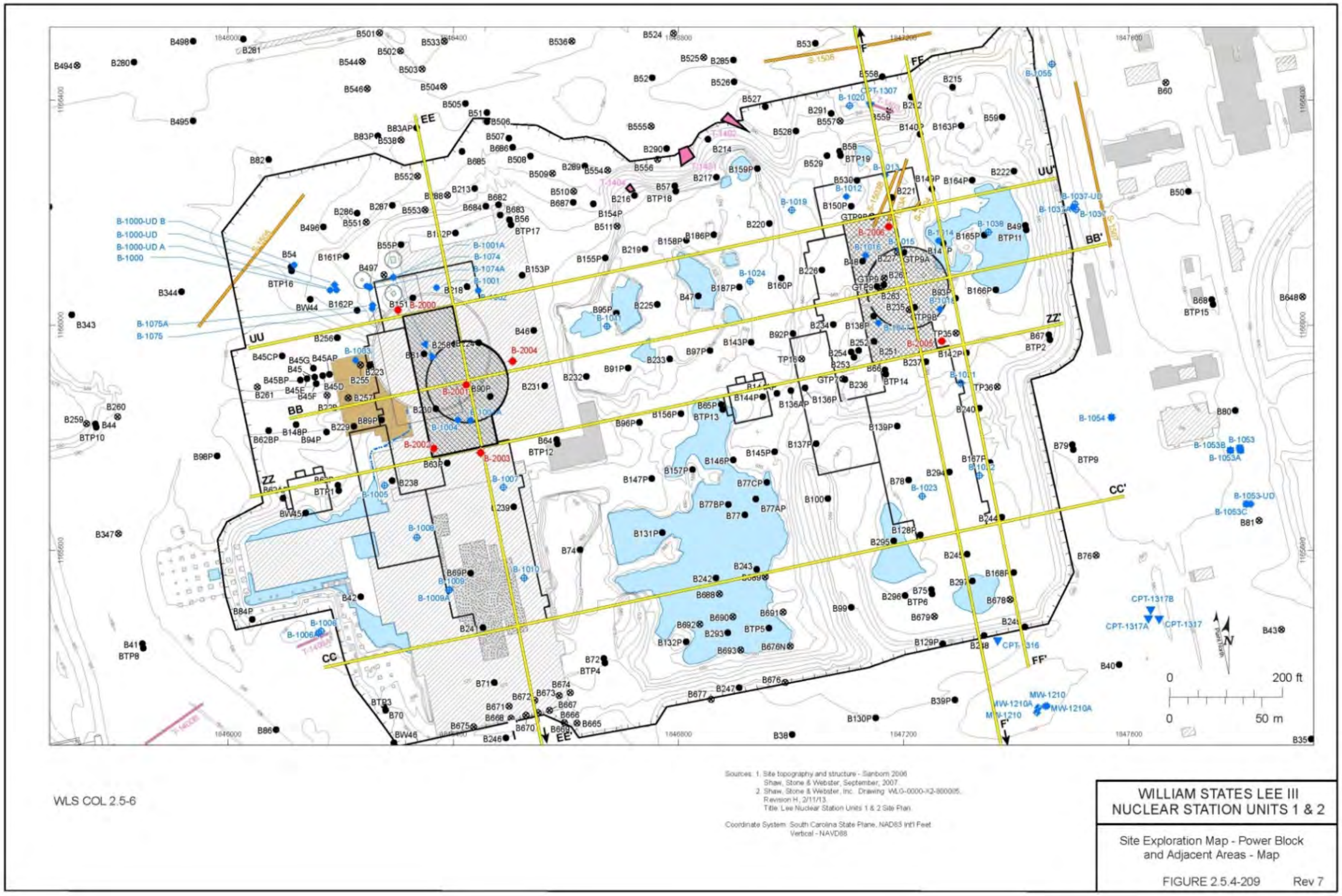


Figure 2.5.4-1 Exploration Points for the Lee Nuclear Station Power Blocks and Adjacent Areas (WLS COL FSAR Figure 2.5.4-209)

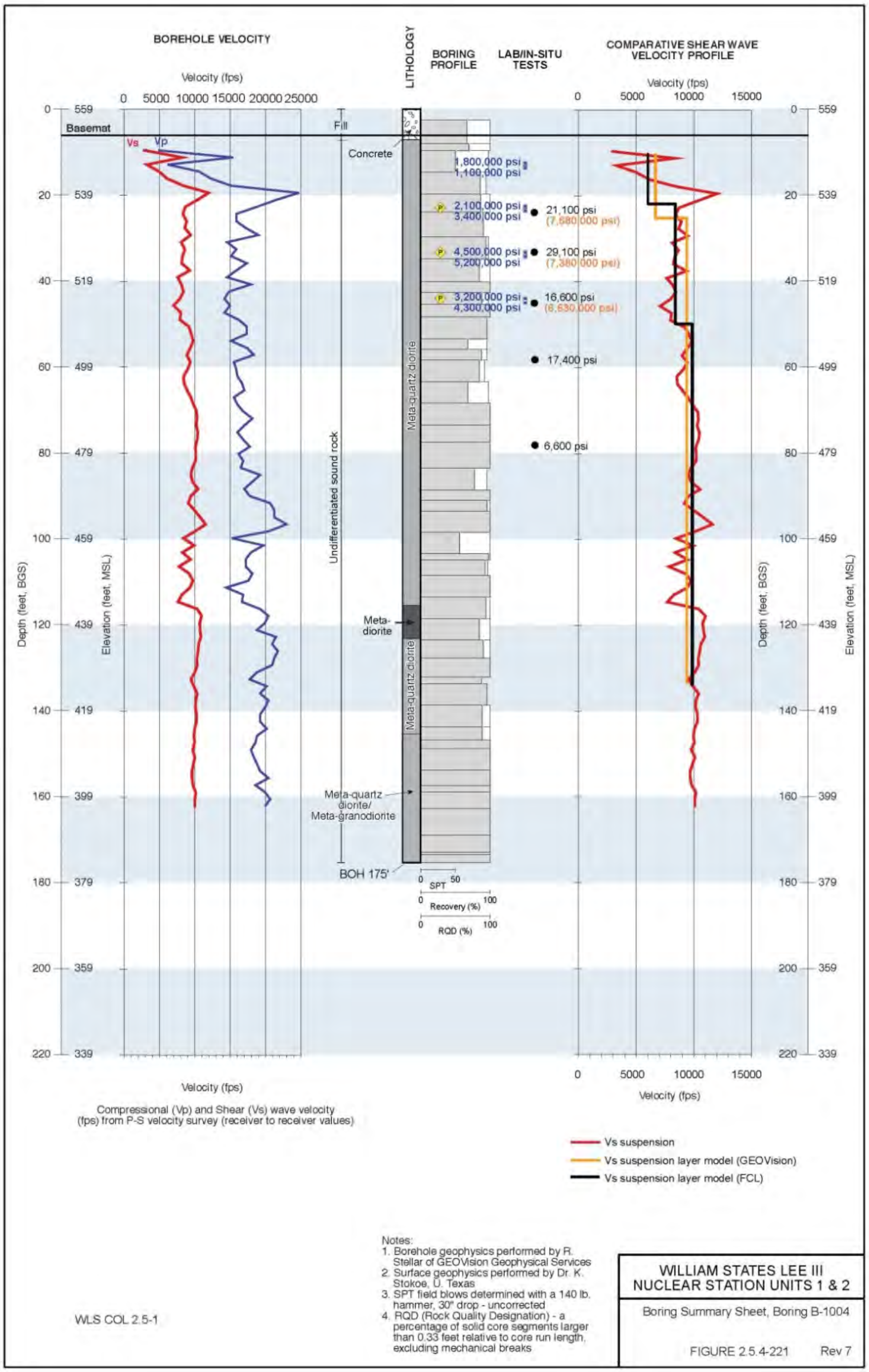
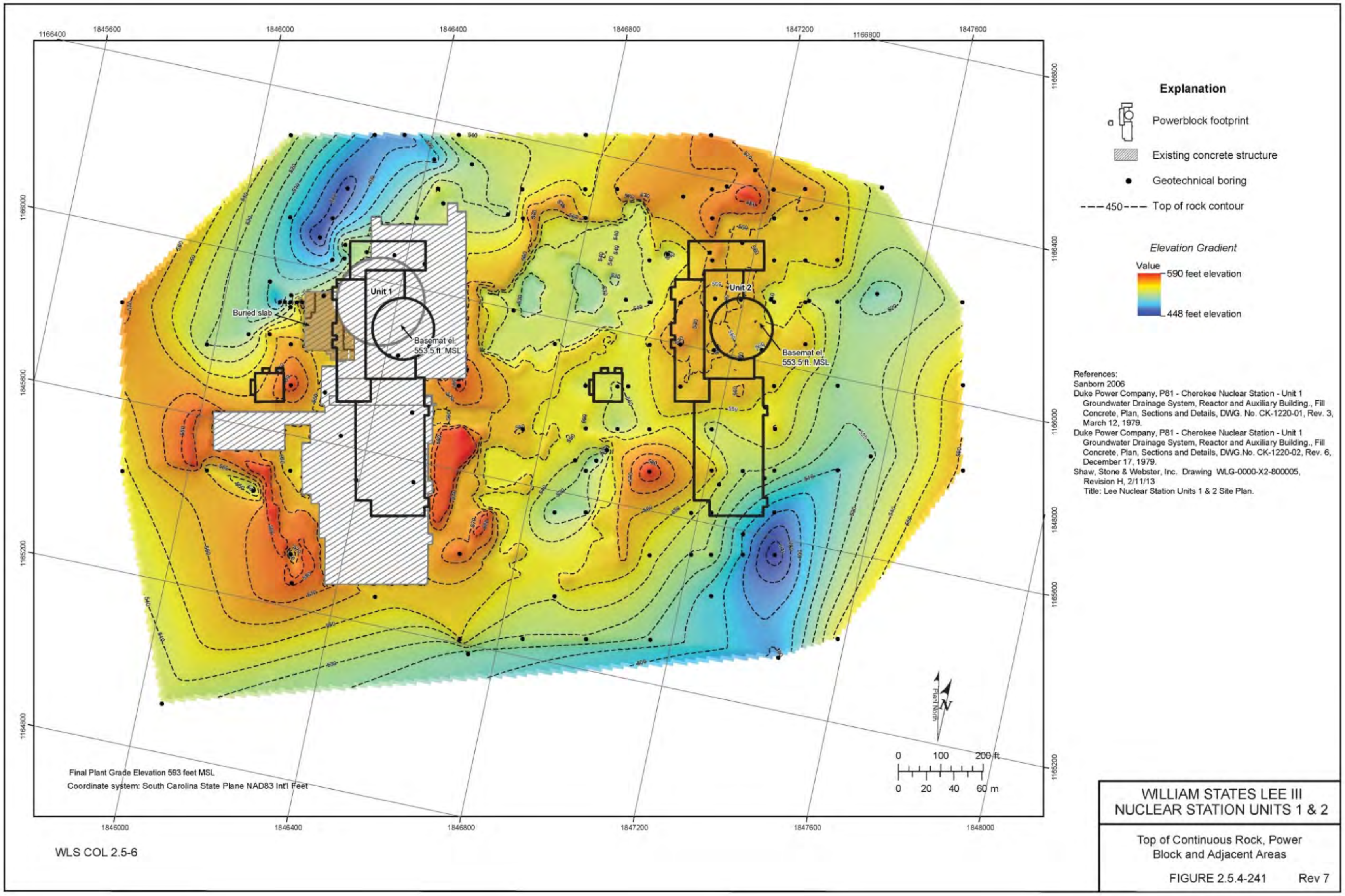


Figure 2.5.4-2 Boring Summary Sheet, Boring B-1004 (WLS COL FSAR Figure 2.5.4-221)

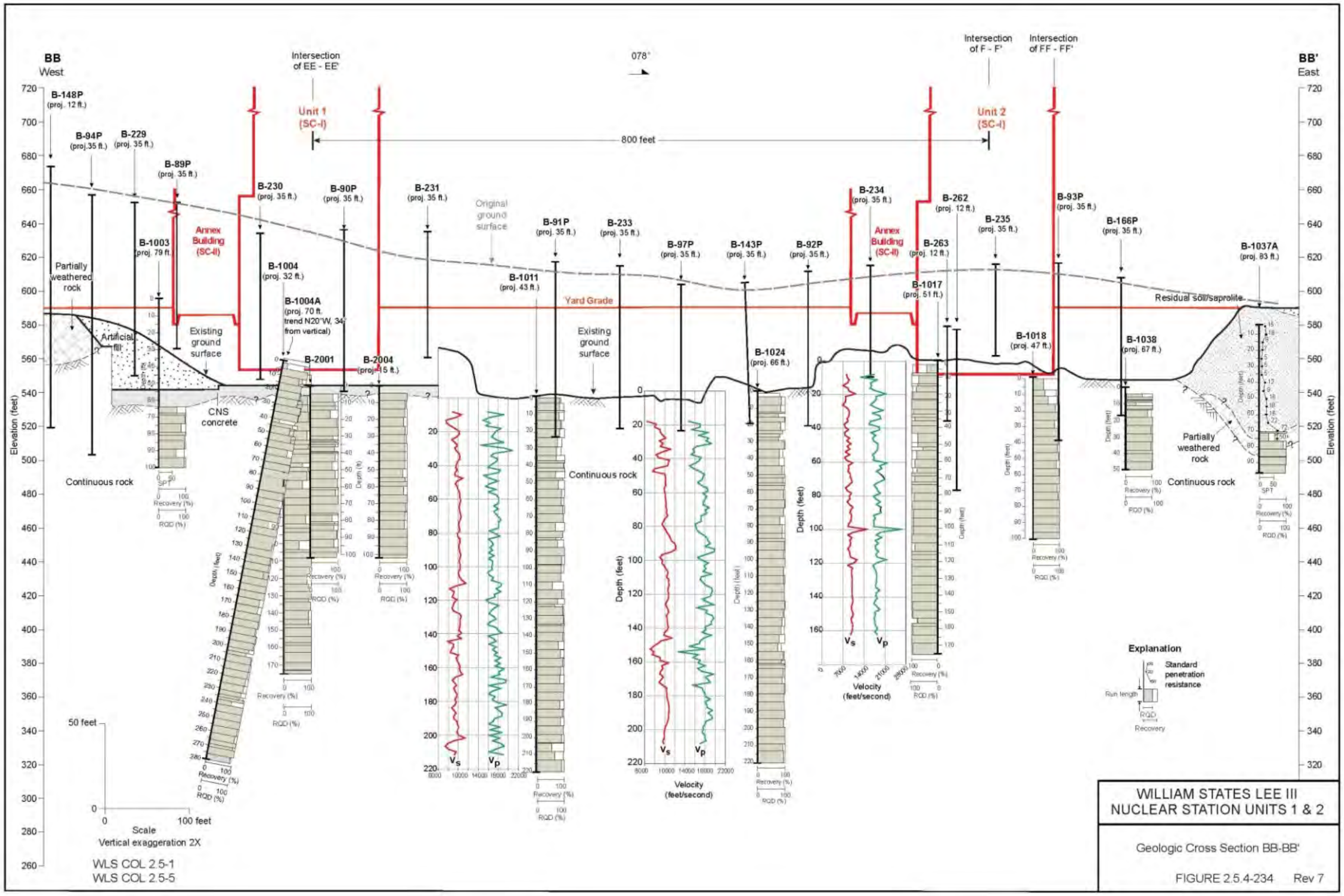


WILLIAM STATES LEE III
NUCLEAR STATION UNITS 1 & 2

Top of Continuous Rock, Power Block and Adjacent Areas

FIGURE 2.5.4-241 Rev 7

Figure 2.5.4-3 Top of Continuous Rock, Power Block and Adjacent Areas (WLS COL FSAR Figure 2.5.4-241)

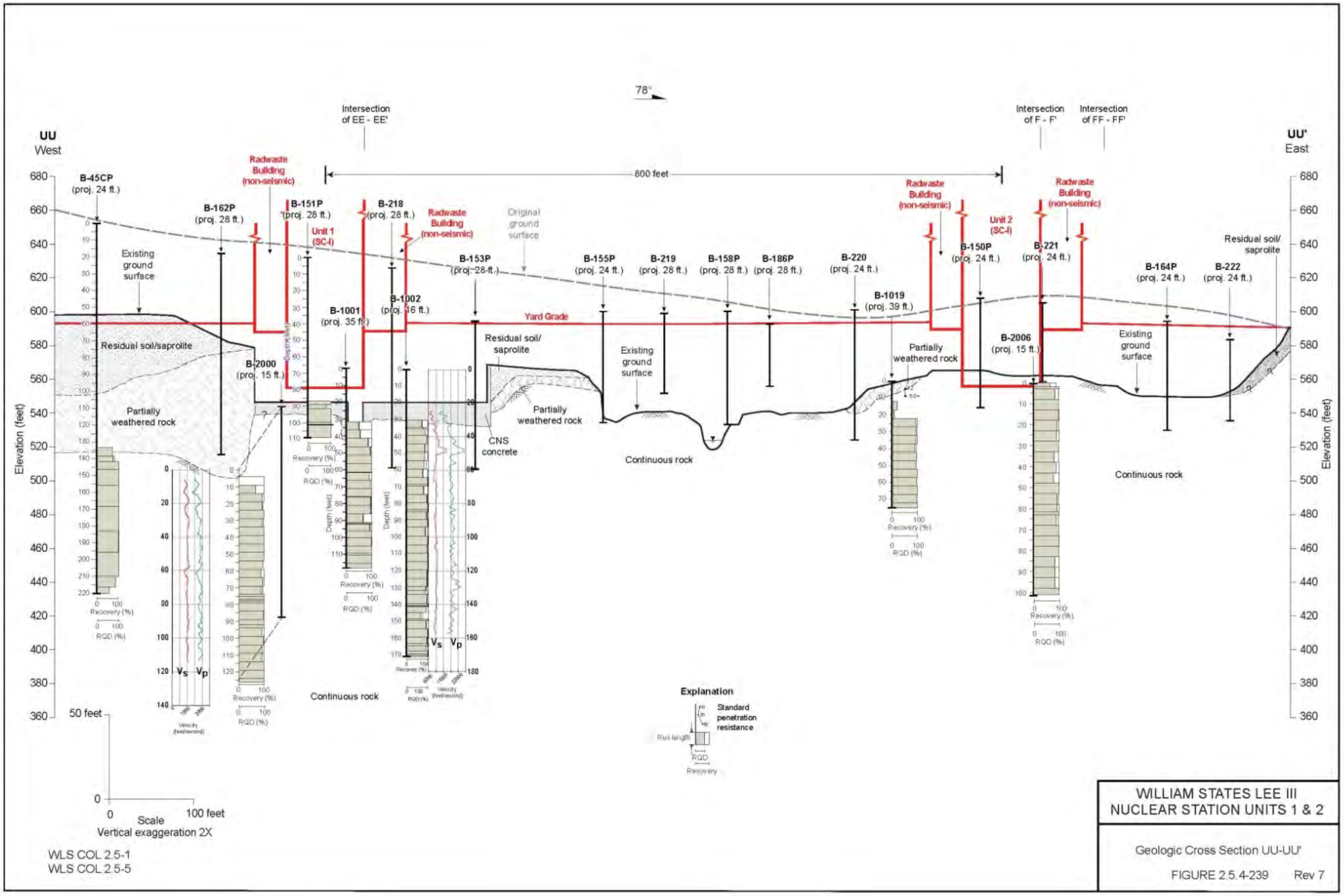


WILLIAM STATES LEE III NUCLEAR STATION UNITS 1 & 2

Geologic Cross Section BB-BB'

FIGURE 2.5.4-234 Rev 7

Figure 2.5.4-4 Cross-Section BB-BB' of West-East Profile through the LEE Unit 1 and Unit 2 centerline (WLS COL FSAR Figure 2.5.4-234)

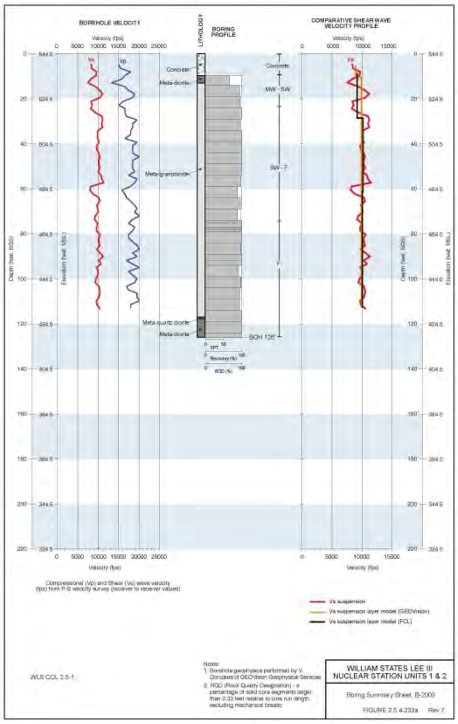


WILLIAM STATES LEE III
NUCLEAR STATION UNITS 1 & 2

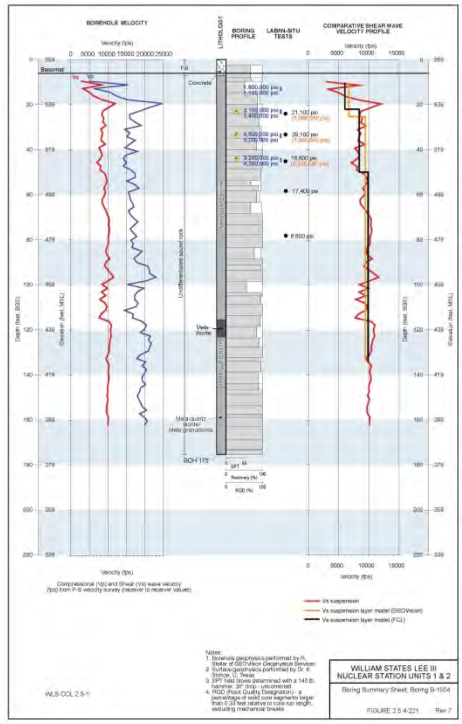
Geologic Cross Section UU-UU'

FIGURE 2.5.4-239 Rev 7

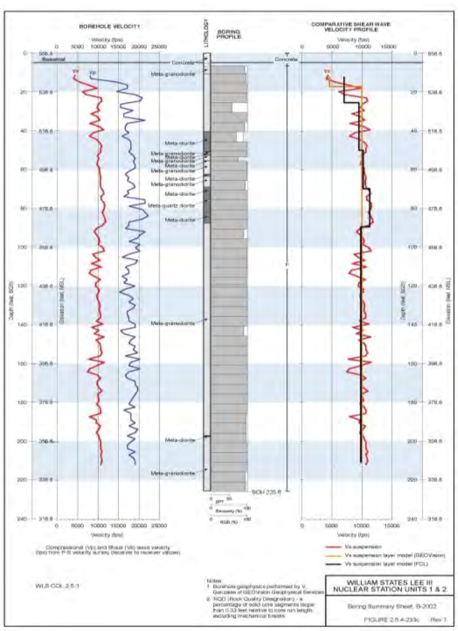
Figure 2.5.4-5 Cross-Section UU-UU’ of West-East Profile Through the North End of the LEE Units 1 and 2 Nuclear Island (WLS COL FSAR Revision 9, Figure 2.5.4-239)



Boring B-1004 Summary Sheet

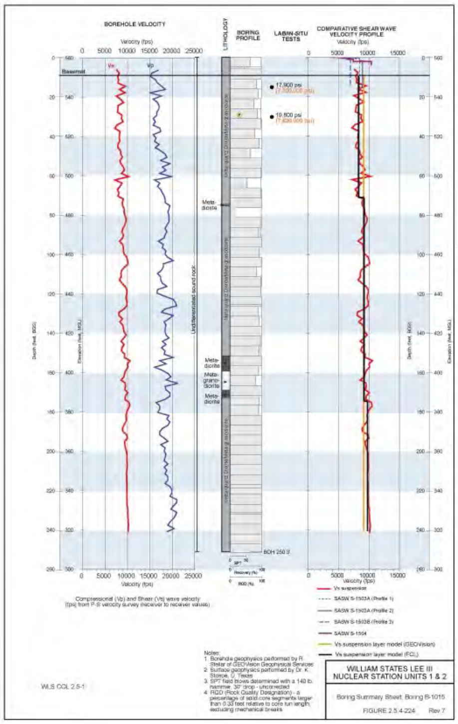


Boring B-2000 Summary Sheet

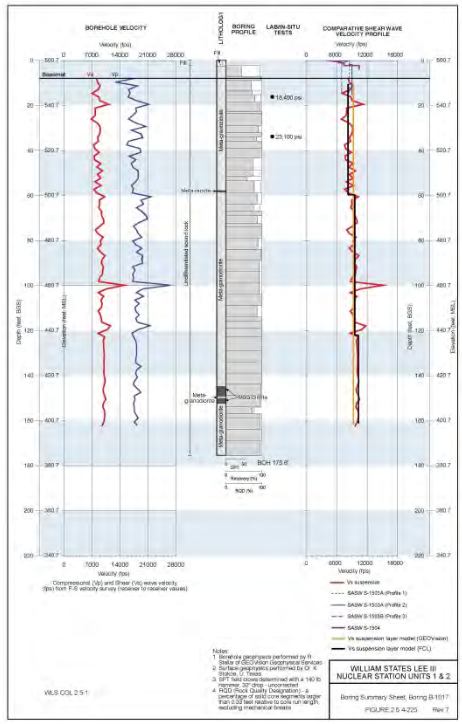


Boring B-2002 Summary Sheet

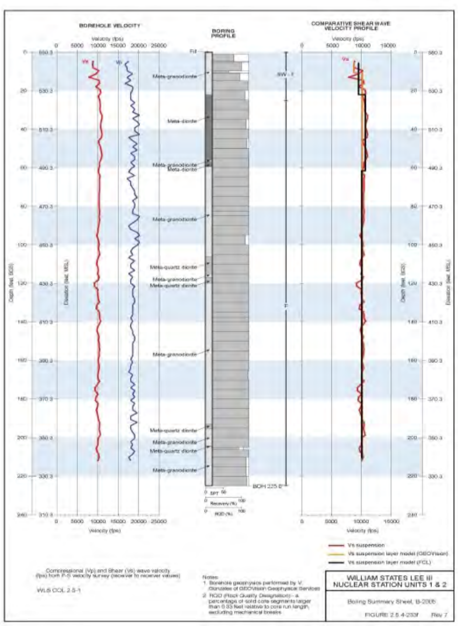
Figure 2.5.4-6 Lee Unit 1 Power Block Area Shear Wave Velocity Profiles



B-1015 Summary Sheet



B-1017 Summary Sheet



B-2005 Summary Sheet

Figure 2.5.4-7 Lee Unit 2 Power Block Area Shear Wave Velocity Profiles

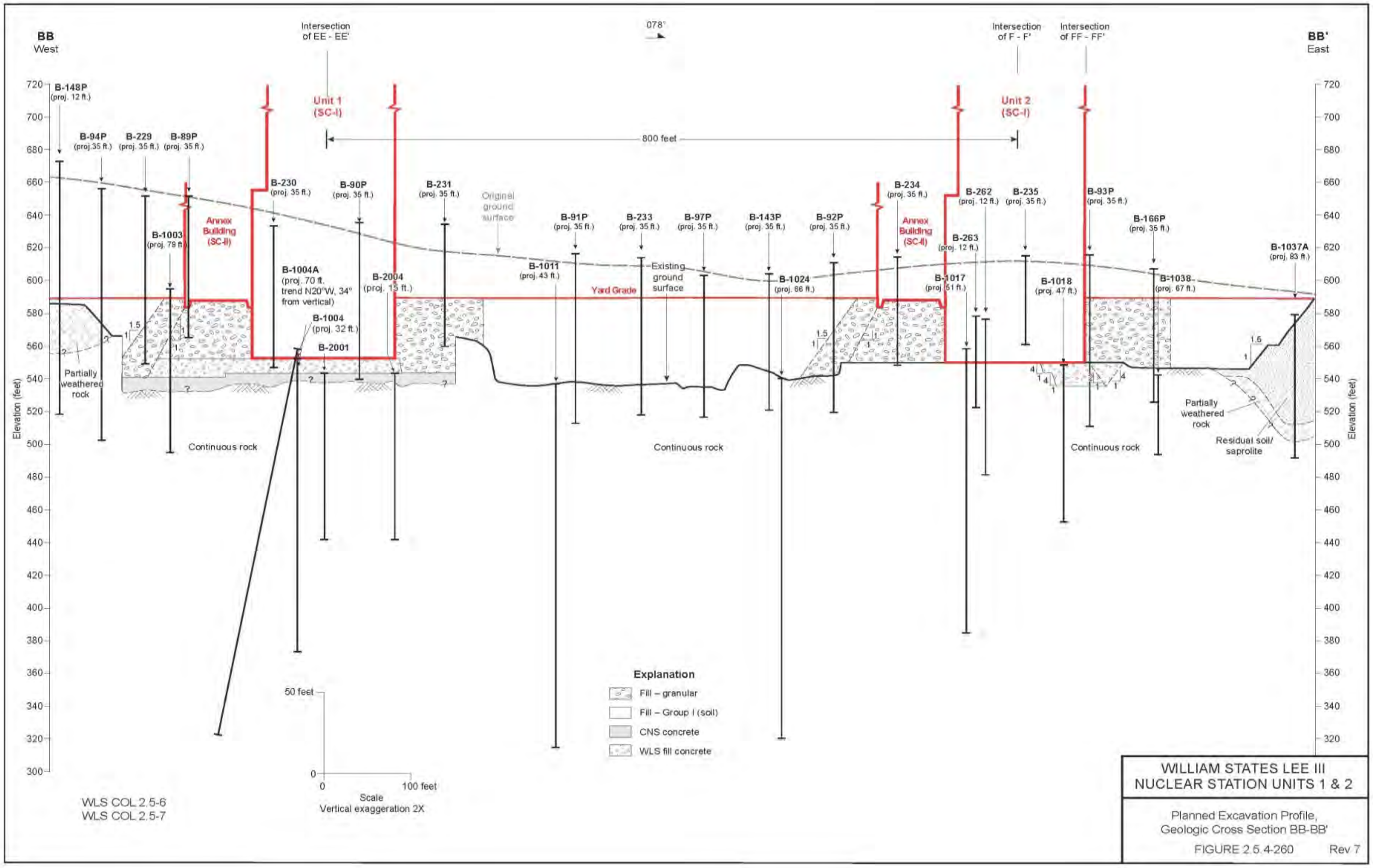


Figure 2.5.4-8 Planned Excavation Profile, Cross-Section BB-BB' (WLS COL FSAR Revision 9, Figure 2.5.4-260)

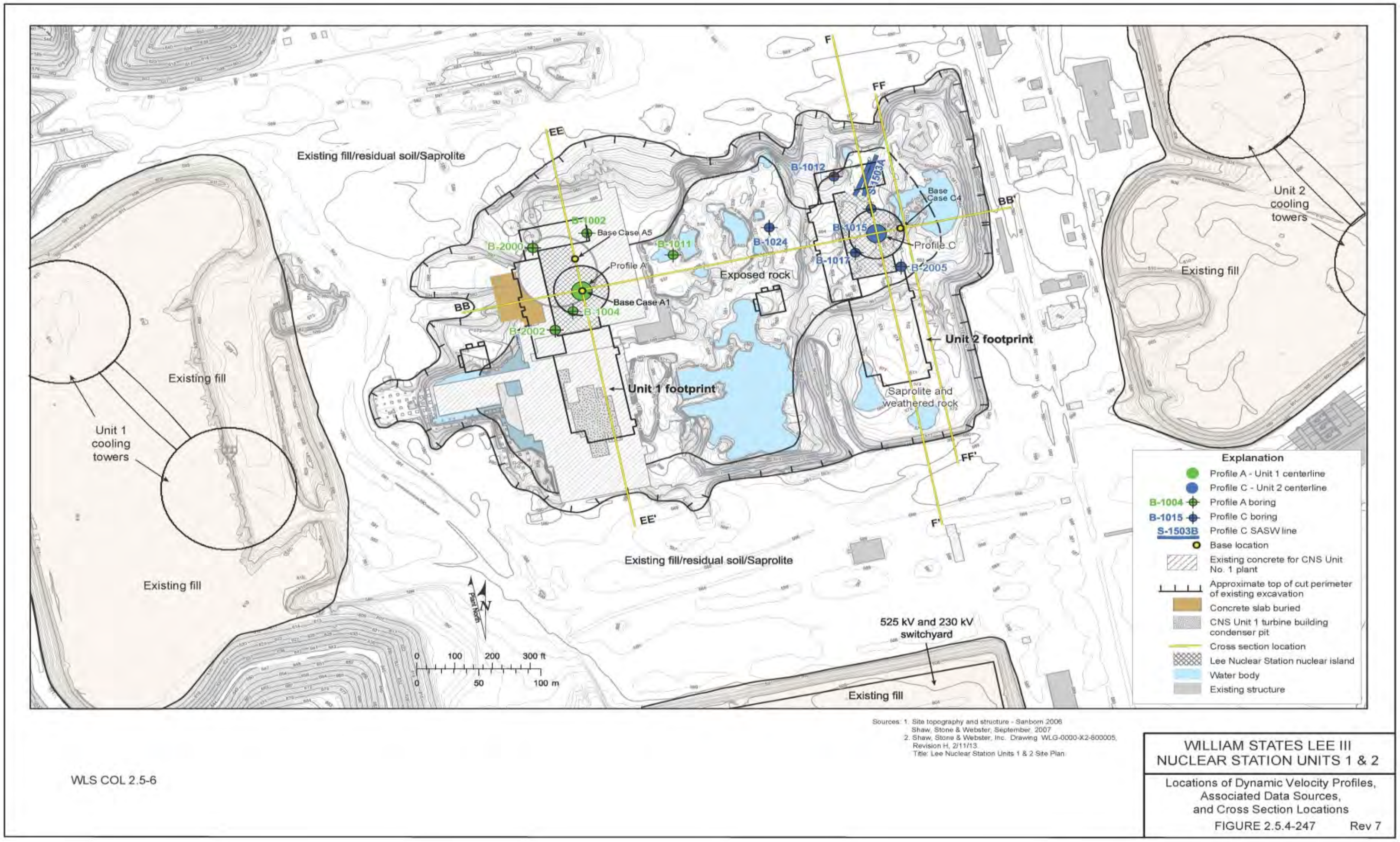


Figure 2.5.4-9 Locations of The Dynamic Profiles (WLS COL FSAR Revision 7, Figure 2.5.4-247)

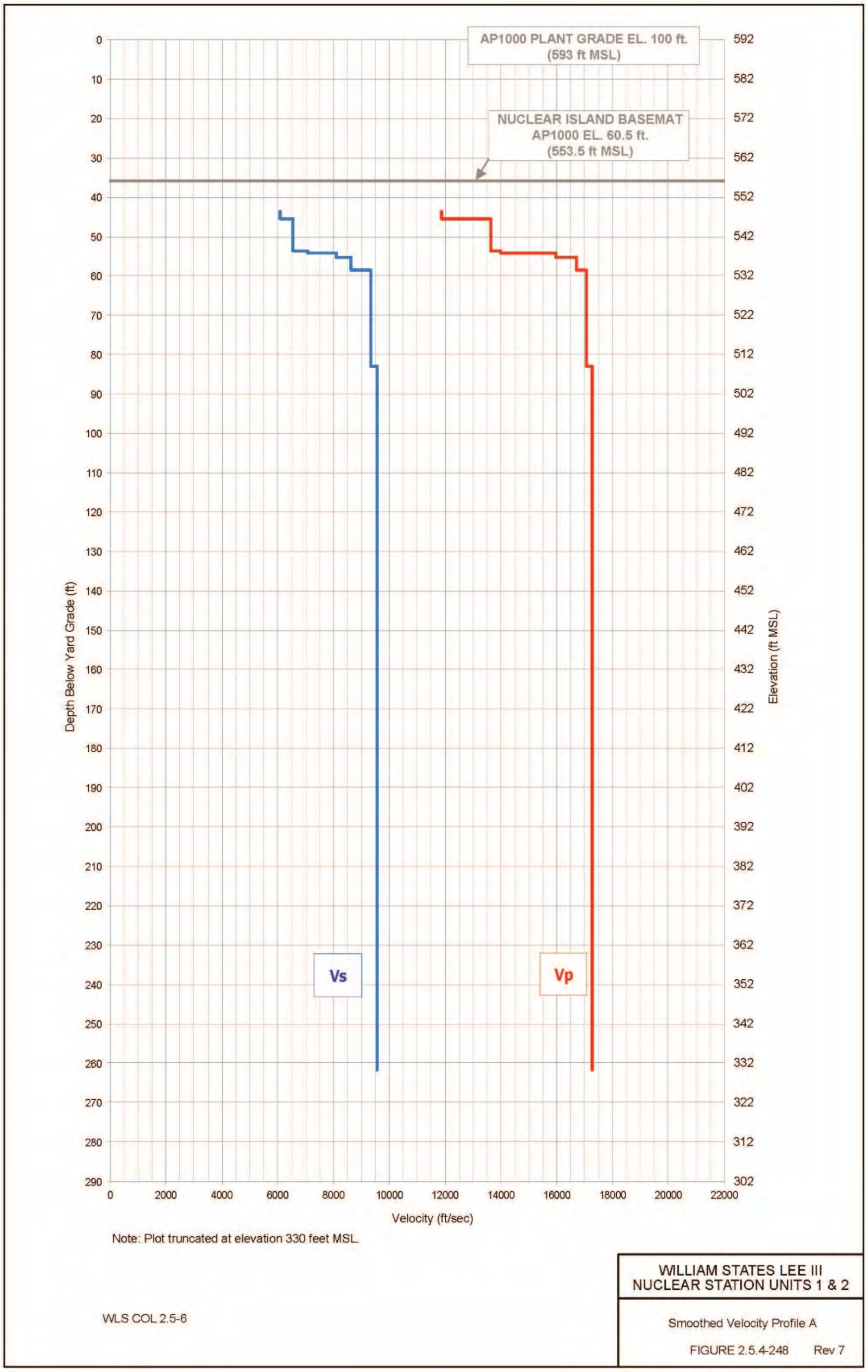


Figure 2.5.4-10 Lee Nuclear Station Unit 1 Centerline Dynamic Profile (WLS COL FSAR Revision 7, Figure 2.5.4-248)

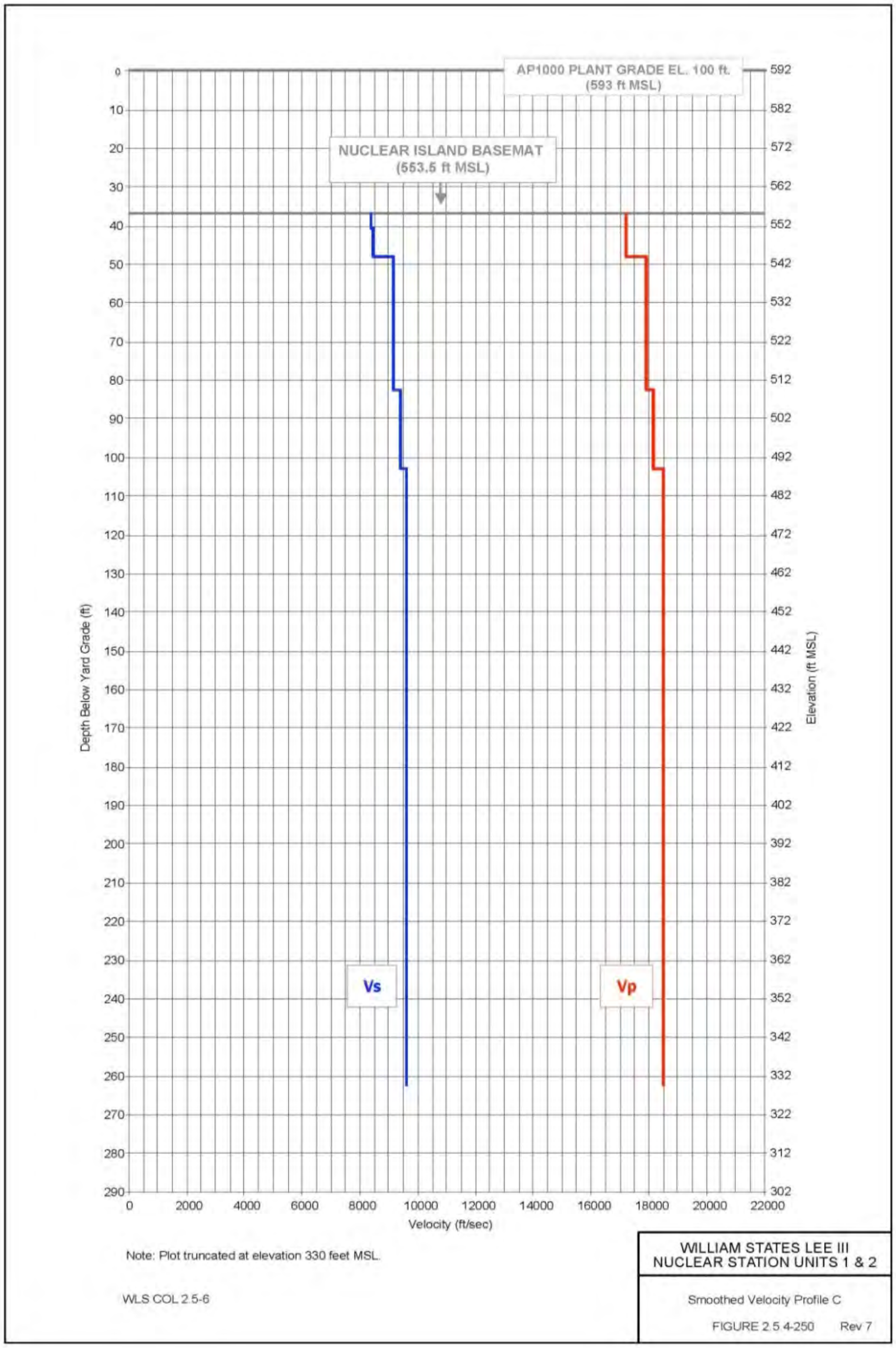
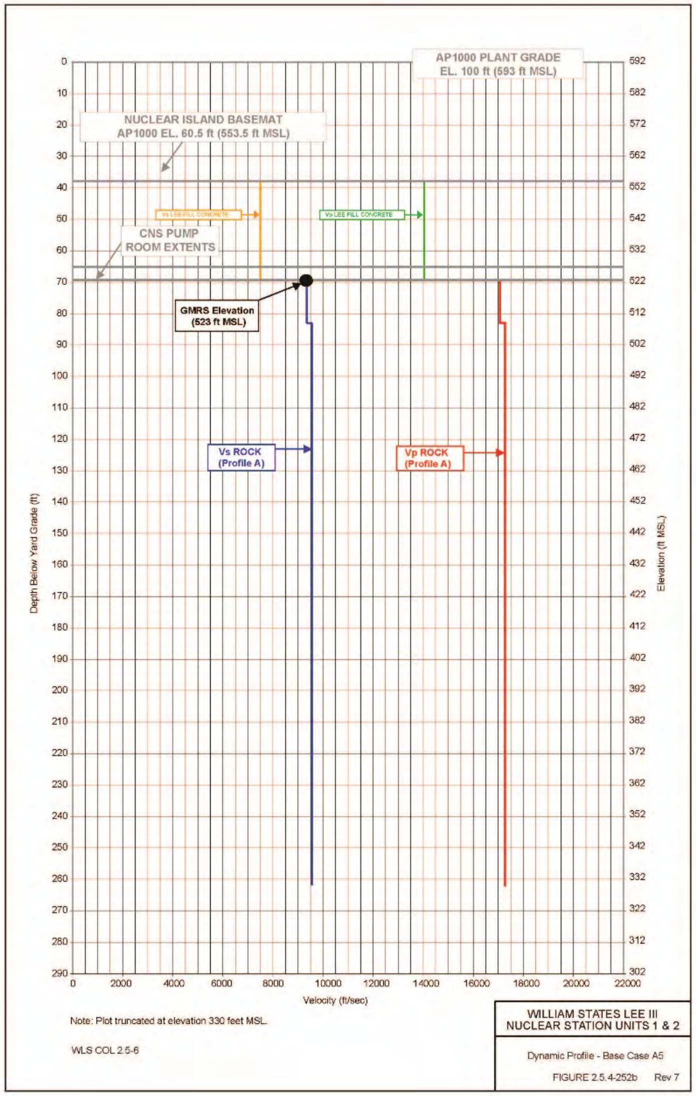


Figure 2.5.4-11 Lee Nuclear Station Unit 2 Centerline Dynamic Profile (WLS COL FSAR Revision 7, Figure 2.5.4-250)



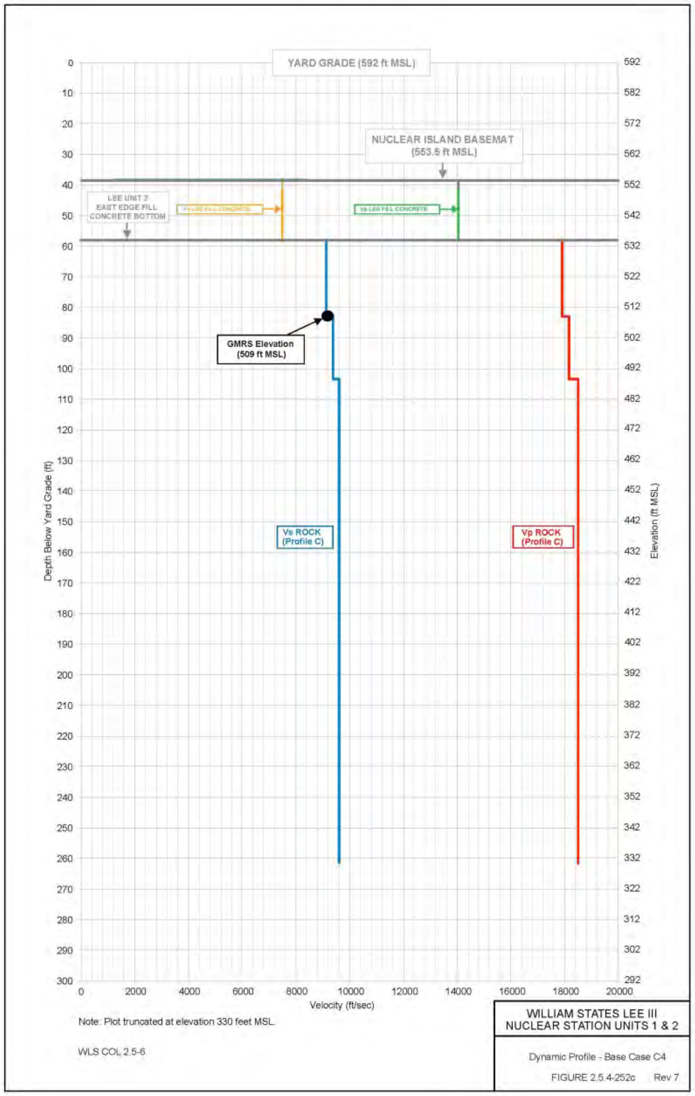
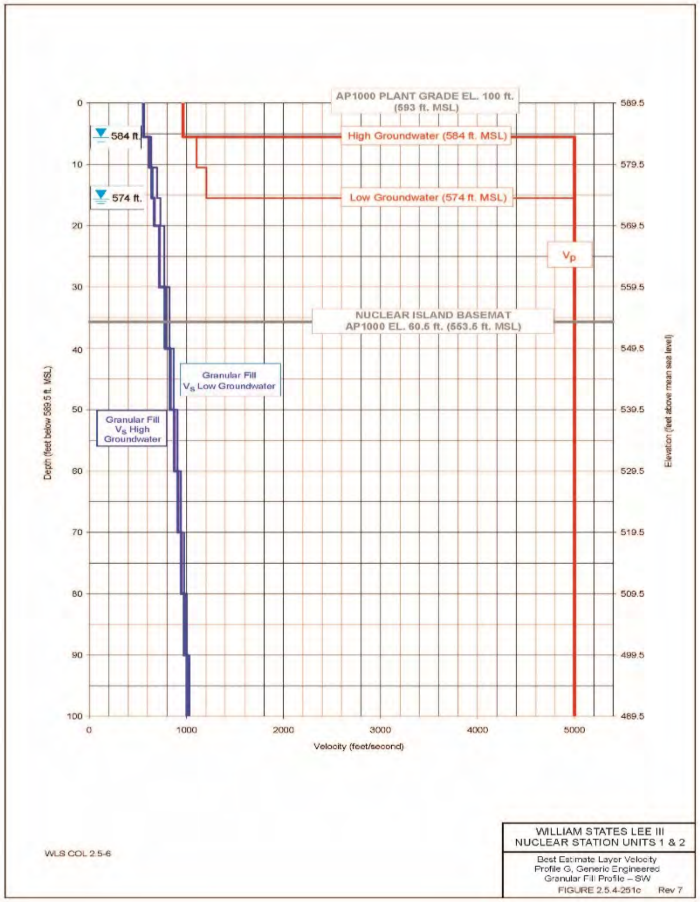


Figure 2.5.4-12 Lee Nuclear Station Unit 1 and 2 Dynamic Profiles Base Case A5 and C4 (WLS COL FSAR Revision 7, Figure 2.5.4-252b and 252c)



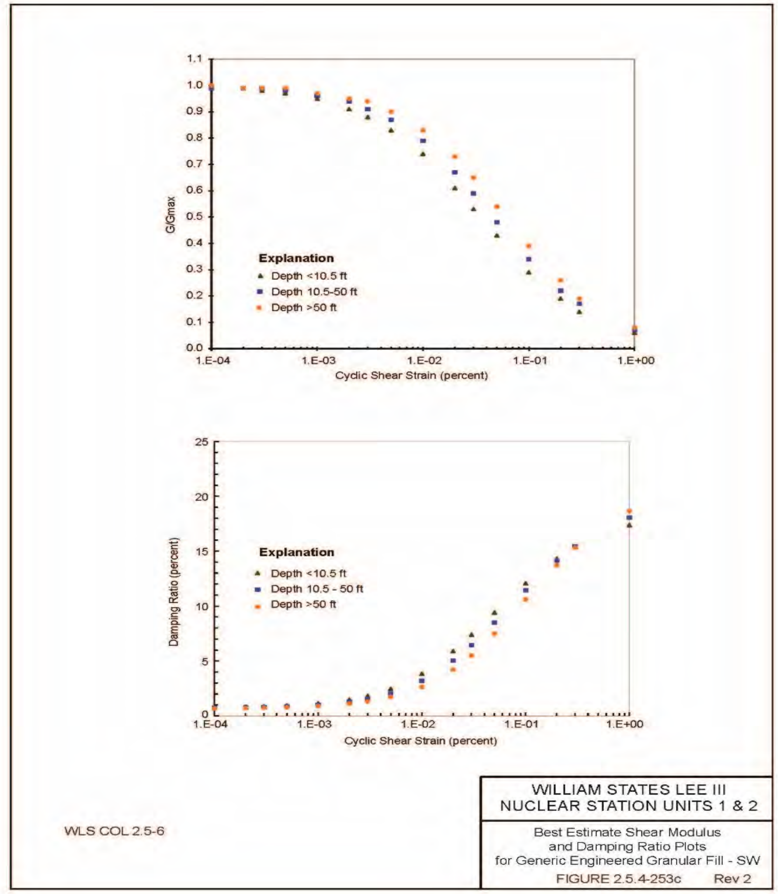


Figure 2.5.4-13 Design Properties of Generic Engineered Granular Fill – SW (WLS COL FSAR Revision 7, Figures 2.5.4-251c and 253c)

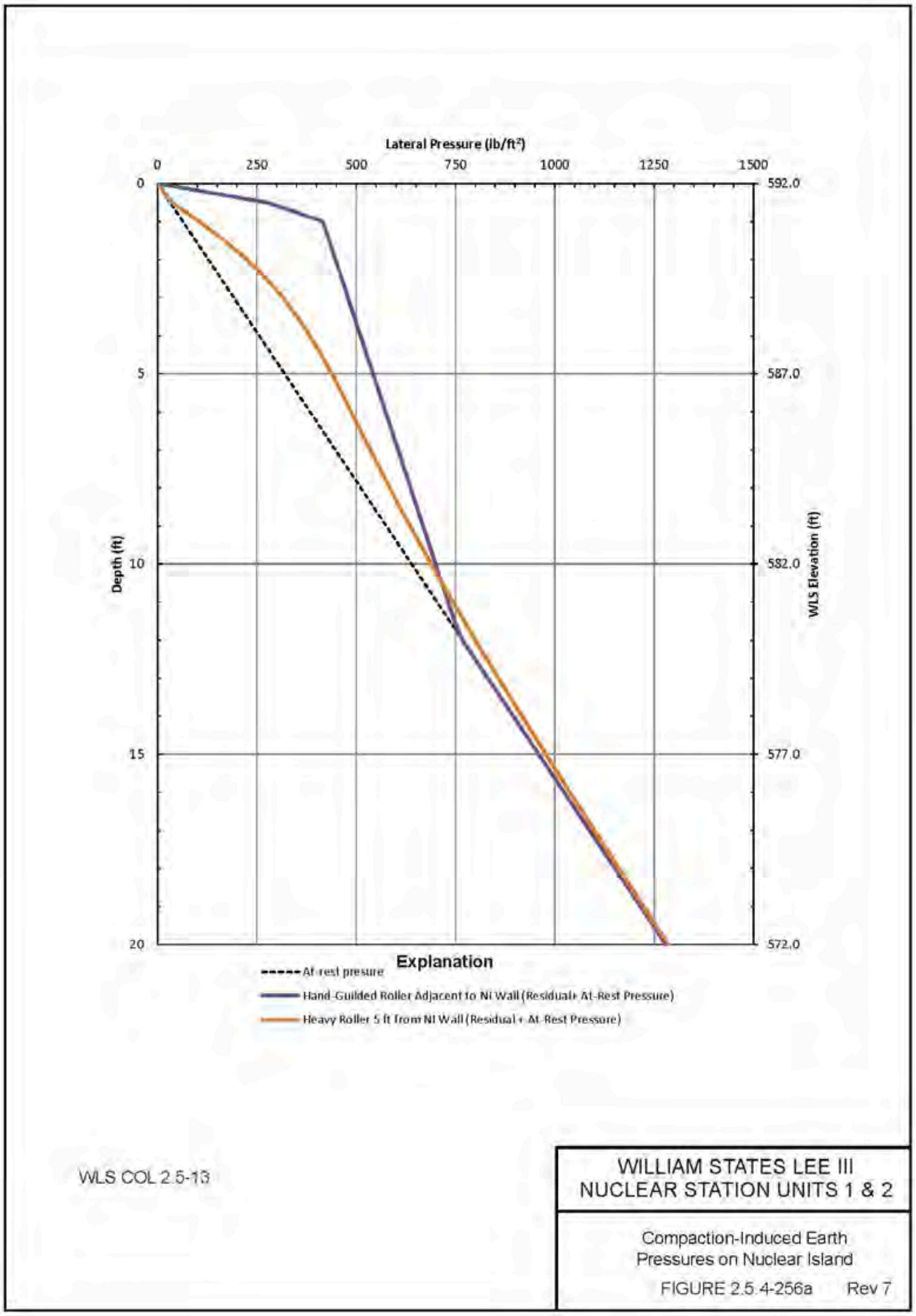


Figure 2.5.4-14 Compaction-Induced Earth Pressures on Nuclear Island (WLS COL FSAR Revision 8, Figure 2.5.4-256a)

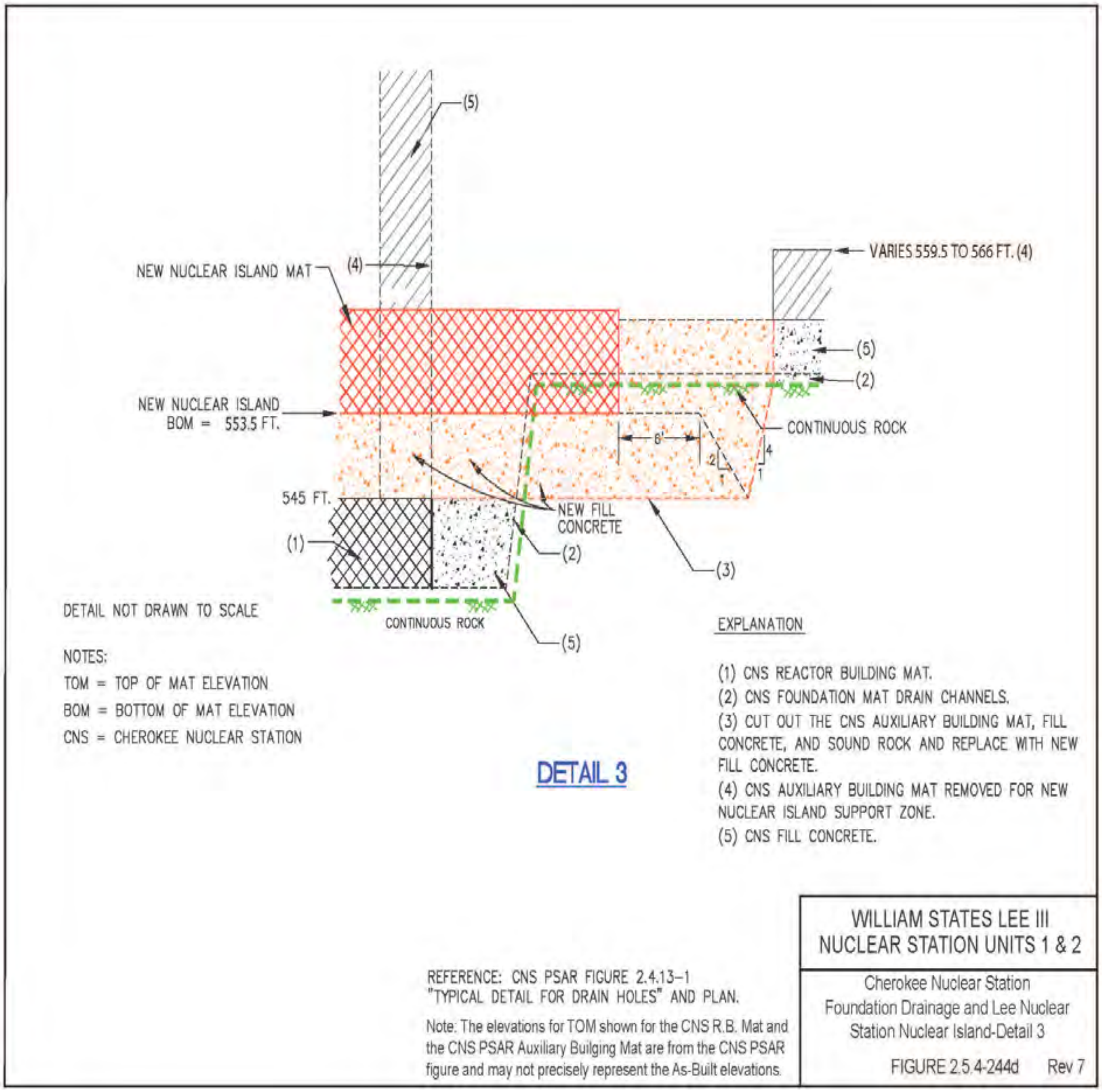


Figure 2.5.4-15 Typical Detail Showing Sealing of Existing Cherokee Foundation Drainage System (WLS COL FSAR Figure 2.5.4-244d)

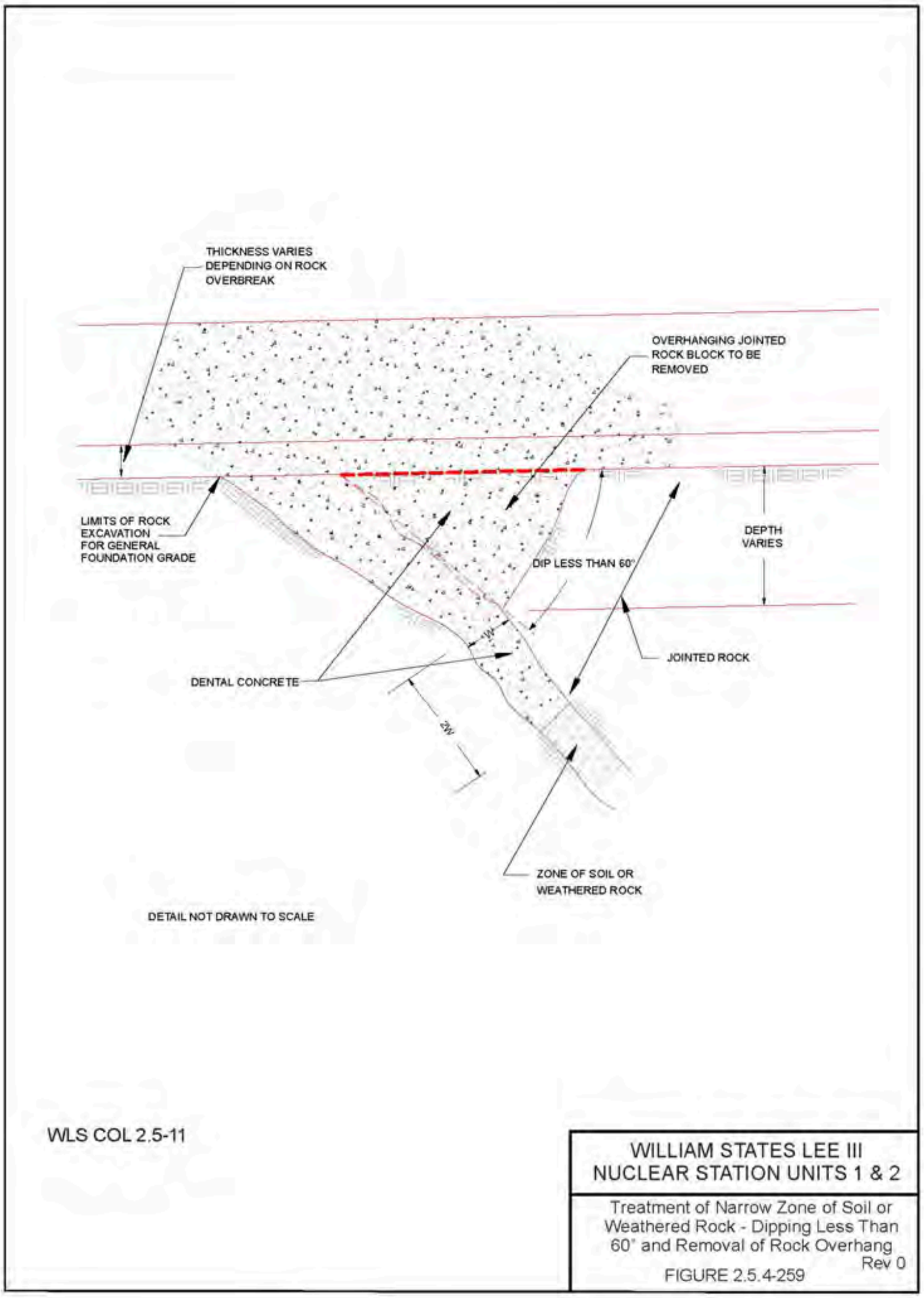


Figure 2.5.4-16 Typical Detail Showing Treatment of Weathered Zones Within Continuous Rock (WLS COL FSAR Figure 2.5.4-259)

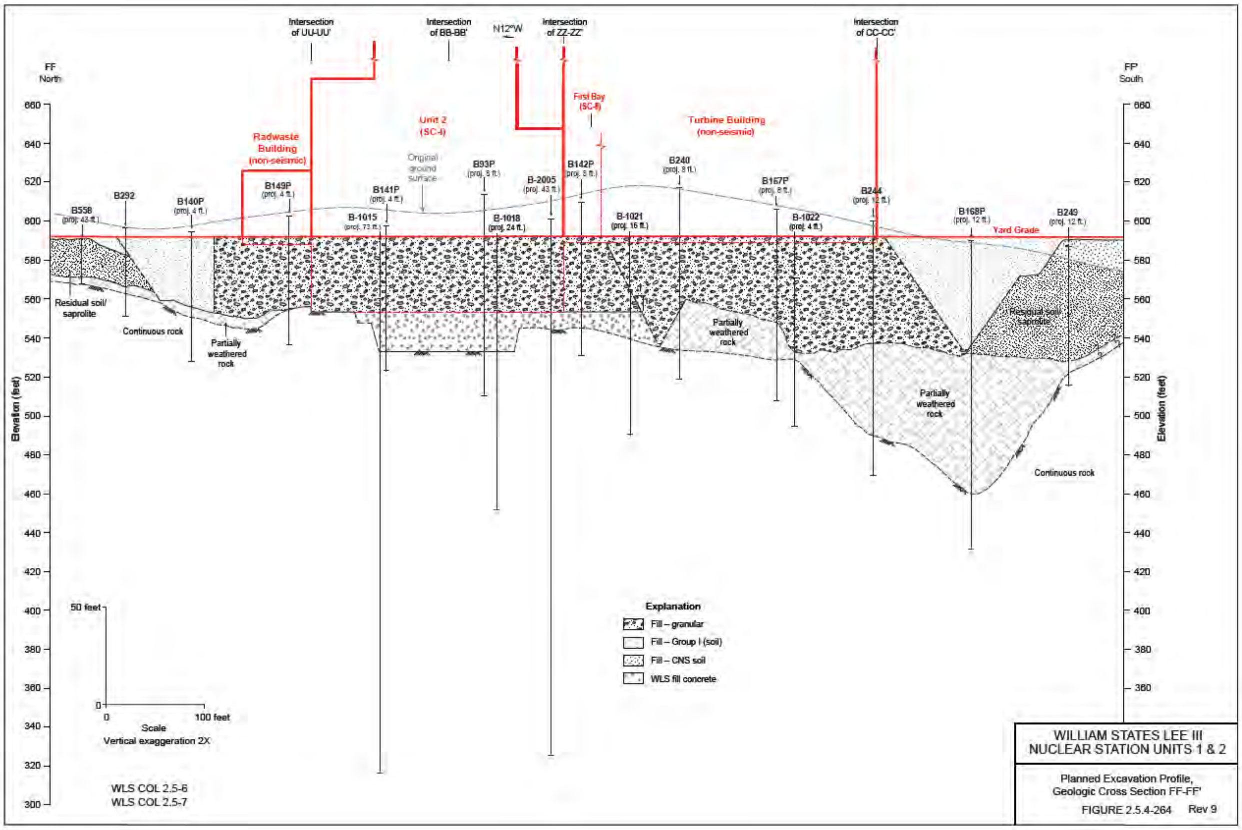


Figure 2.5.4-17 Depressed Areas underneath Lee Unit 2 Foundation That Require Further Excavation to Reach Continuous Rock (Based on WLS COL FSAR Revision 9, Figure 2.5.4-264)

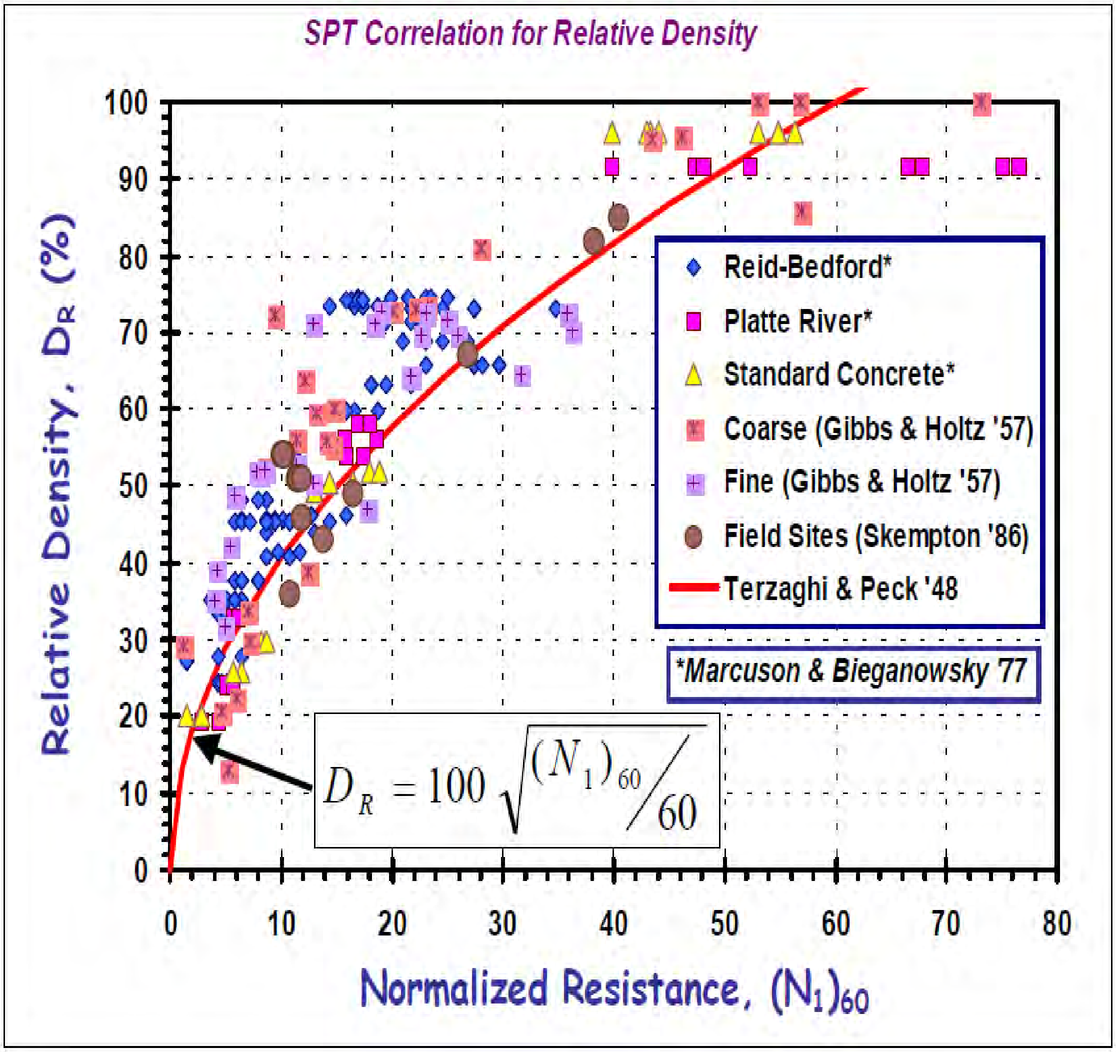


Figure 2.5.4-18 Relationship Between Relative Density and SPT (N1)60 (After Mayne)

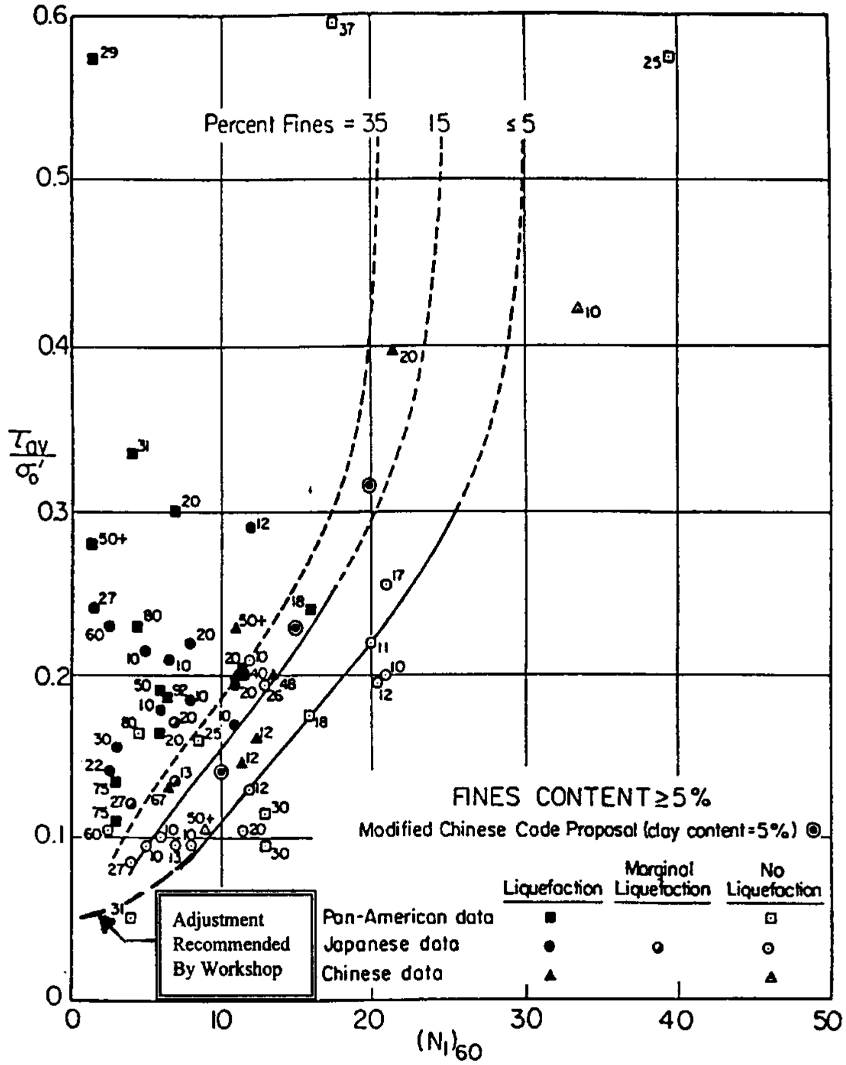


Figure 2.5.4-19 Liquefaction/No liquefaction of Magnitude 7.5 Earthquake of Soils With Various Percentages of Fines (After Youd, et al.)

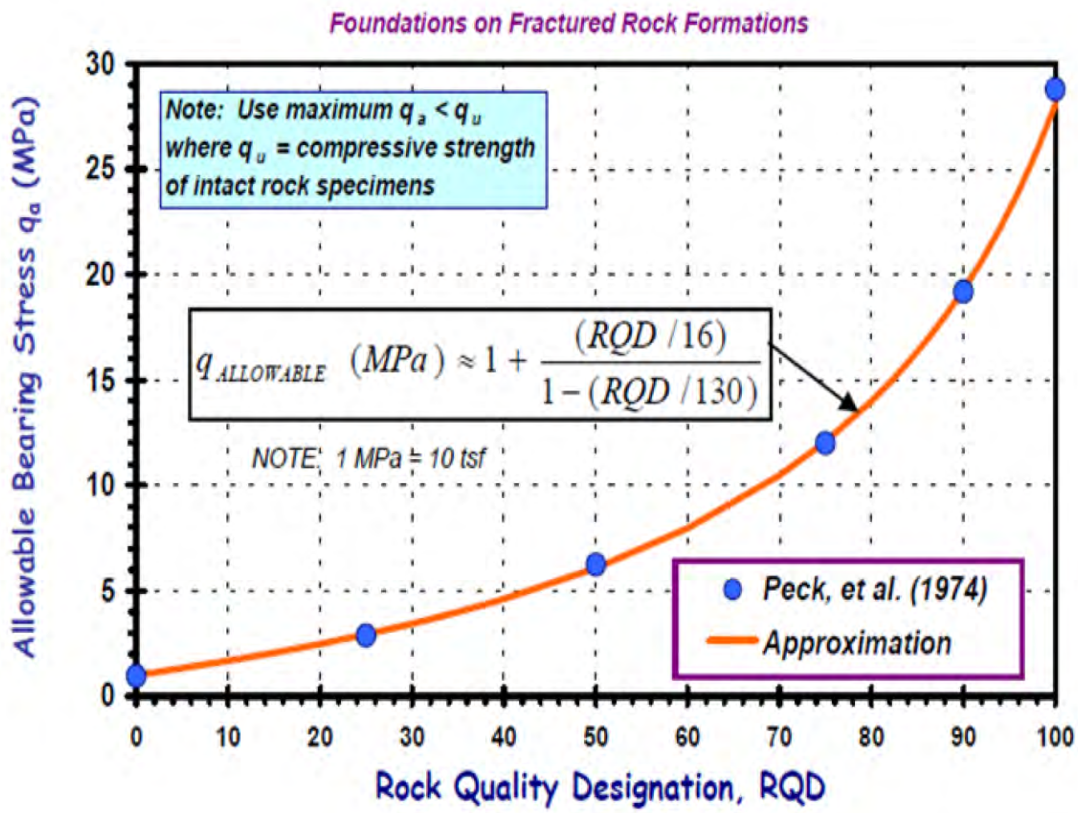


Figure 2.5.4-20 Allowable Bearing Stress on Fractured Rock From RQD (After Peck, et al., 1974)

2.5.5 Stability of Slopes

2.5.5.1 *Introduction and Overview*

Stability of slopes addresses the stability of all earth and rock slopes, both natural and man-made (cuts, fill, embankments, dams, etc.), whose failure, under any of the conditions to which they could be exposed during the life of the plant, could adversely affect the safety of the plant. The following subjects are evaluated using the applicant's data in the WLS COL FSAR and information available from other sources: (1) slope characteristics; (2) design criteria and design analyses; (3) results of the investigations including borings, shafts, pits, trenches, and laboratory tests; (4) properties of borrow material, compaction and excavation specifications; and (5) any additional information deemed necessary in accordance with 10 CFR Part 52.

Section 2.5.5 of this report addresses slope stability information related to the WLS site. Section 2.5.5.2 of this report provides a summary of relevant geologic and seismic information contained in WLS COL FSAR Section 2.5.5 of the WLS application. Section 2.5.5.3 of this report provides a summary of the regulations and guidance used by the applicant in its application and by the staff to review the application. Section 2.5.5.4 of this report provides a review of the staff's evaluation of the WLS COL FSAR Section 2.5.5. Section 2.5.5.5 of this report discusses any post combined license activities. Finally, Section 2.5.5.6 of this report provides an overall summary of the applicant's conclusions, as well as the staff's conclusions, restates any bases covered in the application, and confirms that the application meets the requirements defined in NRC regulations.

2.5.5.2 *Summary of Application*

WLS COL FSAR Section 2.5.5 addresses COL Information Items 2.5-14, “Stability of Slopes,” and 2.5-15, “Embankments and Dams,” of the AP1000 DCD, Revision 19.

In addition, in WLS COL FSAR Section 2.5.5, the applicant provided the following:

AP1000 COL Information Items

- WLS COL 2.5-14

The applicant provided additional information in WLS COL 2.5-14 to address COL Information Item 2.5-14, which addresses the provision of site-specific information about the static and dynamic stability of site-specific soil and rock slopes with regard to how their failure could adversely affect the safety of the Seismic Category I structures.

- WLS COL 2.5-15

The applicant provided additional information in WLS COL 2.5-15 to address COL Information Item 2.5-15, which addresses the provision of site-specific information about the static and dynamic stability of site-specific embankments and dams with regard to how their failure could adversely affect the safety of the nuclear power plant facilities.

The applicant developed WLS COL FSAR Section 2.5.5 for the evaluation of slope stability at the WLS site based on information derived from site investigations, geotechnical characterization studies, and excavation and backfill profiles presented in WLS COL FSAR Sections 2.5.4.1 through 2.5.4.5. These investigations and studies included consideration of geologic features and characteristics, site exploration involving soil and rock boring and sampling, groundwater monitoring, surface geophysical testing, in-situ testing, geotechnical test pits, geologic trench excavations, and laboratory testing; and geophysical surveys.

2.5.5.2.1 Slope Characteristics

WLS COL FSAR Section 2.5.5.1 describes the characteristics of existing permanent slopes, both natural and man-made, which exist within 0.4 km (0.25 mi) of the WLS Units 1 and 2 nuclear islands, including properties of the materials that make up these slopes. The applicant cross-referenced detailed information presented in WLS COL FSAR Sections 2.5.4.1 through 2.5.4.5 derived from investigations performed at the site. Based on the results of these investigations, the distance, height, and inclination of the slopes from the nuclear island, the applicant concluded that no slope, either permanent or man-made, exists for which failure will adversely affect the safety-related structures of WLS Units 1 and 2. The following sections provide a summary of the characteristics and material properties of slopes at the site.

2.5.5.2.1.1 *General Discussion*

In WLS COL FSAR Section 2.5.5.1.1, the applicant stated that the natural slopes at the WLS site existed through most or all of Holocene time (i.e., since 10,000 years ago), and the man-made slopes were part of the initial CNS construction. The applicant indicated that all these slopes exhibit acceptable stability with no visual evidence of groundwater seepage, past failure, incipient movement, or major creep. The applicant further stated that the native (i.e., residual and saprolitic) soils and engineered fill at the WLS site are not prone to liquefaction, and that the potential for a liquefaction-induced slope stability hazard does not exist for Seismic Category I structures under either static or dynamic loading conditions.

The applicant described the two permanent slopes nearest to the nuclear island areas, which are labeled as Slopes 5 and 7 in Figure 2.5.5-1 of this report. The applicant identified the highest slope (Slope 5 in Figure 2.5.5-1 of this report) as a natural hill located southwest of the WLS Unit 1 nuclear island. The applicant stated that this hill rises approximately 24.4 m (80 ft) above the yard grade, has a natural slope of approximately 2.5 horizontal to 1 vertical (2.5h:1v), and is located more than 305 m (1,000 ft) from WLS Unit 1.

Another slope (Slope 7 in Figure 2.5.5-1 of this report) identified by the applicant is an engineered slope about 366 m (1,200 ft) north of WLS Unit 2, which descends 26.8 m (55 ft) below the yard elevation. The applicant stated that this slope is approximately 2 horizontal to 1 vertical (2h:1v), and no credible mechanism exists whereby failure of a descending slope of about 26.8 m (55 ft) high and 366 m (1,200 ft) away could affect safety-related structures at the WLS site.

The applicant concluded that, due to the past stable history of the slopes, slope height and inclination, and its distance from the safety-related structures, the slopes at the WLS site do not pose a hazard to the safety of the plant.

2.5.5.2.1.2 *Exploration Program*

WLS COL FSAR Section 2.5.5.1.2 references information in WLS COL FSAR Sections 2.5.4.1 through 2.5.4.3, in which the applicant describes the assessment of the stability of permanent slopes at the WLS site.

2.5.5.2.1.3 *Groundwater and Seepage*

WLS COL FSAR Section 2.5.5.1.3 references the information provided in WLS COL FSAR Sections 2.4.12 and 2.5.4.6. The applicant concluded that groundwater seepage in natural and manmade slopes does not pose a hazard for the WLS site since no slopes are in close proximity to safety-related structures.

2.5.5.2.1.4 *Slope Materials and Properties*

In WLS COL FSAR Section 2.5.5.1.4, the applicant referenced WLS COL FSAR Section 2.5.4.2 for descriptions of the properties of the existing engineered fill and native residual and saprolitic soil that consist of all permanent slopes. The slope stability assessment performed by the applicant consisted of an evaluation of slope locations, geometries, inclinations, past stability, distance from the WLS Units 1 and 2 nuclear island structures, and observed long-term slope performance. The applicant used information on slope materials and properties as a guide regarding the distances between the slopes and safety-related structures. The applicant also noted that permanent slopes show inclinations of about 26.5 degrees (i.e., 2h:1v) or less, which is smaller than the friction angle of the materials comprising the slopes and, therefore, stated that the cohesive component of shear strength of the materials consisting of the slopes increased the perceived inherent stability of the slopes. The applicant further indicated, based on liquefaction potential of the materials that comprise the slopes as discussed in WLS COL FSAR Section 2.5.4.8, that the slope materials are not prone to liquefaction. Therefore, the applicant concluded that no potential slope stability hazard exists under either static or dynamic conditions to adversely affect safety-related structures at the WLS site.

2.5.5.2.2 Design Criteria and Analyses

WLS COL FSAR Section 2.5.5.2 indicates that the applicant limited the assessment of permanent slope conditions to those slopes within 0.4 km (0.25 mi) of the WLS Units 1 and 2 nuclear island structures and based this assessment on past slope performance, slope height and angle, and distance of the slope from safety-related structures. The applicant reported that the nearest permanent slopes are 305 m (1,000 ft) or more away from the WLS Units 1 and 2 nuclear island structures, and concluded that no permanent slopes required further analyses since none were identified in which failure would pose a hazard to the safety-related structures.

2.5.5.2.3 Logs of Borings

WLS COL FSAR Section 2.5.5.3 states that no borings, test pits, or trenches were used for stability analyses of permanent slope conditions surrounding the WLS safety-related nuclear island structures. The applicant indicated that this geotechnical information was not required since no slopes were determined to pose a hazard to the safety-related structures.

2.5.5.2.4 Compacted Fill

In WLS COL FSAR Section 2.5.5.4, the applicant stated that there are no safety-related permanent dams, dikes, or embankments at the WLS site. Therefore, the applicant concluded that design and performance criteria for compacted fills were not required.

2.5.5.3 *Regulatory Basis*

The applicable regulatory requirements for reviewing the applicant's discussion of stability of slopes are:

- 10 CFR 50.55a, "Codes and Standards," as it relates to the requirement that structures, systems, and components shall be designed, fabricated, erected, constructed, tested, and inspected in accordance with the requirement of applicable codes and standards commensurate with the importance of the safety function to be performed.

- 10 CFR Part 50, Appendix A, GDC 1, "Quality Standards and Records," as it relates to the requirement that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. This regulation also requires that appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

- 10 CFR Part 50, Appendix A, GDC 2, "Design Bases for Protection Against Natural Phenomena," as it relates to 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.

- 10 CFR Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," as it applies to the design of nuclear power plant structures, systems, and components important to safety to withstand the effects of earthquakes.

- 10 CFR Part 100, "Reactor Site Criteria," provides the criteria that guide the evaluation of the suitability of proposed sites for nuclear power and testing reactors.

- 10 CFR 100.23, "Geologic and Seismic Criteria," provides the nature of the investigations required to obtain the geologic and seismic data necessary to determine site suitability and identify geologic and seismic factors required to be taken into account in the siting and design of nuclear power plants

The related acceptance criteria are summarized from Standard Review Plan (SRP) Section 2.5.5:

- Slope Characteristics: To meet the requirements of 10 CFR Part 50 and 10 CFR Part 100, the discussion of slope characteristics is acceptable if the section includes: (1) cross sections and profiles of the slope in sufficient quantity and detail to represent the slope and foundation conditions; (2) a summary and description of static and dynamic properties of the soil and rock comprised by Seismic Category I embankment dams and their foundations, natural and cut slopes, and all soil or rock slopes whose stability would directly or indirectly affect safety-related and Seismic Category I facilities; and (3) a summary and description of groundwater, seepage, and high and low groundwater conditions.

- Design Criteria and Analyses: To meet the requirements of 10 CFR Part 50 and 10 CFR Part 100, the discussion of design criteria and analyses is acceptable if the criteria for the stability and design of all Seismic Category I slopes are described and valid static and dynamic analyses have been presented to demonstrate that there is an adequate margin of safety.

- Boring Logs: To meet the requirements of 10 CFR Part 50 and 10 CFR Part 100, the applicant should describe the borings and soil testing carried out for slope stability studies and dam and dike analyses.

- Compacted Fill: To meet the requirements of 10 CFR Part 50, the applicant should describe the excavation, backfill, and borrow material planned for any dams, dikes, and embankment slopes

In addition, the geologic characteristics should be consistent with appropriate sections from: RG 1.28, "Quality Assurance Program Requirements (Design and Construction)"; RG 1.132, "Site Investigations for Foundations of Nuclear Power Plants"; RG 1.138, "Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants"; RG 1.198, "Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites"; and RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

2.5.5.4 *Technical Evaluation*

The NRC staff reviewed Section 2.5.5 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of information presented in the FSAR and the DCD completely represents the required information related to the stability of slopes. The staff's review confirmed that information contained in the application or incorporated by reference addresses the information required for this review topic. NUREG-1793 and its supplements document the results of the staff's evaluation of the information incorporated by reference into the WLS COL application.

The staff reviewed the information in the WLS COL FSAR:

AP1000 COL Information Items

- WLS COL 2.5-14

The staff reviewed the resolution to the COL information items related to the stability of all earth and rock slopes, both natural and man-made (cuts, fill, embankments, dams, etc.), the failure of which, under any of the conditions to which they could be exposed during the life of the plant, could adversely affect the safety of the plant included under WLS COL FSAR Section 2.5.5.

With respect to COL Information Item WLS COL 2.5-14, the applicant stated that there are no soils or rock slopes that the failure of which could adversely affect the safety-related structures at the WLS site. The staff considered the results of site investigations in conjunction with the applicant's conclusion and concurs with the applicant's assessment that the slopes at the site are a sufficient distance from the WLS safety-related structures. The staff concludes that the applicant has met the criteria of COL Information Item 2.5-14 WLS COL 2.5-14.

- WLS COL 2.5-15

Regarding WLS COL 2.5-15, the applicant stated that there are no dams or embankments the failure of which could adversely affect the safety-related structures at the WLS site. The staff considered the results of site investigations, as well as the applicant's assertion and concurs with the applicant that there are no dams or embankments that might adversely affect safety-related structures of the WLS Units 1 and 2. The staff concludes that the applicant adequately addressed the criteria of COL Information Item 2.5-15 WLS COL 2.5-15.

2.5.5.5 *Post Combined License Activities*

There are no post COL activities associated with this section.

2.5.5.6 *Conclusions*

As discussed above, the applicant presented sufficient information on the evaluation of the stability of all earth and rock slopes, both natural and manmade at the proposed site, and adequately addressed COL Information Items WLS COL 2.5-14 and 2.5-15, which met the design criteria and requirements specified in the AP1000 DCD. The staff reviewed the investigations performed for slope stability studies and the evaluation of the stability of all slopes at the proposed site by the applicant, and concludes that the analyses and evaluations demonstrated that natural and manmade slopes will remain stable under the site-specific static and seismic loading conditions and that safety-related earthwork will function reliably at the site to justify the soil and rock characteristics used in the design. The staff further concludes that the design analyses contain adequate margins of safety for construction and operation of the nuclear power plant and meets the requirements of 10 CFR Part 50, Appendix A (GDC 1 and GDC 2); 10 CFR Part 50, Appendices B and S; and 10 CFR 100.23. Accordingly, the staff concludes that the WLS site is suitable with respect to the criteria governing the stability of slopes and, therefore, considers WLS COL FSAR Section 2.5.5 acceptable.

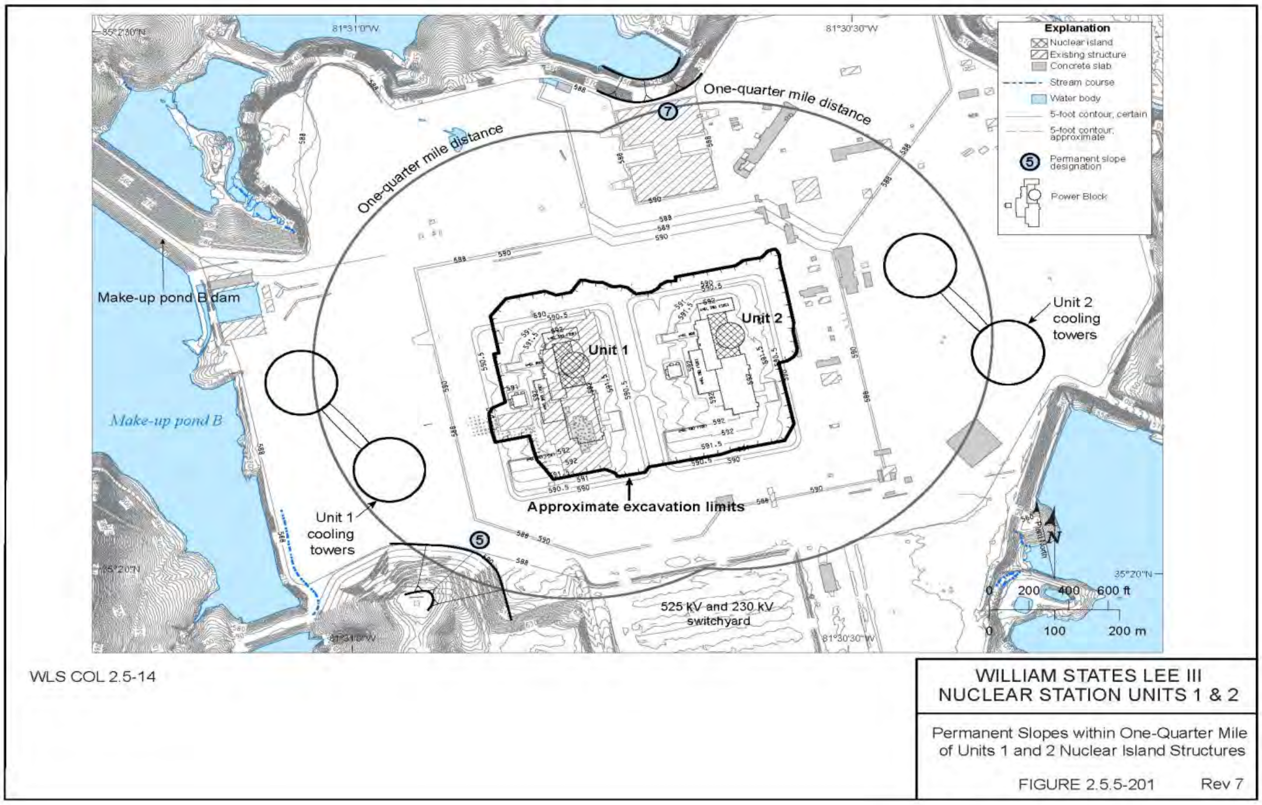


Figure 2.5.5-1 Permanent Slopes at the Lee Nuclear Station Site (WLS COL FSAR Figure 2.5.5-201)

2.5.6 Combined License Information

WLS COL FSAR Section 2.5.6 is an administrative departure from the AP1000 DCD for organization and numbering for the WLS COL FSAR sections. This section provides a list of all COL information items addressed in WLS COL FSAR Section 2.5. Since there is no technical content in this section, no safety evaluation was needed.

4.0 REACTOR

4.1 Introduction

This chapter describes the mechanical components of the AP1000 reactor and reactor core, including the reactor internals, control rod drive and core support structural materials, fuel system design (fuel rods and fuel assemblies), the nuclear design, and the thermal-hydraulic design. It also specifies the principal design criteria with which the mechanical design, the physical arrangement of the reactor components, and the capabilities of reactor control, protection, and emergency cooling systems (when applicable) must comply.

4.2 Summary of Application

Chapter 4 of the William States Lee III Nuclear Station (WLS) combined license (COL) Final Safety Analysis Report (FSAR), Revision 11, incorporates by reference Chapter 4 of the AP1000 Design Control Document (DCD), Revision 19.

In addition, in WLS COL FSAR Section 4.4, the applicant provided the following:

AP1000 COL Information Item

- STD COL 4.4-2

The applicant provided additional information in Standard (STD) COL 4.4-2 to address COL Information Item 4.4-2. This item states that, upon selection of the actual instrumentation, the instrumentation uncertainties of the operating parameters shall be calculated and the validity of the design-limit departure from nucleate boiling ratio (DNBR) values shall be confirmed.

License Condition

Part 10, License Condition 2, Item 4.4-2

The license condition will require the completion of the actions described in STD COL 4.4-2 prior to initial fuel load.

4.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793, “Final Safety Evaluation Report [FSER] Related to Certification of the AP1000 Standard Design.”

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the thermal-hydraulic design are identified in Section 4.4 of NUREG-0800, “Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants.”

To resolve the confirmatory item, the Nuclear Regulatory Commission (NRC) staff also used the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.72, “Immediate notification requirements for operating nuclear power reactors,” and 10 CFR 50.73, “Licensee event report system," and the guidance of NUREG-1022, "Event Reporting Guidelines: 10 CFR 50.72 and 50.73," Revision 2.

4.4 Technical Evaluation

The NRC staff reviewed Chapter 4 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the reactor internals, control rod drive and core support structural materials, fuel system design (fuel rods and fuel assemblies), the nuclear design, and the thermal-hydraulic design. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this safety evaluation report (SER) provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (Vogtle Electric Generating Plant [VEGP], Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from requests for additional information (RAIs).
- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.
- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) contains evaluation material from the SER for the Bellefonte Nuclear Plant (BLN), Units 3 and 4 COL application.

[1] See Section 1.2.2 for a discussion of the staff's review related to verification of the scope of information to be included in a COL application that references a design certification (DC).

The following portion of this technical evaluation section is reproduced from Section 4.4 of the VEGP SER:

<u>AP1000 COL Information Item</u>

- *STD COL 4.4-2*

The NRC staff reviewed STD COL 4.4-2 related to COL Information Item 4.4-2 and related COL Action Item 4.4-1 (from Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793)), included under Section 4.4 of the BLN COL FSAR, Revision 1. STD COL 4.4-2 states:

> *Following selection of the actual plant operating instrumentation and calculation of the instrumentation uncertainties of the operating plant parameters as discussed in DCD Subsection 7.1.6, the design limit DNBR values will be calculated. The calculations will be completed using the revised thermal design procedure (RTDP) with these instrumentation uncertainties and confirm that either the design limit DNBR values as described in DCD Section 4.4 remain valid or that the safety analysis minimum DNBR bounds the new design limit DNBR values plus DNBR penalties, such as rod bow penalty. This will be completed prior to fuel load.*

<u>License Condition</u>

Part 10, License Condition 2, Item 4.4-2

The applicant provided a license condition in Part 10 of the BLN COL application, "Proposed Combined License Conditions," which will require the completion of the actions described in STD COL 4.4-2 prior to initial fuel load.

As reported in FSER Section 4.4 related to the DCD, expected instrument uncertainties are included in the methodology used by the applicant in calculating the design limit DNBR values. The final validation of the design limit DNBR values will be based on the actual uncertainties for instrumentations not yet procured. The quantification of instrument uncertainties includes activities that require procurement and installation of the instruments, including evaluation of changes in sensor design and location, and that can only be completed after installation of the instruments. Confirmation of instrument uncertainties after completion of the installation does not alter the methods of evaluation used to establish setpoints in the technical specifications, since the design limit DNBR values were based on the plant specifications for instrumentation uncertainties. The design limit DNBR values are expected to remain valid through plant procurement.

The NRC staff concluded in FSER Section 4.4 that the methodology for calculating the design limit DNBR values complied with the relevant regulatory requirements. The staff further concluded that it was acceptable to complete the *final verification of the design limit DNBR values when the as-built specifications are available.*

Therefore, the staff concludes that the supplemental information described in FSAR Section 4.4 meets COL Information Item 4.4-2 described in AP1000 DCD Subsection 4.4.7.2, complies with COL Action Item 4.4-1, and is acceptable.

The staff also finds the applicant's proposed license condition that will require completing this analysis prior to fuel load acceptable, since the applicant has committed to confirm that either the design limit DNBR values remain valid, or that the safety analysis minimum DNBR bounds the new design DNBR values plus DNBR penalties, such as rod bow penalty.

Conformance to Regulatory Guide 1.133, Revision 1

In BLN COL FSAR Section 1.9, "Compliance with Regulatory Criteria," Section 1.9.1, "Regulatory Guides," the applicant adds Appendix 1AA, which provides an evaluation of the degree of compliance with Division 1 regulatory guides (RGs) as applicable to the content of this FSAR, or to the site-specific design, construction and/or operational aspects, and Table 1.9-201, which identifies the appropriate regulatory guide to FSAR cross-reference. In Appendix 1AA, the applicant provides an evaluation of its loose-part detection program for compliance with RG 1.133, Revision 1, May 1981, "Loose Part Detection Program for the Primary System of Light-Water-Cooled Reactors." It states that conformance of the design aspects is as stated in the DCD. It also documents conformance with the programmatic and/or operational aspects described in paragraphs C.3a and C.6 of RG 1.133, Revision 1.

RG 1.133, Revision 1, describes a method acceptable to the NRC staff for implementing regulatory requirements with respect to detecting a potentially safety-related loose part in light-water-cooled reactors during normal operation. The AP1000 design includes a digital metal impact monitoring system, which is a non-safety-related system provided for monitoring the reactor coolant system for metallic loose parts. AP1000 DCD Section 4.4.6.4 documents the conformance of this monitoring system to RG 1.133. BLN COL FSAR Appendix 1AA documents its conformance to the design aspects described in DCD Section 4.4.6.4, and also states it conforms to Regulatory Position C.3a, regarding manual mode of data acquisition for detection of loose parts and Regulatory Position C.6, regarding notification to NRC of confirmation of the presence of a loose part.

The NRC staff noted that RG 1.133, Revision 1, was not included in Revision 1 of FSAR Table 1.9-201 for a cross-reference to the appropriate FSAR section, although an evaluation of compliance with RG 1.133 is provided in Appendix 1AA. In response to Request for Additional Information (RAI) 1-7, the applicant added RG 1.133, Revision 1, to Table 1.9-201, as part of Revision 1 to the FSAR. In addition, the response to RAI 1-7 was supplemented by adding a conformance discussion for regulatory guide positions related to the procedures and training program (positions 4g, 4h, 4i and 4j) in the proposed revision to BLN

*FSAR Appendix 1AA, "A Conformance with Regulatory Guides." The proposed change to BLN FSAR is acceptable subject to a formal revision to BLN FSAR. Accordingly, this is **Confirmatory Item 4.4-1**. With the conformance of the programmatic and operational aspects of regulatory positions, the staff concludes that the applicant's loose parts detection program will conform to RG 1.133, Revision 1.*

Resolution of Standard Content Confirmatory Item 4.4-1

The staff notes that RAI 1-11 was mistakenly identified as RAI 1-7 in the standard content SER as it relates to the conformance discussion for RG 1.133. The RAI number related to conformance is 1-11. The staff also notes that the BLN SER did not address Position C.6 of RG 1.133.

Confirmatory Item 4.4-1, as modified by the discussion above, is related to the applicant's conformance with the RG 1.133 Positions C.4g, 4h, 4i, 4j, and 6 as documented in Appendix 1AA of the VEGP COL FSAR. The staff's review of the VEGP COL FSAR indicates that the VEGP COL FSAR Appendix 1AA was updated to include all the information identified in the Confirmatory Item 4.4-1 except for Position C.6.

The response to RAI 1-11 included a conformance discussion for RG 1.133, Position C.6, "Notification of a Loose Part." Position C.6 refers to RG 1.16, "Reporting of Operating Information." The applicant took an exception to this position because this RG had been withdrawn. The staff considered this justification to be inadequate. Although the staff agreed it was no longer relevant to refer to RG 1.16, there remained a need to address reporting requirements. In response to this staff concern, the applicant proposed a revision to Appendix 1AA of its FSAR. In a letter dated January 8, 2010, the applicant stated that it would follow reporting requirements in accordance with requirements of 10 CFR 50.72 and 10 CFR 50.73 using guidance of NUREG-1022. The staff considers the applicant's position adequately addresses reporting requirements for loose part notification and therefore considers the exception acceptable. The staff verified that the VEGP COL FSAR was appropriately revised. As a result, Confirmatory Item 4.4-1 is now closed.

4.5 Post Combined License Activities

For the reasons discussed in the technical evaluation section above, the staff finds the following license condition proposed by the applicant acceptable:

- License Condition (4-1) – Before initial fuel load, the licensee shall calculate the instrumentation uncertainties of the actual plant operating instrumentation to confirm that either the design limit DNBR values remain valid or that the safety analysis minimum DNBR bounds the new design limit DNBR values plus DNBR penalties, such as rod bow penalty.

4.6 Conclusion

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to the reactor internals, control rod drive and core support structural materials, fuel system design (fuel rods and fuel assemblies), the nuclear design, and the thermal-hydraulic design, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this chapter. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable. The staff based its conclusion on the following:

- STD COL 4.4-2 is acceptable because it specifies a commitment on the part of the applicant to confirm the validity of the calculations of the design limit DNBR values, which are based on the plant specifications for instrumentation uncertainties. The confirmation of plant instrument uncertainties will be completed when the as-built specifications are available. The methodology for this calculation was previously approved by the staff in NUREG-1793.

3 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.1 Conformance to General Design Criteria

Section 3.1 of the William States Lee III Nuclear Station (WLS) combined license (COL) Final Safety Analysis Report (FSAR), Revision 11, incorporates by reference, Section 3.1, "Conformance with NRC General Design Criteria," of Revision 19 of the AP1000 Design Control Document (DCD). In addition, in the WLS COL FSAR, the applicant provided the following:

Departure

- WLS DEP 6.4-1

The applicant provided additional information about WLS DEP 6.4-1 in Section 3.1.2 of the FSAR related to design changes affecting habitability of the main control room and changes to the calculated doses to control room operators. This information, as well as related WLS DEP 6.4-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.2 of this report.

The U.S. Nuclear Regulatory Commission (NRC) staff (the staff) reviewed the application and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this section.[1] The NRC staff's review confirmed that the applicant addressed the required information to satisfy the evaluation criteria. There is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements. Section 21.2 of this report evaluates the departure from the DCD provided in WLS DEP 6.4-1.

3.2 Classification of Structures, Components, Equipment, and Systems

3.2.1 Seismic Classification

3.2.1.1 *Introduction*

Nuclear power plant structures, systems, and components (SSCs) important to safety are to be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. Important to safety SSCs are defined in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic licensing of production and utilization facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," as those SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the

[1] See Section 1.2.2 for a discussion of the staff's review related to verification of the scope of information to be included in a COL application that references a design certification (DC).

public. Important to safety SSCs include safety-related SSCs that perform safety-related functions to ensure: (1) the integrity of the reactor coolant pressure boundary (RCPB); (2) the capability to shut down the reactor and maintain it in a safe-shutdown condition; and (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures. The earthquake for which these safety-related plant features are designed is defined as the safe shutdown earthquake (SSE). The SSE is based on an evaluation of the maximum earthquake potential for the site and is an earthquake that produces the maximum vibratory ground motion for which SSCs are designed to remain functional. In a nuclear plant, there may be equipment, considered to be non-safety- related that do not have safety functions, however, they may enhance the ability of the plant to withstand or recover from off-normal conditions. For this equipment that are non-safety-related but said to have a risk-significant function, the regulatory treatment of non-safety systems (RTNSS) process is applied to define seismic requirements for those SSCs.

The methodology in the referenced AP1000 DCD is incorporated by reference in the WLS COL application and classifies SSCs into three seismic categories: Seismic Category I, Seismic Category II and Non-seismic (NS). Those plant features designed to remain functional, if an SSE occurs, are designated Seismic Category I. Seismic Category I applies to both functionality and integrity of equipment for an SSE event, and Seismic Category II applies only to integrity. If the failure of a NS SSC during an SSE could result in the loss of function of safety-related items, then they are designated as Seismic Category II. This methodology is similar to Regulatory Guide (RG) 1.29, “Seismic Design Classification,” Revision 4, except that RG 1.29 does not use the terms Seismic Category II and NS.

3.2.1.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 3.2, incorporates by reference AP1000 DCD, Revision 19, Section 3.2. In addition, in WLS COL FSAR Section 3.2, the applicant provided the following:

Departures

- WLS DEP 3.2-1

The applicant provided additional information about WLS DEP 3.2-1 in Section 3.2 of the FSAR related to design modifications to the condensate return portion of the Passive Core Cooling System. This information, as well as related WLS DEP 3.2-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.1 of the SER.

Supplemental Information

- Standard (STD) Supplement (SUP) 3.2-1

The applicant provided supplemental information by adding text to the end of AP1000 DCD Section 3.2.1, “Seismic Classification,” which stated that there are no safety-related SSCs at WLS outside the scope of the AP1000 DCD. The applicant also stated that the non-safety-related SSCs outside the scope of the AP1000 DCD are classified as non-seismic (NS).

3.2.1.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. The acceptance criteria associated with the relevant requirements of NRC regulations for the seismic classification are given in NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 3.2.1.

The regulatory basis for acceptance of the supplemental information of defining the scope of safety-related SSCs is established in 10 CFR Part 50, Appendix A, General Design Criteria (GDC) 2, "Design Bases for Protection Against Natural Phenomena," which requires that all SSCs that are important to safety be designed to withstand the effects of natural phenomena, including earthquakes and guidance on how to meet this requirement is in RG 1.29.

3.2.1.4 *Technical Evaluation*

The staff reviewed Section 3.2 of the WLS COL application and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to seismic classification. The results of the staff's evaluation of the information incorporated by reference in the WLS COL application is documented in NUREG-1793 and its supplements.

Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform a technical review for each "standard issue" and use this review to evaluate subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the safety evaluation report (SER) for the Reference COL application (i.e., Vogtle Electric Generating Plant (VEGP) Units 3 and 4) were equally applicable to the WLS application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In the comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the WLS COL application, as applicable) resulting from requests for additional information (RAIs).

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed and that those changes were actualized in the WLS COL FSAR.

- The staff verified that the site-specific differences, if any, did not adversely affect any previous relevant evaluation or conclusion.

The staff completed its review and concluded that the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this report by use of italicized, double-indented formatting. Section 1.2.3 of this report provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) contains evaluation material from the SER for the Bellefonte Nuclear Plant (BLN), Units 3 and 4 COL application.

The staff reviewed the following information in the WLS COL FSAR:

Supplemental Information

- STD SUP 3.2-1

The staff reviewed STD SUP 3.2-1, related to the seismic classification of safety-related SSCs included under WLS COL FSAR Section 3.2.1, which states that there are no safety-related SSCs outside the scope of the DCD. Therefore, the seismic classification is acceptable.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 3.2.1.4 and concludes that the seismic classification is acceptable:

> *Important to Safety SSCs*
>
> *GDC 2 states, in part, that SSCs important to safety shall be designed to withstand the effects of earthquakes. BLN COL FSAR Section 3.2.1 states there are no safety-related SSCs outside the scope of the DCD. In request for additional information (RAI) 3.2.1-1, the applicant was requested to clarify if there is any site-specific non-safety-related SSCs outside the scope of the DCD that are important to safety and, if so, identify the appropriate seismic classification of such SSCs. The applicant's response identified that there are no site-specific non-safety-related SSCs outside the scope of the DCD that are important to safety and that non-safety-related SSCs outside the scope of the DCD are classified as non-seismic. In Revision 1 of the BLN COL FSAR, the applicant added the statement that the non-safety-related SSCs outside the scope of the DCD are classified as non-seismic. The revised BLN COL FSAR is acceptable, and the staff's concern is closed. The staff based its conclusion on the applicant's response that there are no site-specific non-safety-related SSCs outside the DCD that are important to safety.*
>
> *Seismic Classification of Other Site-Specific SSCs*
>
> *Section 1.8 of the AP1000 DCD, Revision 16 identified certain site-specific SSCs that are outside the scope of the AP1000 standard plant, such as the circulating water system (CWS) and its heat sink, for which the COL applicant must provide site-specific information. The seismic classification of the CWS is not identified in DCD Table 3.2-3. Section 1.8 of BLN COL FSAR identifies certain COL items that represent interfaces for the standard design, but the seismic classification is not identified for the CWS.*
>
> *In RAI 3.2.1-2, the applicant was requested to clarify if there are any site-specific SSCs outside the scope of the DCD that are not included in DCD Tables 3.2-2 and 3.2-3 that are to be seismically classified in the COL. For example, site-specific structures, the CWS and miscellaneous items such as reactor vessel insulation are not included in the tables. If so, the applicant was requested to identify the appropriate seismic classification of such SSCs. This concern was also identified in an RAI for the review of AP1000 Revision 16 and the DC applicant clarified that the seismic categorization of CWS and reactor vessel insulation are not plant-specific and are to be classified in the DCD. Therefore, this concern is closed and seismic classification of these components is to be addressed in the DCD rather than the BLN COL FSAR.*

<u>Quality Assurance for Seismic Category II SSCs</u>

It is not clear in the BLN COL FSAR how Title 10 of the Code of Federal Regulations (CFR) 50, Appendix B is applied to seismic Category II SSCs, including those that may be site-specific. DCD Appendix 1A identifies that AP1000 conforms to RG 1.29, Regulatory Position C.4 and Section 1.8 identifies COL Information Item 17.5-1 for quality assurance (QA) in the design phase. DCD Section 17.5.2 identifies that the COL applicant will address its QA program and that the QA program will include provisions for seismic Category II SSCs. In RAI 3.2.1-4, the applicant was requested to clarify the extent that pertinent QA requirements of Appendix B to 10 CFR Part 50 in Regulatory Position C.4 of RG 1.29 apply to those activities affecting the safety-related functions of those portions of SSCs covered under Regulatory Positions 2 and 3 of RG 1.29, including any site-specific SSCs. If this issue will be resolved in the DCD rather than the COL for all plant SSCs, including those that are site-specific, the applicant was requested to advise the NRC staff that this was the case. The RAI response identified that there are no site-specific seismic Category II SSCs and that the application of 10 CFR Part 50, Appendix B is addressed by the DCD. Since there are no site-specific seismic Category II SSCs, this COL concern is closed for the BLN COL FSAR.

<u>Consistency with RG 1.29, Revision 4</u>

Section 3.2.1 of the BLN COL FSAR does not identify any departures relative to seismic classification identified in the DCD and BLN COL FSAR, Appendix 1AA identifies conformance with RG 1.29, Revision 3 as stated in the DCD rather than Revision 4 of RG 1.29, dated March 2007. In RAI 3.2.1-3, the applicant was requested to clarify if seismic classifications of site-specific SSCs are consistent with RG 1.29, Revision 4. The RAI response identified that seismic classification of site-specific SSCs not addressed in the DCD is consistent with RG 1.29, Revision 4. This position is acceptable to the staff, since it represents the current RG revision. The applicant revised Appendix 1AA in Revision 1 of the BLN COL FSAR to indicate conformance to RG 1.29, Revision 4.

3.2.1.5 *Post Combined License Activities*

There are no post COL activities related to this section.

3.2.1.6 *Conclusion*

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to seismic classification, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements. The staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the requirements of 10 CFR Part 50, Appendix A, and GDC 2. The staff based its conclusion on the following:

- WLS DEP 3.2-1, related to design modifications to the condensate return portion of the Passive Core Cooling System, is reviewed and found acceptable by the staff in Section 21.1 of this SER.

- STD SUP 3.2-1 is acceptable because the WLS COL FSAR states that there are no safety-related SSCs outside the scope of the AP1000 DCD. The WLS COL FSAR also states that the non-safety-related SSCs outside the scope of the DCD are classified as NS. Therefore, the requirements of 10 CFR Part 50, Appendix A, GDC 2, the acceptance criteria in NUREG-0800, Section 3.2.1, and the guidelines in RG 1.29 are satisfied.

3.2.2 Classification Systems

3.2.2.1 *Introduction*

The system and component quality group classification addresses, in part, the general design criterion that nuclear power plant SSCs that are important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Important to safety SSCs are defined in 10 CFR Part 50, Appendix A as those SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. Important to safety SSCs include safety-related SSCs that perform one of the following safety-related functions to ensure: (1) the integrity of the RCPB; (2) the capability to shut down the reactor and maintain it in a safe-shutdown condition; and (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures. The RTNSS process is applied to define supplemental quality requirements for SSCs that are non-safety-related but perform risk significant functions.

The system and component quality group classification in combination with the RTNSS process define appropriate classifications, codes and standards and special treatment for important to safety pressure-retaining components and their supports, depending on their safety function. RG 1.26, “Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” Revision 4, provides the regulatory guidance for classifying SSCs important to safety and the appropriate quality standards.

3.2.2.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 3.2, incorporates by reference AP1000 DCD, Revision 19, Section 3.2. In addition, in WLS COL FSAR Section 3.2, the applicant provided the following:

Supplemental Information

- STD SUP 3.2-1

The applicant provided supplemental information by adding text to the end of AP1000 DCD Section 3.2.2, “AP1000 Classification System,” and stated that there are no safety-related SSCs at WLS outside the scope of the AP1000 DCD.

3.2.2.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. In addition, the acceptance criteria associated with the relevant requirements of NRC regulations for the system quality group classification are given in NUREG-0800, Section 3.2.2.

The basis for acceptance is established in RG 1.26, "Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Revision 4, and applicable American Society of Mechanical Engineers (ASME) Codes and industry standards. RG 1.26 provides regulatory guidance for classifying SSCs important to safety and applying the appropriate quality standards. Conformance to the guidance contained in RG 1.26 is one way to ensure that component quality will be commensurate with the importance of the safety functions of these systems. Thus, this constitutes the basis for satisfying GDC 1, "Quality Standards and Records" for pressure-retaining components and their supports.

3.2.2.4 *Technical Evaluation*

The staff reviewed Section 3.2 of the WLS COL application and checked the referenced DCD to ensure that the combination of the DCD and the WLS COL application represents the complete scope of information relating to this review topic. The staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the system quality group classification. The results of the staff's evaluation of the information incorporated by reference in the WLS application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform a technical review for each "standard issue" and use this review to evaluate subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application FSAR (i.e., VEGP Units 3 and 4) were equally applicable to the WLS application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In the comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the application, as applicable) resulting from responses to RAIs.
- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed and that those changes were actualized in the COL FSAR.
- The staff verified that the site-specific differences, if any, did not adversely affect any previous relevant evaluation or conclusion.

The staff completed its review and concluded that the evaluation performed for the standard content is directly applicable to the WLS COL application. This standard content material is identified in this report by use of italicized, double-indented formatting. Section 1.2.3 of this report provides an explanation of why the standard content material from the SER for the Reference COL application (VEGP) contains evaluation material from the SER for the BLN Units 3 and 4 application.

The staff reviewed the following information in the WLS COL FSAR:

Supplemental Information

- STD SUP 3.2-1

The staff reviewed STD SUP 3.2-1 related to the seismic classification of safety-related SSCs included under WLS COL FSAR Section 3.2.2, which states that there are no safety-related SSCs outside the scope of the AP1000 DCD at WLS.

The staff reviewed STD SUP 3.2-1 related to quality group classification of systems included under WLS COL FSAR Section 3.2.2. The staff notes that STD SUP 3.2-1 is identical to STD SUP 3.2-1 in the BLN FSAR with respect to quality group classification of systems included under WLS COL FSAR Section 3.2.2. The staff noted that additional information was needed to evaluate BLN STD SUP 3.2-1 resulting in RAIs being issued to the BLN COL applicant. The WLS COL applicant endorsed the BLN RAI response in a February 5, 2009, letter. As such, the staff's review of STD SUP 3.2-1 is addressed through the comparison with the BLN SER. As discussed below, there are no site-specific non-safety-related SSCs outside the scope of the AP1000 DCD that are important to safety, so there are no changes to the quality group classifications listed in WLS COL FSAR Section 3.2.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 3.2.2.4:

> *Special Treatment for Risk-Significant SSCs*
>
> *GDC 1 identifies, 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 supplemented or modified as necessary to assure a quality product in keeping with the required safety function. Supplemental quality standards and QA programs applicable to passive SSCs used in non-safety-related regulatory treatment of non-safety systems that may be important to safety are not clearly defined in the BLN COL FSAR for site-specific SSCs.*
>
> *In RAI 3.2.2-2, the applicant was requested to clarify what supplemental quality standards are applied to non-safety-related site-specific SSCs that are important to safety to ensure that all SSCs important to safety are designed, fabricated, erected, and tested to quality standards commensurate with the safety function to be performed. Any site-specific SSCs that are considered important to safety may also require special treatment, but the response to RAI 3.2.1-1 identified that there are no site-specific non-safety-related SSCs outside the scope of the DCD that are important to safety. Therefore, this concern is closed.*
>
> *Codes and Standards*
>
> *The Staff Requirements Memorandum (SRM), dated July 21, 1993, concerning SECY-93-087 identified that the staff will review passive plant design applications using the newest codes and standards endorsed by the NRC and unapproved revisions to the codes will be reviewed on a case by case basis. Editions of*

various codes and standards referenced in DCD Section 3.2.6 are not current and newer codes and standards are not referenced in BLN COL FSAR Sections 3.2 or 1.8. In RAI 3.2.2-3, the applicant was requested to clarify if any different or current codes and standards are applied to the design and procurement of site-specific SSCs, other than those identified in the DCD. The RAI response identified that the applicant intends to implement the DCD identified codes and standards and that the codes and standards applied to the design and procurement of non-safety-related site-specific SSCs are those identified in various sections of the BLN COL FSAR. Although codes and standards for site-specific SSCs would be expected to be identified and reviewed in the COL application rather than the DCD, the response to RAI 3.2.1-1 identified that there are no site-specific non-safety-related SSCs outside the scope of the DCD that are important to safety. Therefore, this concern is closed.

Consistency with RG 1.26, Revision 4

Section 3.2.2 of the BLN COL FSAR does not identify any departures relative to quality group classification identified in the DCD and BLN COL FSAR, Appendix 1AA identifies conformance with RG 1.26, Revision 3 in the DCD rather than Revision 4, dated March 2007. In RAI 3.2.2-1, the applicant was requested to clarify if quality group classifications of site-specific SSCs are consistent with RG 1.26, Revision 4. The applicant's response clarified that the quality group classification of site-specific SSCs is consistent with RG 1.26, Revision 4. This position is acceptable to the staff, since it represents the current RG revision. This staff concern is closed and the BLN COL FSAR Appendix 1AA has been revised accordingly to reflect this RAI response.

3.2.2.5 *Post Combined License Activities*

There are no post COL activities related to this section.

3.2.2.6 *Conclusion*

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to the system quality group classification, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements. The staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the requirements of 10 CFR Part 50, Appendix A, GDC 1. The staff based its conclusion on the following:

- STD SUP 3.2-1 is acceptable with regard to quality group classifications because no change was made to the quality group classifications in AP1000 DCD Section 3.2 and there are no site-specific non-safety-related SSCs outside the scope of the AP1000 DCD that are important to safety. Therefore, the staff finds that the requirements of 10 CFR Part 50, Appendix A, GDC 1, the acceptance criteria in NUREG-0800, Section 3.2.2, and the guidelines in RG 1.26 are satisfied.

3.3 Wind and Tornado Loadings

Seismic Category I and II buildings and structures are designed to withstand extreme wind and tornado loading conditions in compliance with the requirements in GDC 2 in Appendix A to 10 CFR Part 50 that states SSCs that are important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures shall reflect the appropriate consideration of the most severe of the natural phenomena that have been historically reported in the area of the plant, with sufficient margin to account for limited accuracy, quantity, and period of time for collection of data. In this section of the report, the staff reviewed the Seismic Category I and II structures subjected to wind and tornado loadings. Other natural phenomena effects, such as earthquakes, floods, tsunami, and seiches, are evaluated in Sections 3.4, 3.7 and 3.8 of this report.

3.3.1 Wind Loadings

3.3.1.1 *Introduction*

Seismic Category I structures must withstand the effects of the specified design wind speed for the plant to ensure conformance with 10 CFR Part 50, Appendix A, GDC 2. The specific areas of review are the design wind speed, its recurrence interval, speed variation with height, and applicable dust factors from the standpoint of use in defining the input parameters for the appropriate structural design criteria for wind loading. The staff also reviews the procedures that are used to transform the design wind speed into an equivalent pressure applied to structures taking into consideration the geometrical configuration and physical characteristics of the structures and the distribution of wind pressure on the structures.

3.3.1.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 3.3, incorporates by reference AP1000 DCD, Revision 19. In WLS COL FSAR Section 3.3.1, the applicant provided the following:

<u>AP1000 COL Information Items</u>

- WLS COL 3.3-1

The applicant provided information in WLS COL FSAR Section 3.3.1.1, “Design Wind Velocity,” to address COL Information Item 3.3-1 stating that the wind velocity characteristics for the WLS site are given in WLS COL FSAR Section 2.3.1.2.8. The applicant stated that these values are bounded by the design wind velocities specified in AP1000 DCD Section 3.3.1.1 for the standard AP1000 plant design. In addition, the applicant stated, in part, that the effects of wind on the safety-related SSCs due to failures in an adjacent AP1000 plant are bounded by the evaluation of the buildings and structures in a single unit. The portion of WLS COL 3.3-1 relating to tornado loadings is reviewed in Section 3.3.2 of this report.

3.3.1.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. The acceptance criteria associated with the relevant requirements of NRC regulations for wind loadings are given in NUREG-0800, Section 3.3.1.

The regulatory basis for WLS COL 3.3-1 is 10 CFR Part 50, Appendix A, GDC 2, that states SSCs that are important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornados, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions and related regulatory guidance in RG 1.76, “Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants,” Revision 1.

3.3.1.4 *Technical Evaluation*

The staff reviewed WLS COL Section 3.3 and checked the referenced DCD to ensure that the combination of the DCD and the WLS COL application represents the complete scope of information relating to this review topic.[1] The staff’s review confirmed that the information in the application and incorporated by reference addresses the required information relating to wind loadings. The results of the staff’s evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements. The staff reviewed the information in the WLS COL FSAR:

AP1000 COL Information Item

- WLS COL 3.3-1

The staff reviewed WLS COL 3.3-1 related to design wind loads applied on safety-related SSCs included under WLS COL FSAR Section 3.3.1.1. The application states in WLS COL 3.3-1 that the wind velocity characteristics for WLS are given in WLS COL FSAR Section 2.3.1.2.8. The COL states that these values are bounded by the DCD design wind velocity values for the standard AP1000 plant.

In Section 2.3.1.4 of this report, the staff concluded that a site characteristic 3-second gust basic wind speed value of 43 meters per second (m/s) (96 miles per hour (mph)) is an acceptable wind speed for this site. Since this value is bounded by the AP1000 design wind speed of 62.8 m/s (145 mph), the staff concludes that the design wind velocities for the WLS site comply with GDC 2 and are acceptable; therefore, WLS COL 3.3-1 is resolved.

3.3.1.5 *Post Combined License Activities*

There are no post COL activities related to this section.

3.3.1.6 *Conclusion*

The staff reviewed the application and checked the referenced DCD. The staff’s review confirmed that the applicant addressed the required information relating to wind loadings, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements. The staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the requirements of GDC 2.

- WLS COL 3.3-1, as it relates to design wind loads, is acceptable based on the site-specific wind velocities, reviewed in Section 2.3 of this report, being bounded by the AP1000 DCD design wind velocities and, therefore, complies with GDC 2.

3.3.2 Tornado Loading

3.3.2.1 *Introduction*

Tornado loadings are considered for design in accordance with AP1000 DCD Section 3.3.2, "Tornado Loadings." AP1000 DCD Section 3.3.2 addresses tornado loadings for Seismic Category I structures using applicable tornado design parameters to determine forces on structures as explained in AP1000 DCD Section 3.3.1.2. Also AP1000 DCD Section 3.3.2.1 states that the estimated probability of tornado wind speeds to be greater than the design basis tornado is between $10E^{-6}$ and $10E^{-7}$ per year at a "worst location" anywhere within the contiguous United States. The WLS COL FSAR discusses and supplements the AP1000 DCD in Sections 3.3.2.1, "Applicable Design Parameters," and 3.3.2.3, "Effect of Failure of Structures or Components Not Designed for Tornado Loads."

3.3.2.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 3.3, incorporates by reference AP1000 DCD, Revision 19. In WLS COL FSAR Section 3.3.2, the applicant provided the following:

AP1000 COL Information Items

- WLS COL 3.3-1

The applicant provided information in WLS COL FSAR to resolve COL Information Item 3.3-1. In WLS COL 3.3-1, the applicant stated in WLS COL FSAR Section 3.3.2.1 that tornado characteristics for WLS, given in WLS COL FSAR Section 2.3.1.2.2, are bounded by the tornado design parameters given in AP1000 DCD Section 3.3.2.1 for the standard AP1000 plant. In addition, the applicant stated that the effects of wind and tornado on the safety-related SSCs due to failures in an adjacent AP1000 plant are bounded by the evaluation of the buildings and structures in a single unit. The portion of WLS COL 3.3-1 relating to design wind velocity characteristics is reviewed in Section 3.3.1 of this report.

- STD COL 3.3-1

The information provided in WLS COL FSAR Section 3.3.2.3 to address STD COL 3.3-1 states that the effects of wind and tornado on the safety-related SSCs due to failures in an adjacent AP1000 plant are bounded by the evaluation of the buildings and structures in a single plant.

3.3.2.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. The acceptance criteria associated with the relevant requirements of NRC regulations for tornado loading are given in NUREG-0800, Section 3.3.2. Acceptance of WLS COL 3.3-1 is established based on site-specific parameters and verification of bounding conditions for relevant parameters related to the AP1000 DCD interface criteria for tornado, site arrangement, and building construction. The design of AP1000 safety-related SSCs for tornado loads must meet the requirements of 10 CFR Part 50, Appendix A, GDC 2 that states SSCs important to safety shall be designed to withstand the effects of natural phenomena, such as earthquakes, tornados, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.

3.3.2.4 *Technical Evaluation*

The staff reviewed WLS COL FSAR Section 3.3.2 and checked the referenced DCD to ensure that the combination of the DCD and the WLS COL application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to tornado loading. The results of the staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements. The staff reviewed the information in the WLS COL FSAR:

AP1000 COL Information Item

- WLS COL 3.3-1

The staff reviewed WLS COL 3.3-1 included under WLS COL FSAR Sections 3.3.2 and 3.5.1. Specific information provided by the applicant includes development of site-specific parameters and verification of bounding conditions, site arrangement and building construction. In WLS COL 3.3-1, the applicant stated that the tornado characteristics for WLS, given in WLS COL FSAR Section 2.3.1.2.2, are bounded by the tornado design parameters given in AP1000 DCD Section 3.3.2.1 for the standard AP1000 plant design. In addition, the applicant stated that the effects of wind and tornado on the safety-related SSCs due to failures in an adjacent AP1000 plant are bounded by the evaluation of the buildings and structures in a single unit. In Section 2.3.1 of this report, the staff concluded that tornado site characteristics chosen by the applicant were acceptable. Since these values match the design tornado site characteristics included in the AP1000 DCD, the staff concludes that the design tornado site characteristics for the WLS site comply with GDC 2.

The scope of WLS COL 3.3-1 also includes the effects of wind and tornado on the safety-related SSCs due to failure of non-safety-related buildings in an adjacent AP1000 plant and WLS. The applicant stated that these effects are bounded by the evaluation of the buildings and structures in a single unit. To assure the failure of structures or components not designed for wind or tornado loadings does not affect the capability of safety-related SSCs to perform their intended safety functions, the COL applicants had the following three options in AP1000 DCD Section 3.3.2.3.

1. design the adjacent non-safety-related structure to the design basis tornado loading

2. analyze the effect of failure of adjacent non-safety-related structures on nuclear island (NI) structures to ensure that no impairment of safety function will result

3. design a structural barrier to protect Seismic Category I SSCs from adjacent structural collapse

The applicant used Option 2 for WLS COL 3.3-1, indicating that the effects of wind and tornado on the safety-related SSCs due to failure of an adjacent non-safety-related building are bounded by the evaluation of the structures in a single unit at WLS. The analysis of the impact of building collapse on the nuclear island (NI) structures is in AP1000 DCD Section 3.7.2.8 and the staff's review of this analysis is provided in NUREG-1793 and its supplements. Based on the above discussion, the staff finds WLS COL 3.3-1 acceptable and resolved.

- STD COL 3.3-1

COL standard information item STD COL 3.3-1 addresses the effect of failure of SSCs not designed for tornado loadings in WLS COL FSAR Section 3.3.2.3. COL standard information item STD COL 3.3-1 involves consideration of a tornado-initiated failure of site-specific structures and components whose failure could compromise the safety of AP1000 safety-related structures and components at WLS site.

The staff reviewed the resolution to the COL Information Item STD COL 3.3-1 relating to the effect of failure of SSCs not designed for tornado loadings included under WLS COL FSAR Section 3.3.2.3. To ensure the failure of structures or components not designed for wind or tornado loadings does not affect the capability of safety-related SSCs to perform their intended safety functions, the COL applicants were offered three options in AP1000 DCD Section 3.3.2.3:

1. design the adjacent non-safety-related structure to the design basis tornado loading

2. analyze the effect of failure of adjacent non-safety-related structures on nuclear island (NI) structures to assure that no impairment of safety function will result

3. design a structural barrier to protect Seismic Category I SSCs from adjacent structural collapse

In STD COL 3.3-1, the applicant used Option 2. Since the applicant has not placed any additional structures adjacent to the standard unit, the original analysis done in AP1000 DCD Section 3.7.2.8 is still valid. The staff's review of this analysis is provided in NUREG-1793 and its supplements. Based on the above discussion, the staff consider STD COL 3.3-1 resolved.

3.3.2.5 *Post Combined License Activities*

There are no post COL activities related to this section.

3.3.2.6 *Conclusion*

The staff reviewed the application and checked the referenced DCD. The staff’s review confirmed that the applicant addressed the required information relating to tornado loading, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements. The staff concludes that the relevant information presented in the WLS COL FSAR Section 3.3.2 is acceptable and meets the requirements of 10 CFR Part 50, Appendix A, GDC 2.

- WLS COL 3.3-1, as it relates to design tornado loads, is acceptable based on the design tornado site characteristics, reviewed in Section 2.3 of this report, matching the AP1000 DCD design tornado site characteristics and, therefore, complying with GDC 2. WLS COL 3.3-1, as it relates to the effects of wind and tornado on the safety-related SSCs due to failure of non-safety-related buildings in an adjacent AP1000 plant and WLS is acceptable because the applicant incorporated by reference acceptable methodology from AP1000 DCD Section 3.7.2.8.

- STD COL 3.3-1, as it relates to the effects of wind and tornado on the safety-related SSCs due to failure of non-safety-related buildings in an adjacent AP1000 plant is acceptable because the applicant incorporated by reference an acceptable assessment methodology from AP1000 DCD Section 3.7.2.8.

3.4 Water Level (Flood) Design

3.4.1 Flood Protection

3.4.1.1 *Introduction*

Seismic Category I SSCs have flood protection measures for both external flooding and postulated internal flooding from plant component failures.

3.4.1.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 3.4 incorporates by reference AP1000 DCD, Revision 19, Section 3.4. In WLS COL FSAR Section 3.4, the applicant provided the following:

AP1000 COL Information Item

- WLS COL 3.4-1

The applicant provided information to resolve COL Information Item 3.4-1, which addresses plant-specific information on site-specific flooding hazards protective measures. That information, for WLS COL 3.4-1, in WLS COL FSAR Section 3.4.1.3, “Permanent Dewatering System,” and WLS COL FSAR Section 3.4.3, “Combined License Information,” states that no permanent dewatering system is required because site groundwater levels are 0.6 meters (m) (2 feet (ft) or more below site grade level and the site-specific water levels satisfy the interface requirements identified in AP1000 DCD Section 2.4, respectively.

3.4.1.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. The regulatory acceptance criteria associated with the relevant requirements of NRC regulations for flood protection measures are given in NUREG-0800, Section 3.4.1. The acceptance criteria associated with the relevant requirements of NRC regulations for the identification of floods and flood design considerations are given in NUREG-0800, Section 2.4.12.

3.4.1.4 *Technical Evaluation*

The staff reviewed this application and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The staff’s review confirmed that the information in the application and incorporated by reference addresses the required information relating to flood protection measures. The results of the staff’s evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements. The staff reviewed the following information in the WLS COL FSAR:

AP1000 COL Information Item

- WLS COL 3.4-1

The staff reviewed WLS COL 3.4-1 addressing the permanent dewatering system and site-specific water levels in WLS COL FSAR Sections 3.4.1.3 and 3.4.3, respectively. This site-specific COL item states that the COL applicant will demonstrate that the site satisfies the interface requirements as described in Section 2.4. If these criteria cannot be satisfied because of site-specific flooding hazards, the applicant may propose protective measures as discussed in Section 2.4. In WLS COL FSAR Section 3.4, the applicant provided the following plant-specific information:

- WLS COL FSAR Section 3.4.1.3, “Permanent Dewatering System,” states that no permanent dewatering system is required because site groundwater levels are 0.6 m (2 ft) or more below site grade level as described in WLS COL FSAR Section 2.4.12.5.

- WLS COL FSAR Section 3.4.3, “Combined License Information,” states that the site-specific water levels given in WLS COL FSAR Section 2.4 satisfy the interface requirements identified in AP1000 DCD Section 2.4.

In Section 2.4.12 of this report, the staff accepted the WLS applicant's position that no permanent dewatering system is required and that the site-specific groundwater characteristics for the WLS site fall within the Tier 1 and Tier 2 AP1000 DCD parameter values. Therefore, the staff concludes that the site-specific information in WLS COL 3.4-1 is acceptable.

3.4.1.5 *Post Combined License Activities*

There are no post COL activities related to this section.

3.4.1.6 *Conclusion*

The staff reviewed the application and checked the referenced DCD. The staff’s review confirmed that the applicant addressed the required information relating to flood protection measures, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the regulatory guidance in NUREG 0800, Sections 2.4.12 and 3.4.1. :

- WLS COL 3.4-1, is acceptable based the staff’s conclusions in NUREG-1793 regarding the need for a permanent dewatering system and on the staff’s conclusions in Section 2.4.12 of this report that no permanent dewatering system is required and that the site-specific groundwater characteristics for the WLS site fall within the Tier 1 and Tier 2 AP1000 DCD groundwater parameter values.

3.4.2 Analytical and Test Procedures

Analysis methods and test procedures are described for the design of AP1000 standard plants to assess the maximum water levels due to internal flooding caused by equipment failure or external flooding caused by natural phenomena and make sure that they do not jeopardize the safety of the plant or the ability to achieve and maintain safe shutdown conditions. WLS COL FSAR, Revision 11, Section 3.4 incorporates by reference, with no departures or supplements, AP1000 DCD, Revision 19, Section 3.4.2, “Analytical and Test Procedures.” AP1000 DCD Section 3.4.2 states that the analytical approach for external and internal flooding events is described in AP1000 DCD Section 3.4.1.2, “Evaluation of Flooding Events.” The staff reviewed the application and checked the referenced AP1000 DCD to ensure that no issue relating to this section remained for review.[1] The staff’s review confirmed that there is no outstanding issue related to this section. The results of the staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

3.5 Missile Protection

Seismic Category I structures are analyzed and designed to be protected from a wide spectrum of missiles (e.g., missiles from rotating and pressurized equipment, gravitational missiles, and missiles generated from tornado winds). When a missile hazard is identified, its statistical significance is determined (a missile is significant when it could cause unacceptable consequences or violate the guidelines of 10 CFR Part 100, “Reactor Site Criteria”).

3.5.1 Missile Selection and Description

3.5.1.1 *Introduction*

The design credits safety-related structures, systems, and components to establish and maintain safe shutdown conditions following a postulated event such as internally generated missiles. The SSCs needed to bring the plant to safe shutdown, including the main control room and the recirculating service water system are located inside the auxiliary building and containment shield building, respectively. Both the auxiliary and containment shield buildings are Seismic Category I NI structures having thick structural concrete walls that provide internal and external missile protection. The missiles generated outside containment by rotating or pressurized (high-energy fluid system) equipment are included. Aircraft hazards and missiles generated by human activities offsite as well as those generated by weather are also considered. SSCs considered “important to safety” are protected against internally generated missiles (outside containment), in accordance with NUREG-0800, Section 3.5.1.1.

3.5.1.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 3.5, “Missile Protection,” incorporates by reference AP1000 DCD, Revision 19, with site-specific information and supplements. WLS COL FSAR Section 3.5.1, “Missile Selection and Description,” which includes Sections 3.5.1.1, “Internally Generated Missiles (Outside Containment)”; 3.5.1.2, “Internally Generated Missiles (Inside Containment)”; 3.5.1.3, “Turbine Missiles”; 3.5.1.4, “Missiles Generated by Natural Phenomena”; 3.5.1.5,” Missiles Generated by Events Near the Site”; and 3.5.1.6, “Aircraft Hazards.” In WLS COL FSAR Section 3.5, the applicant provided the following:

AP1000 COL Information Item

- WLS COL 3.5-1

The applicant provided information in the WLS COL FSAR to resolve WLS COL 3.5-1. WLS COL FSAR Section 3.5.1.5, "Missiles Generated by Events Near the Site," states that certain buildings, such as the gate house, administrative building water service building, et al., at the WLS site are common structures located at a nuclear power plant. Therefore, any missiles resulting from a tornado-initiated failure of those common structures are not more energetic than tornado missiles postulated for design of the AP1000. Furthermore, the WLS COL FSAR states that the missiles generated by events near the site are evaluated in accordance with WLS COL FSAR Section 2.2.3. With regard to WLS COL 2.5-1, WLS COL FSAR Section 3.5.1.6, "Aircraft Hazards," states that the approach and methodology from NUREG-0800, Section 3.5.1.6 has been used to calculate the probability of an aircraft crash into areas of safety-related structures. Descriptions of Airports and Airways are addressed in WLS COL FSAR Section 2.2.2.7 and aircraft hazards are evaluated in WLS COL FSAR Section 3.5.1.6.

Supplemental Information

- STD SUP 3.5-1

The applicant provided information in WLS COL FSAR Section 3.5.1.3, "Turbine Missiles," to address STD SUP 3.5-1. This supplemental information states that the potential for a turbine missile from another AP1000 plant in close proximity has been considered for WLS in accordance with RG 1.115, "Protection Against Low-Trajectory Turbine Missiles," Revision 1. The WLS COL FSAR also states that in addition to low potential for turbine missile strike, the design of safety-related structures such as the shield building provide additional protection for safety-related SSCs from this event.

- STD SUP 3.5-2

The applicant provided supplemental information WLS COL FSAR Section 3.5.1.3, "Turbine Missiles," by stating that the turbine system maintenance and inspection program is discussed in WLS COL FSAR Section 10.2.3.6.

3.5.1.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," September 2004, and its supplements. The acceptance criteria associated with the relevant requirements of NRC regulations for missile selection and description are given in NUREG-0800, Sections 3.5.1.1 through 3.5.1.6.

The design of safety-related structures for protection against missiles using acceptable procedures must meet the requirements of 10 CFR Part 50, Appendix A, GDC 2.
10 CFR 100.21(e), "Non-seismic site criteria," provides regulatory requirements for potential hazards associated with nearby transportation routes, industrial and military facilities. The regulatory basis for acceptance of WLS COL 3.5-1 is that the applicant developed sufficient site-specific parameters and verification of bounding conditions compared to the AP1000 DCD interface criteria for missile generation, site arrangement, and building construction. Additional regulatory guidance related to the review in this report is contained in RG 1.76 on design-basis

tornado and tornado missiles, RG 1.91 on evaluation of explosions postulated to occur on transportation routes, RG 1.115, Sections C.1 and C.3, on protection against low trajectory turbine missiles, and RG 1.117, Regulatory Positions C.1 through C.3 on tornado design classification.

3.5.1.4 *Technical Evaluation*

The staff reviewed this application and checked the referenced DCD to ensure that the combination of the DCD and the WLS COL application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the WLS COL application and incorporated by reference addresses the required information relating to missile protection of safety-related SSCs. The staff's evaluation of the information incorporated by reference in the WLS COL application is documented in NUREG-1793 and its supplements. Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform a technical review for each "standard issue" and use this review to evaluate subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the Reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS application, the staff undertook the following reviews.

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In the comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting responses to from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed and that those changes were actualized in the COL FSAR.

- The staff verified that the site-specific differences, if any, did not adversely affect any previous relevant evaluation or conclusions.

The staff completed its review and concluded that the evaluation performed for the standard content is directly applicable to the WLS COL application. This standard content material is identified in this report by use of italicized, double-indented formatting. Section 1.2.3 of this report provides an explanation of why the standard content material from the SER for the Reference COL application (VEGP) contains evaluation material from the SER for the BLN Units 3 and 4 COL application. The staff reviewed the information in the WLS COL FSAR:

AP1000 COL Information Items

- WLS COL 3.5-1

The staff reviewed information related to missiles generated near the site included under WLS COL FSAR Section 3.5.1.5 related to missiles generated by events near the site included under WLS COL FSAR Section 3.5.1.5. The applicant provided site-specific information to resolve the COL information items stating that the effects of explosions have been evaluated and it has been determined that the over pressure criteria of RG 1.91 is not exceeded. Since the staff did not identify any over pressure criteria, no further evaluation of postulated missiles is required as the effect of postulated missiles will be less than those associated with the over-pressure levels considered in RG 1.91.

WLS COL FSAR Section 3.5.1.6, “Aircraft Hazards,” states that based on the description of nearby aircraft handling facilities and air routes in WLS COL FSAR Section 2.2.2.7, and the methodology outlined in NUREG-0800, “Standard Review Plan,” Section 3.5.1.6, “Aircraft Hazards,” the applicant concludes that it can be qualitatively shown that the total aircraft crash hazard probability of for WLS is much lower than was conservatively calculated to be 1.8×10^{-7}/year. Therefore, the applicant concludes the aircraft hazards pose no undue risk to the health and safety of the public.

The applicant evaluated potential aircraft hazards and effects on safety-related structures following the approach and methodology outline in NUREG-0800 Section 3.5.1.6, “Aircraft Hazards.” and determined the effects of an aircraft crash on safety-related structures in the site. The probability of aircraft accidents resulting in radiological consequences that may exceed the 10 CFR Part 100 radiological dose requirements was evaluated by the applicant based on its one Federal airway that passes within 6.4 km (4 mi) of the plant. Low altitude Airway V54 runs between Spartanburg Downtown Memorial Airport (SPA) (located 42 km (26.1 mi) from WLS) and Charlotte/Douglas International Airport (CLT). CLT is located 55 km (34.4 mi) from WLS. Information was provided regarding the number of flights using this airway V54. No airports having flights more than $500D^2$ per year located within 16 km (10 mi) of WLS. There are no military training routes within 16 km (10 mi) of the site.

The staff performed independent confirmatory probability calculations using the most conservative total highest annual flight data within 9 km (5 mi) of the plant obtained from the Federal Aviation Administration (FAA) covering the 5-year period from 2004-2008 and applying to Airway V54. Based on using this FAA annual flight data for the year 2007 conservatively, the staff also concluded the total aircraft accident probability of about 1.8×10^{-7} per year, which is less than the acceptance criteria of 10^{-6} per year in NUREG-0800, Section 3.5.1.6.

On the basis of the confirmatory analysis and the review of the applicant’s assumptions and data used for the estimation of aircraft accident probability, the staff concludes that the operation of the WLS in the vicinity of the CLT does not present an undue risk to the health and safety of the public and meets the relevant requirements of 10 CFR Part 100 and 10 CFR 100.10 (or 10 CFR 100.20, as appropriate) and meets the acceptance criteria provided in NUREG-0800, Sections 3.5.1.5 and 3.5.1.6. This conclusion is based on information provided in the WLS COL FSAR and the staff’s independent verification of the applicant’s assessment of aircraft hazards at the site that resulted in a probability on the order of magnitude of 10^{-7} per year for an accident having radiological consequences worse than the exposure guidelines of 10 CFR Part 100.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 3.5.1.4:

Supplemental Information

- *STD SUP 3.5-1*

The NRC staff reviewed the standard supplementary information (STD SUP 3.5-1) on the probability of turbine missiles from another AP1000 plant in close proximity affecting SSCs. The applicant proposes to add to the AP1000 DCD, Section 3.5.1.3, a statement that the potential for a turbine missile from another AP1000 plant in close proximity is less than 1×10^{-5} per year, and *that the reinforced concrete shield building and auxiliary building walls, roofs, and floors satisfies the guidance of RG 1.115 for two AP1000 plants side-by-side.*

It should be noted that AP1000 DCD, Section 1.2.2 refers to Figure 1.2 2 of the AP1000 DCD for the building structure orientation with respect to the turbine building and the nuclear island. Figure 1.2 2 illustrates the AP1000 plant as a single unit. Section 1.2.1.3.1 of the AP1000 DCD also states that the turbine orientation minimizes potential interaction between turbine missiles and safety-related structures and components. In addition, Section 3.5.1.3 of the AP1000 DCD states that the turbine generator is located north of the nuclear island with its shaft oriented north-south so that safety-related systems are located outside the high-velocity, low trajectory missile strike zone. With this information, the AP1000 design is considered to favorably orient the turbine building with respect to safety-related SSCs as defined in RG 1.115. However, since BLN Units 3 and 4 will be side-by-side, the staff notes that each turbine generator may not be oriented favorably with respect to the other plant's safety-related SSCs (i.e., BLN Unit 3 turbine generator not favorably orientated to BLN Unit 4 safety-related SSCs, and vice versa).

In Revision 1 of the BLN COL FSAR, the applicant revised STD SUP 3.5-1 to state that when two or more AP1000 units are situated side-by-side, the turbine generators are orientated unfavorably with respect to the other nuclear island which contains safety-related SSCs. The BLN site has two AP1000 units situated side-by-side. Therefore, the staff notes that to meet the guidance of RG 1.115 and Section 3.5.1.3 of NUREG-0800, for an unfavorable turbine generator orientation, the probability of generating a turbine missile must be equal to or less than $1x10^{-5}$ *per year. As stated in the BLN COL FSAR, Section 3.5.1.3, the probability of generating a missile for the AP1000 turbine generator is less than* $1x10^{-5}$ *per year as calculated in the applicable bounding turbine missile analysis topical report referenced in the AP1000 DCD, Sections 3.5.1.3 and 10.2.8. The staff has not completed its review of the DCD with respect to this issue. Therefore, the staff is unable to make final determination. This is Open Item 1-1.*

- *STD SUP 3.5-2*

STD SUP 3.5-2 to BLN COL, Section 3.5.1.3 states, "The turbine system maintenance and inspection program is discussed in Section 10.2.3.6." This statement refers to Section 10.2.3.6 of the BLN COL, for information concerning the turbine maintenance and inspection program. The staff's review of the turbine maintenance and inspection program is included in Section 10.2.3 [sic 10.2] *of this SER.*

Resolution of the Standard Content Evaluation Concerning Open Item 1-1 for Turbine Missiles

The NRC staff identified a statement in the text reproduced above from Section 3.5.1.4 of the BLN SER that requires clarification for the VEGP COL application. The BLN SER states that the review of the AP1000 DCD with respect to the probability of generating a turbine missile was not completed and, *therefore, identified it as Open Item 1-1. The results of the NRC staff's technical evaluation of the AP1000 DC amendment application are documented in NUREG-1793 and its supplements, and include the final staff conclusions on the issue of probability of a missile striking a safety-related component.*

The NRC staff identified a statement in the text reproduced above from Section 3.5.1.4 of the BLN SER that requires clarification for the VEGP COL application. The BLN SER states that the review of the AP1000 DCD with respect to the probability of generating a turbine missile was not completed and, therefore, identified it as Open Item 1-1. The results of the NRC staff's technical evaluation of the AP1000 DC amendment application are documented in NUREG-1793 and its supplements, and include the final staff conclusions on the issue of probability of a missile striking a safety-related component.

3.5.1.5 *Post Combined License Activities*

There are no post COL activities related to this section.

3.5.1.6 *Conclusion*

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the WLS COL applicant addressed the required information relating to missile protection, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements. In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the regulatory guidance in NUREG-0800, Sections 3.5.1.1 through 3.5.1.6.

- WLS COL 3.5-1 is acceptable because it meets the regulatory requirements by meeting the acceptance criteria provided in NUREG-0800, Sections 3.5.1.5 and 3.5.1.6.

- STD SUP 3.5-1 is acceptable because the turbine missile evaluation for co-located AP1000 units meets the guidance of NUREG-0800 Section 3.5.1.3; therefore, it ensures that the requirements of 10 CFR Part 50, Appendix A, GDC 4, "Environmental and Dynamic Effects Design Bases," are met for protecting safety-related SSCs against the effects of turbine missiles.

- STD SUP 3.5-2 provides information on the turbine maintenance and inspection program. The staff's review of, and conclusions on, the turbine maintenance and inspection program is included in Section 10.2 of this report.

3.5.2 Protection from Externally Generated Missiles

Systems required for safe shutdown are protected from the effects of missiles. Protection of SSCs from external missiles, including those generated by natural phenomena, is generally provided by the external walls and roof of the Seismic Category I NI structures. The external walls and roofs are generally reinforced concrete. The structural design requirements for the shield building and auxiliary building are outlined in AP1000 DCD Section 3.8.4. Where openings through these walls are provided, they are evaluated on a case-by-case basis to demonstrate that a missile passing through the opening would not prevent safe shutdown and would not result in an offsite release exceeding the limits defined in 10 CFR Part 100.

WLS COL FSAR, Revision 11, Section 3.5, incorporates by reference AP1000 DCD, Revision 19, Section 3.5.2, "Protection from Externally Generated Missiles," without any departures or supplements. The staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The staff's review confirmed that there is no outstanding issue related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

3.5.3 Barrier Design Procedures

Missile barriers and protective structures are designed to withstand and absorb missile impact loads to prevent damage to safety-related systems or components. Formulae used for missile penetration calculations into steel or concrete barriers are the Modified National Defense Research Committee formula for concrete and either the Ballistic Research Laboratory or Stanford formulae for steel as documented in AP1000 DCD Section 3.5.3.

WLS COL FSAR, Revision 11, Section 3.5, incorporates by reference AP1000, Revision 19, Section 3.5.3, "Barrier Design Procedures," without any departures or supplements. The staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The staff's review confirmed that there is no outstanding issue related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

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

3.6.1 Introduction

The design basis and criteria are described to demonstrate that safety-related systems are protected from pipe ruptures. This section also evaluates design bases for locating postulated breaks and cracks in high- and moderate-energy piping systems inside and outside the containment; the procedures used to define the jet thrust reaction at the break location; the procedures used to define the jet impingement loading on adjacent essential SSCs; pipe whip restraint design; and the protective assembly design. Pipe breaks in several high-energy systems, including the reactor coolant loop (RCL) and surge line, are replaced by small leakage cracks when the leak-before-break (LBB) criteria are applied. Jet impingement and pipe whip effects are not evaluated for these small leakage cracks. Mechanistic pipe break evaluations (also referred to as LBB) demonstrate that for piping lines meeting the criteria, sudden catastrophic failure of the pipe is not credible. The evaluations demonstrate that piping that satisfies the criteria leaks at a detectable rate from postulated flaws prior to growth of the flaw to a size that would fail due to applied loads resulting from normal conditions, anticipated transients, and a postulated SSE.

3.6.2 Summary of Application

WLS COL FSAR, Revision 11, Section 3.6, incorporates by reference AP1000 DCD, Revision 19, Section 3.6. In WLS COL FSAR Section 3.6.4, the applicant provided the following additional information:

AP1000 COL Information Items

- STD COL 3.6-1

The applicant provided information in WLS COL FSAR Section 3.6.4.1, "Pipe Break Hazard Analysis," and in 14.3.3, "CDM Section 3.0, Non-System Based Design Descriptions and ITAAC," for STD COL 3.6-1 to address the applicable AP1000 DCD information item. Specifically, the applicant stated: (1) the as-designed pipe rupture hazards evaluation will be in accordance with AP1000 DCD criteria with SSCs identified to be essential targets will be protected, and (2) that a pipe rupture hazard analysis is part of the piping design. It is used to identify postulated break locations and layout changes, support design, whip restraint design, and jet shield design for high- and moderate energy piping. The applicant further stated that the final design of these activities will be completed prior to installation of the piping and connected components. The as-built reconciliation of the pipe rupture hazards evaluation whip restraint and jet shield design will be completed prior to fuel load and will be available for NRC review.

- STD COL 3.6-4

The applicant provided information in STD COL 3.6-4 to address the information item, regarding the use of Alloy 690 and as-built verification of LBB piping.

License Condition

- Part 10, License Condition 2, "COL Holder Items," Item 3.6-1

The applicant proposed a license condition addressing the as-designed pipe rupture hazards analysis completion schedule and the contents of a related design report.

Inspections, Tests, Analyses and Acceptance Criteria (ITAAC)

In a November 3, 2010, letter, the applicant endorsed the April 23, 2010, letter from the VEGP applicant that proposed adding to the AP1000 DCD, Tier 1, Section 3.3, an ITAAC in the WLS COL, Part 10, Appendix B, "Inspections, Tests, Analysis, and Acceptance Criteria," denoted as Table 3.3-8, "Pipe Rupture Hazards Analysis (Sheet 1 of 1)," requiring the completion of an as-designed pipe rupture hazards analysis to demonstrate that SSCs required to be functional during and following a design basis event are protected against or qualified to withstand the dynamic and environmental effects with analyses of postulated pipe failures in high- and moderate-energy piping.

3.6.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. In addition, the acceptance criteria associated with the relevant requirements of NRC regulations (10 CFR Part 50, Appendix A, GDC 4) for the piping design against pipe breaks, pipe break locations and characteristics in safety-related piping, and LBB evaluation procedures are given in NUREG-0800, Sections 3.6.1, 3.6.2, and 3.6.3.

3.6.4 Technical Evaluation

The staff reviewed this application section and checked the referenced DCD to ensure that the combination of the DCD and the WLS COL application represents the complete scope of information relating to this review topic. The staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the protection against dynamic effects associated with the postulated rupture of piping. The staff's evaluation of the information incorporated by reference in the WLS application is documented in NUREG-1793 and its supplements. Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform a technical review for each "standard issue" and use this review to evaluate subsequent COL applications. To ensure that the staff's findings on standard content for the Reference COL application FSAR (i.e., VEGP Units 3 and 4) were equally applicable to the WLS application, the staff undertook the following reviews.

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In the comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the application, as applicable) resulting from responses to RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed and that those changes were actualized in the WLS COL FSAR.

- The staff verified that site-specific differences, if any, did not adversely affect any previous relevant evaluation or conclusion.

The staff completed its review and concluded that the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this report by use of italicized, double indented formatting. Section 1.2.3 of this report provides an explanation of why the standard content material from the SER for the Reference COL application (VEGP) contains evaluation material from the SER for the Bellefonte Nuclear Plant, Units 3 and 4 COL application. The staff reviewed the following information in the WLS COL FSAR.

AP1000 COL Information Items

The following portion of this technical evaluation section is reproduced from VEGP SER Section 3.6.4:

- *STD COL 3.6.-1*

The staff notes that there are two different actions to be addressed: 1) the COL holder item addresses the as-designed pipe rupture hazard analysis report; and 2) the ITAAC addresses as-built reconciliation of the pipe rupture hazard analysis report. The ITAAC has a stated schedule, prior to fuel load, and a regulatory requirement that the ITAAC schedule be provided one year after the license is granted.

Based on the review of the information included in the BLN COL FSAR, it is unclear to the staff when the as-designed pipe rupture hazard analysis report will be completed by the applicant. As identified in 10 CFR 52.79(d)(3), the applicant should supply the NRC with a schedule for completion of detailed engineering information, in this case, the as-designed pipe rupture hazard analysis report. The applicant is requested to revise the implementation milestone for the License Condition to address the as-designed pipe rupture hazard analysis report (as opposed to as-built reconciliation) to allow coordination of activities with the NRC construction inspection program following the issuance of the COL such that the analysis would be made available to verify the design was completed in accordance with the regulations and DCD prior to fabrication and installation of the piping and connected components. In RAI 3.6.2-1, the staff requested the applicant provide a description pertaining to the closure milestone of the as-designed pipe rupture hazard analysis activities.

The applicant responded to RAI 3.6.2-1, however, based on its review of the applicant's response, the staff determined that it is not acceptable. Specifically, RAI 3.6.2-1 requested that the applicant address the implementation milestone of the as-designed pipe rupture hazard analysis report. However, the applicant's RAI response addressed the as-built rather than the as-designed aspect. Therefore, RAI 3.6.2-1 remains unresolved and will be tracked as Open Item 3.6-1.

- *STD COL 3.6-4*

The BLN COL FSAR replaced the first paragraph of Section 3.6.4.4 of AP1000 DCD with the following text:

> *Alloy 690 is not used in leak-before-break [LBB] piping. No additional or augmented inspections are required beyond the inservice inspection [ISI] program for leak-before-break [LBB] piping. An as-built verification of the leak-before-break piping is required to verify that no change was introduced that would invalidate the conclusion reached in this subsection.*

Based upon its review of the replaced Section 3.6.4.4, the staff determined that additional information was needed by the COL applicant to address whether Alloy 690 material is being used in the BLN-specific LBB piping systems. Accordingly, the staff issued several RAIs.

In RAI 3.6.3-1, the staff noted that it was unclear why Alloy 690 was not used in LBB piping applications. If Alloy 690 base material and Alloy 52/152 weld material was not being used, the staff asked the applicant to identify what material was being used for the piping.

In RAI 3.6.3-2, the staff asked if another base material was being used other than Alloy 690/52/152, then the applicant should provide its reasons for using this material in LBB piping applications based upon operating experience, and provide justification as to why no augmented inspection plans and evaluation criteria were considered necessary. Additionally, the staff requested that the applicant provide a discussion which supports the use of an alternative material *and discuss why concerns for potential PWSCC [primary water stress-corrosion cracking] should not be considered a factor.*

In RAI 3.6.3-3, for piping requiring dissimilar metal welds, the applicant was requested to address that if Alloy 52/152 is not being used for the weld material, then they should identify the weld material and provide justification for its use. In addition, the applicant should provide a discussion which supports the use of an alternative weld material and why concerns regarding the potential for PWSCC should not be considered a factor. The staff noted that there are currently ASME Code cases being developed for dissimilar-metal welds due to PWSCC concerns.

In its response to these RAIs, the applicant provided additional information to clarify the material that is used for LBB piping systems. The applicant stated that there is some limited use of Alloy 690 base material as safe ends in components connected to LBB piping, and there is some limited use of Alloy 52/152 weld material associated with these safe ends. However, the applicant noted that the base material for most of the LBB piping is 316LN stainless steel material. The applicant further stated that the material used in the AP1000 LBB piping is the same material currently used for LBB piping in operating nuclear power plants. Alloy 690 and Alloy 600 are not used as base material for LBB piping in the AP1000 design and are not commonly used in the LBB piping in current operating nuclear power plants. The applicant also stated that even though the material used in the LBB piping for the AP1000 design do not presently require an augmented ISI program, if ASME Code cases are developed and approved to address PWSCC concerns for dissimilar metal welds used in the AP1000 DCD, they will be evaluated and implemented.

The staff notes that in a final rule to amend 10 CFR 50.55a (73 FR [Federal Register] 52730) issued on September 10, 2008, a new requirement was added for licensees to augment their ISI program to use ASME Code Case N-722 for ISI of Alloy 600/182/82 materials to address PWSCC concerns. The applicant stated that there will be no Alloy 600/182/82 material used for new reactor construction of AP1000 plants. The staff notes that the final rule did not impose any additional requirements for augmented ISI of Alloy 690/152/52 materials. Based on the applicant's response discussed above and its commitment to evaluate and implement ASME Code cases that are developed and approved for augmented inspections of Alloy 690/152/52 material to address PWSCC concerns, the staff concludes the applicant's changes to COL Information Item 3.6-4 is consistent with current industry practice and NRC regulations as amended in 10 CFR 50.55a and is thus, acceptable.

<u>Resolution of Standard Content Open Item 3.6-1</u>

To address Open Item 3.6-1 in the BLN SER with open items, the VEGP applicant proposed in its letter dated April 23, 2010, an ITAAC for as-designed pipe rupture hazards analysis in ITAAC Table 3.8-# [where # is the next sequential number] and a revision to the proposed License Condition 2, Item 3.6-1 in Part 10 of the VEGP COL application. In addition, the applicant *proposed to revise VEGP COL FSAR Section 3.6.4.1 and to add VEGP COL FSAR Section 14.3.3related to pipe rupture hazards analysis.*

Specifically, the proposed ITAAC includes a post-COL requirement related to the completion of the as-designed pipe rupture hazards analysis report. The proposed VEGP COL FSAR Section 3.6.4.1 states that the completed as-designed pipe rupture hazards analysis will be in accordance with the criteria outlined in AP1000 DCD Sections 3.6.1.3.2 and 3.6.2.5. The applicant stated that the completed as-designed pipe rupture hazards analysis report will be completed prior to installation of the piping and connected components and will be made available to the NRC staff. The applicant's proposed license condition that will require completion of the as-designed pipe rupture hazards analysis report prior to installation of the piping and connected components in their final location is proposed License Condition 2, Item 3.6-1. In the proposed VEGP COL FSAR Section 14.3.3.# [where # is the next sequential number], the applicant stated that the as-designed pipe rupture hazards analysis completed for the first standard AP1000 plant will be available to subsequent standard AP1000 plants under the "one issue, one review, one position" approach for closure.

The staff reviewed the applicant's April 23, 2010, response to BLN open items for Chapter 3, and has determined that the use of a plant-specific ITAAC to verify that the as-design pipe rupture hazards evaluation has been performed in accordance with the criteria outlined in AP1000 DCD Sections 3.6.1.3.2 and 3.6.2.5 is acceptable. The applicant's proposed license condition requiring completion of the as-designed pipe rupture hazards analysis report prior to installation of the piping and connected components in their final location, through the above discussed ITAAC, will allow the staff sufficient time to review the as-design pipe rupture hazards evaluation in a timely matter in order to identify and address any design issues. Therefore, the staff finds the response acceptable and concludes that Standard Content Open Item 3.6-1 has been satisfactorily resolved. The incorporation of the planned VEGP COL FSAR changes will be tracked as Confirmatory Item 3.6-1.

<u>Resolution of Standard Content Confirmatory Item 3.6-1</u>

Confirmatory Item 3.6-1 is an applicant commitment to revise its FSAR Section 3.6.4.1 and, Section 14.3.3.2, to verify the incorporation of the as-designed pipe rupture hazard analysis and add an ITAAC (Table 3.8-1) for the as-designed pipe rupture hazard analysis. The staff verified that the VEGP COL FSAR and Part 10 of the application (ITAAC Table 3.8-1) were appropriately updated. As a result, Confirmatory Item 3.6-1 is now closed.

- WLS COL 3.6-1

The staff reviewed WLS COL 3.6-1 included under WLS COL FSAR Section 3.6. The applicant replaced the last paragraph in AP1000 DCD Section 3.6.4.1, stating that after a COL is issued, the COL holder will complete an as-designed pipe rupture hazard evaluation that will be available for review. The evaluations will be provided prior to fabrication and installation of the piping and connected parts. In a July 22, 2011, letter, the applicant committed to remove this additional information because the standard content provided in WLS COL FSAR, Revision 2 provides all the necessary information for resolving STD COL 3.6-1. This is being tracked as Confirmatory Item 3.6-2.

Resolution of Confirmatory Item 3.6-2

Confirmatory Item 3.6-2 was an applicant commitment to remove excess information from the WLS COL FSAR Section 3.6.4.1. The staff verified that the information was removed. As a result, Confirmatory Item 3.6-2 is now closed.

3.6.5 Post Combined License Activities

The license condition language in this section has been clarified from previously considered language. In a letter dated March 22, 2016 (ADAMS Accession No. ML16084A099), the applicant did not identify any concerns with the clarified license condition language. The changes do not affect the staff's above analysis of the conditions, and therefore, for the reasons discussed in the technical evaluation section above, the staff finds the following ITAAC and license condition acceptable:

- The licensee shall perform and satisfy the pipe rupture hazards analysis described in WLS COLA, Part 10, Appendix B, ITAAC Table 3.3-8, "Pipe Rupture Hazards Analysis (Sheet 1 of 1)."

- License Condition (3-1) – Before commencing installation of individual piping segments and connected components in their final locations, the licensee shall complete the as-designed pipe rupture hazards analysis for compartments (rooms) containing those segments in accordance with the criteria outlined in the AP1000 DCD, Rev. 19, Sections 3.6.1.3.2 and 3.6.2.5, and shall inform the Director of NRO, or the Director's designee, in writing, upon the completion of this analysis and the availability of the as-designed pipe rupture hazards analysis reports.

3.6.6 Conclusion

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to the pipe design against pipe break, pipe break locations, characteristics in safety-related piping, and LBB evaluation procedures and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the relevant information presented in the WLS COL FSAR is acceptable and meets the requirements of 10 CFR Part 50, Appendix A, GDC 4. The staff based its conclusion on the following:

- STD COL 3.6-1 is acceptable because the applicant's proposed resolution in WLS COL FSAR Section 3.6.4.1 meets the relevant guidelines of NUREG-0800, Sections 3.6.1 and 3.6.2 and 10 CFR 52.79(d)(3) and is, thus, acceptable. Conformance to these guidelines provides an acceptable basis to satisfy, in part, the requirements of 10 CFR Part 50, Appendix A, GDC 4.

- STD COL 3.6-4 is acceptable because the applicant's proposed resolution in WLS COL FSAR Section 3.6.4.4 meets the relevant guidelines of NUREG-0800, Section 3.6.3 and RG 1.206, Section C.III.1, Chapter 3, C.I.3.6.3 and is, thus, acceptable. Conformance to these guidelines provides an acceptable basis to satisfy, in part, the requirements of 10 CFR Part 50, Appendix A, GDC 4

3.7 Seismic Design

Section 3.7 of this report focuses on the seismic analyses of NI structures (Seismic Category I) and adjacent Seismic Category II structures. Seismic design of the AP1000 Seismic Category I and II structures, systems, equipment, and components are based on the SSE. Low-level seismic effects are included in the design of certain equipment that are potentially sensitive to a number of low-level events based on a percentage of the responses calculated for the SSE. Criteria for evaluating the need to shut down the plant following an earthquake are established. Seismic Category I SSCs are designed to withstand the effects of the SSE and maintain the specified design functions. Seismic Category II and NS SSCs are designed or physically arranged (or both) so that the SSE would not cause unacceptable structural interaction with or failure of Seismic Category I SSCs.

On April 25, 2012, the staff issued RAI 105, Question 01.05-1 pertaining to the implementation of the Fukushima Near-Term Task Force (NTTF) Recommendations for the WLS. RAI 105, Question 01.05-1 addressed NTTF Recommendation 2.1, requested that the applicant (1) evaluate the potential impacts of the new Central and Eastern United States Seismic Source Characterization model (CEUS-SSC; NUREG-2115) on the seismic hazard curves and the site-specific ground motion response spectra (GMRS)/foundation input response spectra (FIRS) and (2) modify the GMRS/FIRS as necessary.

In a January 30, 2014, response, the applicant addressed the implementation of the CEUS-SSC in updating the site-specific seismic hazards and response spectra for the WLS and the respective effects on the Nuclear Island (NI) structures. In a supplemental response, dated February 28, 2014, the applicant addressed the effects of the updated site-specific hazards on the seismic Category II structures adjacent to the NI. The aforementioned responses also included changes to the WLS FSAR. The staff's review of the development of the updated site-specific hazard is discussed in Section 2.5 of this report. Section 3.7 of this report focuses on the seismic analyses of NI structures and adjacent seismic Category II structures. These seismic analyses are based on the new WLS site-specific seismic hazard. On the basis of the above discussion, the staff considers this RAI resolved

3.7.1 Seismic Design Parameters

3.7.1.1 *Introduction*

In this section, the input seismic design GMRS for the SSE in the free field at plant grade is addressed.

3.7.1.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 3.7, incorporates by reference AP1000 DCD, Revision 19, In addition, in WLS COL FSAR Section 3.7, the applicant provided the following:

Departures

- WLS DEP 2.0-1

The seismic design of the AP1000 standard plant is based on the Certified Seismic Design Response Spectra (CSDRS) as addressed in AP1000 DCD Section 3.7.1.1. The AP1000 DCD also includes hard rock high frequency (HRHF) spectra for evaluation of site-specific GMRS. The WLS site-specific horizontal and vertical spectra exceed the CSDRS and HRHF spectra; therefore, constituting a departure from the AP1000 certified design. This departure is identified in WLS COL FSAR Table 2.0-201 and WLS COL FSAR Sections 3.7.1.1.1, 3.7.2.8.4, 3.7.2.15, Appendix 3I, and WLS COL FSAR Section 19.55.6.3. Consistent with the requirements of the AP1000 DCD Tier 1, Table 5.0-1, AP1000 DCD Section 2.5.2.1, paragraph 4b, and AP1000 DCD Section 3.7.2.8.4, the applicant performed site-specific analysis to demonstrate the adequacy of the standard design for the WLS site. The staff's evaluation of WLS DEP 2.0-1 and supporting site-specific analysis is included in Section 3.7.2.4 of this report.

Supplemental Information

- WLS SUP 3.7-3

The applicant provided information in WLS SUP 3.7-3 by adding Section 3.7.1.1.1, "Design Foundation Spectra," to the WLS COL FSAR, with site-specific foundation input response spectra for each WLS unit.

3.7.1.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. The acceptance criteria associated with the relevant requirements of NRC regulations for the seismic design parameters are given in NUREG-0800, Section 3.7.1 and interim staff guidance (ISG) in the form of DC/COL-ISG-1, "Interim Staff Guidance on Seismic Issues of High Frequency Ground Motion in Design Certification and Combined License Applications."

3.7.1.4 *Technical Evaluation*

The horizontal and vertical design GMRS for the AP1000 were developed based on the response spectra in RG 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," Revision 1, with special consideration of high-frequency amplification effects. The staff reviewed WLS COL FSAR Section 3.7 and checked the referenced DCD to ensure that the combination of the DCD and the WLS COL application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and that incorporated by reference addresses the required information relating to seismic design parameters. The results of the staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements. The staff's review of the information in the WLS COL FSAR is as follows:

Supplemental Information

- WLS SUP 3.7-3

WLS SUP 3.7-3 addresses site-specific NI FIRS for each of the WLS units, separately. The applicant stated that individual foundation response spectra were provided for the certified design portion of the Units 1 and 2 plants based on their unique foundation conditions. WLS COL FSAR Sections 2.5.2.7 and 3.7.1.1.1 describe the applicant's site-specific seismic velocity model and associated calculation of the WLS site-specific FIRS. The applicant developed site-specific dynamic velocity models and used the probabilistic method of random vibration theory (RVT), as described in NUREG/CR-6728 as Approach 3, to compute the location-specific FIRS at the WLS site.

The applicant developed three site-specific dynamic velocity models to represent localized foundation conditions – Unit 1 A1, Unit 1 A5, and Unit 2 C4. WLS COL FSAR Section 2.5.4.7 describes the material dynamic properties and WLS COL FSAR Figures 2.5.4-252a, 2.5.4-252b, and 2.5.4-252c show the dynamic velocity profiles for Base Cases A1, A5, and C4, respectively, which represent the Unit 1 FIRS A1, Unit 1 FIRS A5, and Unit 2 FIRS C4 configurations.

As described in WLS COL FSAR Section 2.5.2.7 the WLS Unit 1, FIRS A1 represents the Unit 1 nuclear island centerline foundation input motion that supports the Unit 1 nuclear island. The applicant based its Unit 1 FIRS A1 on the GMRS developed at the hard rock condition, as defined by CEUS attenuation relationships (2,830 m/s (9,282 fps)), at 161 m (530 ft) (NAVD), transferred up through previously placed Cherokee Nuclear Station (CNS) concrete materials and newly placed WLS concrete materials to the basemat foundation level at 168.7 m (553.5 ft) (NAVD). The applicant's site-specific dynamic velocity model for Unit 1 A1 contains approximately 2.4 m (8 ft) of new fill concrete overlying an average of about 4.5 m (15 ft) of existing fill concrete, structural basemat concrete and native rock from the former CNS foundation. In total, the applicant's Unit 1 FIRS A1 site-specific seismic velocity profile contains 7 m (23.5 ft) of fill concrete material of shear wave velocities of 2286 m/s (7,500 fps) overlying hard rock (WLS COL FSAR Figure 2.5.4-252a).

The applicant also described that the WLS Unit 1 FIRS A5 represents the Unit 1 localized condition where the nuclear island overlies legacy CNS pump rooms. The applicant based its WLS FIRS A5 on the WLS GMRS developed at the hard rock condition and transferred up through 9 m (30.5 ft) of previously placed CNS concrete materials and newly placed WLS concrete materials to the basemat foundation level at 168.7 m (553.5 ft) (NAVD) (WLS COL FSAR Figure 2.5.4-252b).

The applicant described that the WLS Unit 2 FIRS C4 represents the eastern edge of the Unit 2 nuclear island which may be supported by up to 6 m (20 ft) of new leveling fill concrete. The applicant's Unit 2 FIRS C4 is based on the WLS GMRS developed at the top of the hard rock condition fixed at 155 m (509 ft) (NAVD) and transferred up through 6 m (20 ft) of newly placed WLS concrete materials to the basemat foundation level at 168.7 m (553.5 ft) (NAVD) (WLS COL FSAR Figure 2.5.4-252c). The applicant randomly varied each site-specific seismic velocity base profile by ± 1 m (± 3 ft).

The applicant first generated randomized site-specific seismic velocity model profiles and associated shear moduli and damping parameters that represent possible variations from the base seismic mode, consistent with RG 1.208. Then the applicant calculated site-specific response amplification functions for each randomized profile using the RVT methodology. The use of RVT in site response calculations is described in RG 1.208 as a possible methodology that can be used. RG 1.208 specifically states, "…RVT methods are acceptable as long as the strain dependent soil properties are adequately accounted for in the analysis." Similar to the time series methodology, RVT analysis produces an amplification function that is then applied to the rock spectra to obtain the response spectra defined at the ground surface (or at any intermediate point within the soil profile), which accounts for the effects of soil amplification (or deamplification) on the input base hard rock ground motion. To accommodate the possibility of distance-dependent transfer functions in a linear analysis, the applicant used a suite of spectral shapes as control motions at distances of 1, 20, 100, 200, and 400 km (0.6, 12, 62,125, 250 mi). The applicant shows those results in WLS COL FSAR Figures 2.5.2-241a, 2.5.2-241b, and 2.5.2-241c, which display a small site resonance at high frequencies. Following the guidance of RG 1.208, the staff focused its review on the input parameters used in the site amplification functions calculations. Inputs to the RVT method include response spectra which are based on the hard rock UHRS, 60 randomized velocity profiles, effective strain ratio, and strong motion duration. Having reviewed the applicant's input parameters; the staff concludes that the applicant's selection of input parameters is adequate to calculate the FIRS amplification functions at the WLS site.

In WLS COL FSAR Section 3.7.1.1.1, the applicant described that it used the envelope of the Unit 1 and Unit 2 FIRS, referring to the enveloped spectra as NI FIRS. The resulting horizontal NI FIRS is shown in Figure 3.7.1.4-1 of this report and the vertical NI FIRS is shown in WLS COL FSAR Figure 3.7-202.

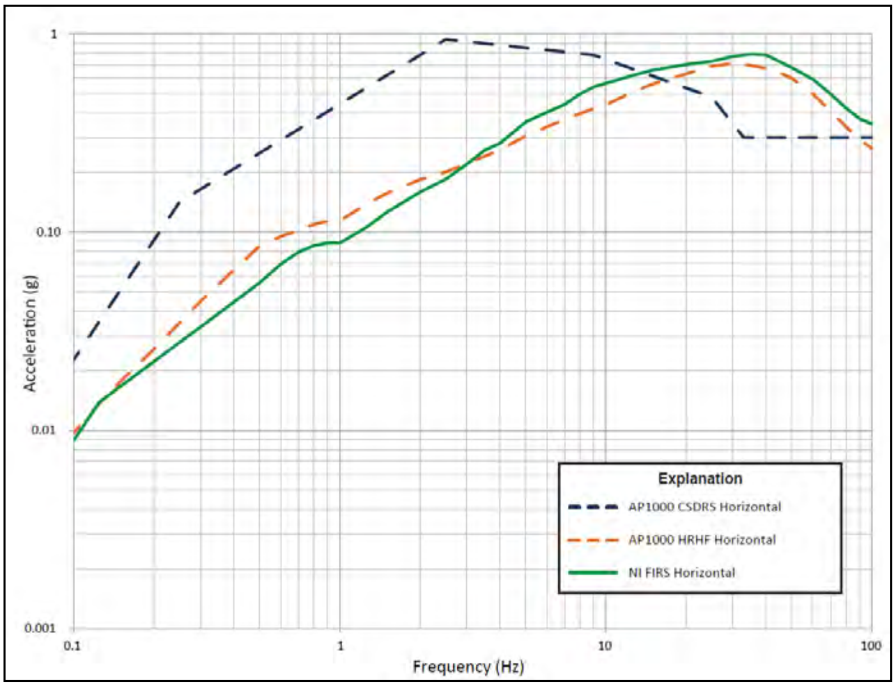


Figure 3.7.1.4-1 The WLS horizontal NI FIRS, the AP1000 horizontal certified seismic design response spectra (CSDRS), and the AP1000 horizontal hard rock high frequency spectra (HRHF). (Ref. WLS COL FSAR Revision 9, FSAR 3.7-201)

To determine the adequacy of the applicant's FIRS calculations, the staff completed confirmatory calculations. As input, the staff used the static and dynamic properties provided in WLS COL FSAR Section 2.5.4 and 2.5.2.7 for base cases A1, A5, and C4. To represent the input rock motions, the staff used the applicant's low- and high-frequency 10^{-4} and 10^{-5} rock spectra. The staff completed its site response calculations using the Strata software (Kottke and Rathje, 2008). The staff compared its calculated horizontal NI FIRS with the applicant's in Figure 3.7.1.4-2 of this report. Figure 3.7.1.4-2 of this report shows that the staff's NI FIRS calculation is similar to the applicant's NI FIRS across the frequency range typically important for engineering purposes (i.e., 0.5 to 10 Hz).

The staff noted a difference between the applicant's and the staff's NI FIRS results at high frequencies (between approximately 40 and 100 Hz). The staff noted the WLS FIRS amplification transfer functions show amplification from approximately 30 to 100 Hz due to the varying fill concrete beneath the WLS proposed structures, which means that at those frequencies the FIRS should have larger spectral accelerations than the WLS site-specific GMRS. In contrast, the WLS NI FIRS is less than the WLS GMRS at high frequencies. To investigate this issue, in RAI 118, Question 03.07.01-6, the staff requested that the applicant assess and evaluate the inconsistency of WLS FIRS amplification transfer functions, FIRS, and the GMRS and to provide a thorough description of the assessment and evaluation. The staff also requested that the applicant explain the basis and justify the conclusions regarding the relationship between the WLS site-specific amplification functions, FIRS, and GMRS.

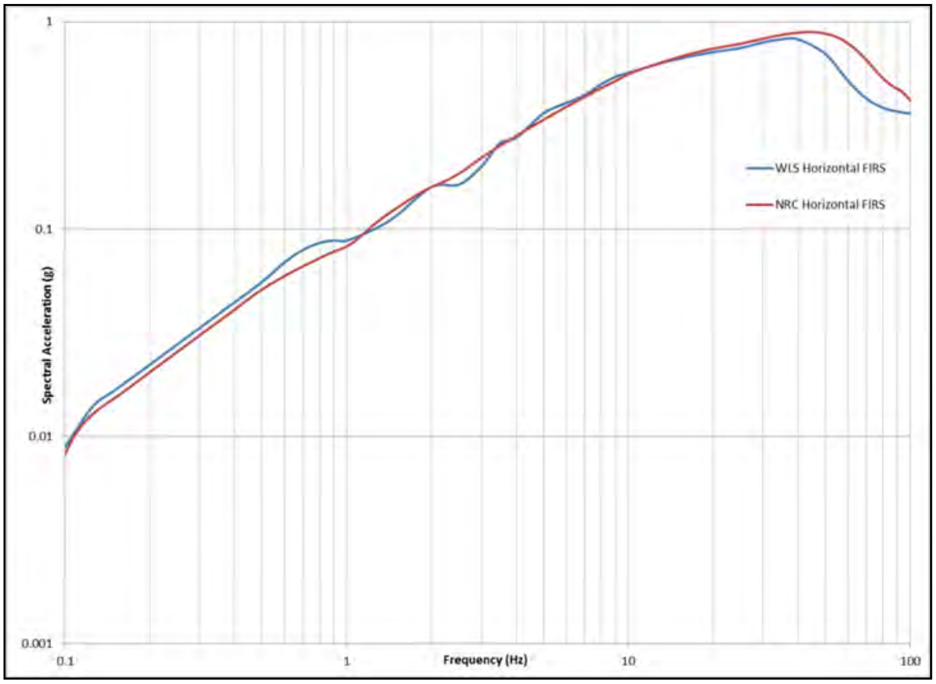


Figure 3.7.1.4-2 The WLS horizontal NI FIRS and the staff's (NRC) horizontal NI FIRS

In a July 24, 2014, response to RAI 118, Question 03.07.01-6, the applicant described that the noted differences in the relationships of the horizontal GMRS and FIRS are attributed to differences in high-frequency interpolation methods used to develop the rock (GMRS) and concrete (FIRS) spectral shapes. In Figure 3.7.1.4-3 of this report, the applicant presented the horizontal GMRS and FIRS A1 from WLS COL FSAR, Sections 2.5.2.6 and 2.5.2.7 (labeled as

"Orig") and compared them to alternative calculations of the horizontal GMRS and FIRS A1 based on the FIRS high-frequency interpolation method ("Altern"). The applicant's RAI response results demonstrated that using the same interpolation methods lead to consistent spectral shapes and resulting in a GMRS that is generally lower than the GMRS presented in WLS COL FSAR Revision 9.

Based on this review, the staff concludes that the WLS COL FSAR Revision 9 horizontal GMRS and FIRS spectra are considered appropriate. Additionally, the staff also concludes that the difference in interpolation methods explains the differences between the WLS NI FIRS and the NRC NI FIRS shown in Figure 3.7.1.4-2 of this report. Accordingly, the staff concludes that the WLS NI FIRS satisfy the acceptance criteria of RG 1.208 and the requirements of 10 CFR 100.23, and considers RAI 118, Question 03.07.01-6 resolved. Based upon its review of WLS COL FSAR Section 2.5.2.7 and 3.7.1.1.1, the applicant's response to RAI 118, Question 03.07.01-6, and the staff's confirmatory analysis, the staff concludes that the applicant's methodology and results for calculations of the site-specific NI FIRS are acceptable and resolved.

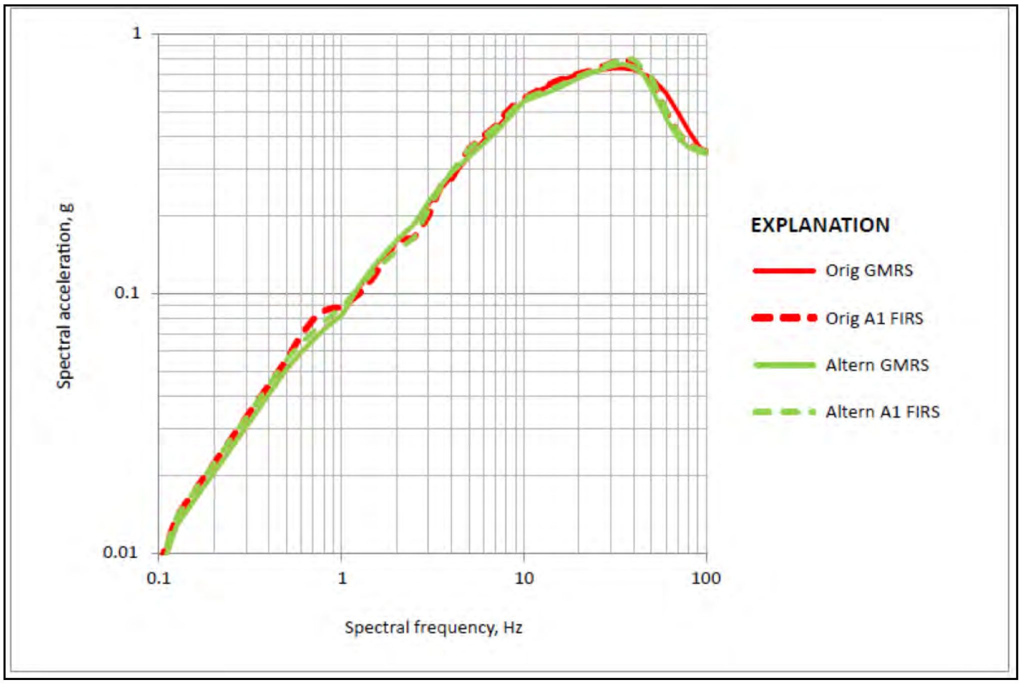


Figure 3.7.1.4-3 Comparison of the horizontal GMRS and FIRS A1 from the FSAR ("Orig") and horizontal GMRS and FIRS A1 using the methodology consistent with that used to develop the FSAR NI FIRS ("Altern"). (Ref. RAI 03.07.01-6 response Figure 9)

As shown in WLS COL FSAR Figures 3.7-201 (Figure 3.7.1.4-1 of this report) and 3.7-202, the horizontal NI FIRS exceeds the standard horizontal AP1000 CSDRS at frequencies above approximately 14 Hertz (Hz) and the vertical NI FIRS exceeds the CSDRS at frequencies above approximately 16 Hz. The peak ground acceleration (PGA) values for horizontal and vertical NI FIRS are 0.35g and 0.32g, respectively. The applicant also provided a comparison of the NI FIRS to the HRHF spectra. The horizontal NI FIRS exceeds the horizontal AP1000 HRHF spectrum above 3 Hz and the vertical NI FIRS exceeds the vertical HRHF at frequencies between about 3 and 55 Hz and between 80 and 100 Hz. The aforementioned exceedances of the NI FIRS to both the CSDRS and the HRHF spectra are identified by the applicant as departure WLS DEP 2.0-1. To address this departure, the applicant stated that consistent with AP1000 DCD Section 2.5.2.1, paragraph 4b, a site-specific analysis of the AP1000 has been performed, similar to the analysis described in AP1000 DCD Appendix 3I, to demonstrate that the high frequency spectra exceedances are within the seismic design margin of the AP1000 certified design and will not adversely affect the structures, systems, or components of the plant.

The WLS site-specific analyses for NI structures are described in WLS COL FSAR Section 3.7.2.15 and discussed in detail in Reference 206 to WLS COL FSAR Section 3.7 (“Effects of William S. Lee Site Specific Seismic Requirements on AP1000 SSCs,” WLG-GW-GLR-815, Revision 0, January 30, 2014). Reference 206, Section 2.0 states that two site-specific Soil Structure Interaction (SSI) models were developed with dynamic soil profiles corresponding to the varied conditions beneath the WLS Unit 1 and 2 NIs. WLS COL FSAR Figures 2.5.4-250 and 2.5.4-252a show the dynamic velocity profiles for Unit 2 and Unit 1 respectively. The shear wave velocity immediately below Unit 2 is greater than 2,500 m/s (8,000 fps) and approximately 2,900 m/s (9,500 fps) at lower depths, whereas Unit 1 has a shear wave velocity of 2,300 m/s (7,500 fps_ (i.e., concrete layer) immediately below its NI foundation and approximately 2,900 m/s (9,500 fps) at lower depths. Reference 206, states that a comparison of the WLS SSI in-structure response spectra (ISRS) at the six key NI locations for Units 1 and 2 was made for both dynamic profiles and determined to be similar. On this basis, the WLS site-specific analyses are based on the SSI model with the Unit 1 dynamic velocity profile, which includes the effects of fill concrete beneath the NI basemat. The staff reviewed Reference 206 and noted that the applicant did not provide the ISRS comparisons for the models with Units1 and Unit 2 dynamic soil profiles. These ISRS comparisons were reviewed by the staff during the May 2014 WLS structural audit. Additionally, the staff issued RAI 119, Question 03.07.01-7, requesting that the applicant provide the ISRS (for Units 1 and 2 dynamic velocity profiles) at the six key NI locations. In an August 14, 2014, response, the applicant provided the ISRS at the six key locations for the two SSI analyses, one based on Unit 1 dynamic soil profile and the other based on Unit 2 dynamic soil profile. The staff also noted that the ISRS at the six key locations corresponding to the SSI models with Units 1 and 2 dynamic soil profiles were consistent in spectral shape with negligible differences in spectral acceleration amplitude. Additionally, the staff noted that the HRHF analyses in AP1000 DCD Appendix 3I are based on a shear wave limitation defined at the bottom of the basemat equal to or higher than 2286 m/s (7,500 fps), while maintaining a shear wave velocity equal to or above 2438 m/s (8,000 fps) at the lower depths. Based on the consistency with the limiting lower bound shear wave velocity for the AP1000 DCD Appendix 3I analyses and the consistent results of the ISRS at the six key locations for Units 1 and 2 dynamic profiles, the staff finds the Unit 1 dynamic profile acceptable for use in the WLS site-specific SSI analyses and considers the RAI resolved.

3.7.1.5 *Post Combined License Activities*

There are no post COL activities related to this section.

3.7.1.6 *Conclusion*

The staff reviewed the application and checked the referenced DCD. The staff’s review confirmed that the applicant addressed the required information relating to the seismic design parameters, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements. In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the requirements of 10 CFR Part 50, Appendix A, Appendix S, and other pertinent staff guidance. The staff based its conclusion on the following:

- WLS SUP 3.7-3 is acceptable because the applicant addressed the relevant information and performed adequate analyses that meet the acceptance criteria in NUREG-0800, Section 3.7.1 and ISG-1. In conclusion, the applicant has provided sufficient information to comply with the requirements of 10 CFR Part 50, Appendix A, GDC 2; 10 CFR Part 50, Appendix S; and 10 CFR 100.23.

3.7.2 Seismic System Analysis

3.7.2.1 *Introduction*

This section describes seismic analysis methods and acceptance criteria for all Seismic Category I SSCs. The description includes basic assumptions, procedures for modeling, seismic analysis methods, development of ISRS envelopes, consideration of torsional effects, evaluation of overturning and sliding of Seismic Category I structures, and determination of composite damping. The effects of soil structure interaction (SSI) on the seismic responses of the NI structures are included in the section. The staff review also covered design criteria and procedures for evaluating the interaction of NS Category I structures with Seismic Category I structures and the effects of parameter variations on floor response spectra (FRS).

3.7.2.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 3.7, incorporates by reference AP1000 DCD, Revision 19. In WLS COL FSAR Section 3.7.2, the applicant provided the following departure and supplements:

Departures

- WLS DEP 2.0-1

The seismic design of the AP1000 certified nuclear plant is based, in part, on the CSDRS as addressed in AP1000 DCD Section 3.7.1.1. The AP1000 DCD also includes HRHF spectra, which provide an alternative set of spectra for evaluation of site-specific GMRS. The WLS site-specific horizontal and vertical spectra exceed the CSDRS and HRHF spectra, constituting a departure from the AP1000 certified design. This departure is identified in WLS COL FSAR Table 2.0-201 and Sections 3.7.1.1.1, 3.7.2.8.4, 3.7.2.15, AP1000 DCD Appendix 3I, and Section 19.55.6.3. Consistent with the requirements of AP1000 DCD Table 5.0-1 (Tier 1), AP1000 DCD Section 2.5.2.1, paragraph 4b, and AP1000 DCD Section 3.7.2.8.4, the applicant performed site-specific analysis to demonstrate the adequacy of the standard design for the WLS site. The staff‘s evaluation of WLS DEP 2.0-1 and supporting site-specific analysis is included in Section 3.7.2.4 of this report.

AP1000 COL Information Items

- WLS COL 3.7-1

The applicant provided information in WLS COL 3.7-1 in WLS COL FSAR Sections 3.7.2.12 and 3.7.5.1, regarding seismic analysis of existing and new dams that could affect the site to address COL Information Item 3.7- discussed in AP1000 DCD Section 3.7.5.1. The information in WLS COL FSAR Section 3.7.5.1 references Section 3.7.2.12 and will be discussed in Section 3.7.2.4 of this report and will not be discussed elsewhere in this report.

- STD COL 3.7-3

The applicant provided information to address STD COL 3.7-3 on the seismic interaction review, which is discussed in AP1000 DCD Section 3.7.5.3. The information added to the AP1000 DCD to address STD COL 3.7-3 is located in WLS COL FSAR Section 3.7.5.3 and is the subject of a proposed license condition (Part 10, License Condition 2, "COL Holder Items," Item 3.7-3, "Seismic Interaction Review," below). STD COL 3.7-3 will not be discussed elsewhere in this report.

- STD COL 3.7-4

The applicant provided information in STD COL 3.7-4 on the reconciliation of seismic analyses of NI structures, which is discussed in AP1000 DCD Section 3.7.5.4. The information added to the AP1000 DCD to address STD COL 3.7-4 is located in WLS COL FSAR Section 3.7.5.4 and is the subject of a proposed license condition (Part 10, License Condition 2, "COL Holder Items," Item 3.7-4, "Reconciliation of Seismic Analyses of Nuclear Island Structures," below). This COL item will not be discussed elsewhere in this report.

Supplemental Information

- WLS SUP 3.7-4

The applicant provided information at the end of AP1000 DCD Section 3.7.2.8.4 regarding seismic modeling and analysis of Seismic Category II building structures.

- WLS SUP 3.7-5

The applicant provided information in WLS SUP 3.7.5 by adding Section 3.7.2.15 to the WLS COL FSAR which addresses site-specific analysis of NI Seismic Category I structures.

- WLS SUP 3.7-6

The applicant added supplemental information to the end of AP1000 DCD Section 3.7.2.1.2 regarding the development of time histories for use in site-specific analyses.

License Conditions

- Part 10, License Condition 2, "COL Holder Items," Item 3.7-3

The applicant proposed a license condition requiring an update to the seismic interaction review for as-built information to be completed prior to initial fuel load. This review is performed in parallel with the seismic margin evaluation and will follow the methodology in AP1000 DCD Section 3.7.5.3. The review is based on as-procured data, as well as the as-constructed condition.

- Part 10, License Condition 2, "COL Holder Items," Item 3.7-4

The applicant proposed a license condition requiring a reconciliation of seismic analysis for detail design changes, such as those due to as-procured or as-built changes in component mass, center of gravity, and support configuration based on as-procured equipment information. Acceptance criteria for deviations are specified. The reconciliation of seismic analysis of NI structures will be completed prior to initial fuel load.

3.7.2.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. In addition, the acceptance criteria associated with the relevant requirements of NRC regulations for the seismic system analysis are given in NUREG-0800, Section 3.7.2 and ISG-1, which provides guidance on implementation of evaluation methodology to determine the effects of high-frequency ground motion.

3.7.2.4 *Technical Evaluation*

The staff reviewed this application and checked the referenced DCD to ensure that the combination of the DCD and the WLS COL application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and that incorporated by reference addresses the required information relating to the seismic system analysis. The staff's evaluation of the information incorporated by reference in the WLS COL application is documented in NUREG-1793 and its supplements. Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform a technical review for each "standard issue" and use this review to evaluate subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the Reference COL application FSAR (i.e., VEGP Units 3 and 4) were equally applicable to the WLS application, the staff undertook the following reviews.

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In the comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the application, as applicable) resulting from responses to RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed and that those changes were actualized in the WLS COL FSAR.

- The staff verified that the site-specific differences, if any, did not adversely affect any previous relevant evaluation or conclusion.

The staff completed its review and concluded that the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this report by use of italicized, double-indented formatting. Section 1.2.3 of this report provides an explanation of why the standard content material from the SER for the Reference COL application (i.e., VEGP) contains evaluation material from the SER for the BLN, Units 3 and 4 COL application. The staff reviewed the following information in the WLS COL FSAR:

Departures

- WLS DEP 2.0-1

The staff's evaluation of WLS DEP 2.0-1 is accomplished through the evaluation of three supplemental information sections: WLS SUP 3.7-4, WLS SUP 3.7-5, and WLS SUP 3.7-6. This supplemental information is evaluated below.

AP1000 COL Information item

- WLS COL 3.7-1

The staff reviewed the resolution to the COL information item related to the evaluation of existing and new dams included under WLS COL FSAR Section 3.7.2.12. WLS COL 3.7-1 addresses the evaluation of existing and new dams whose failure could affect the site interface flood level specified in AP1000 DCD Section 2.4.1.2. The applicant references WLS COL FSAR Section 2.4.4 for the details of the evaluation. The applicant stated that the WLS site is not subject to flooding from dam failures. The staff's review of WLS COL FSAR Section 2.4. found it to be acceptable. Therefore, the staff finds the information in the WLS COL FSAR addressing WLS COL 3.7-1 acceptable.

Supplemental Information

- WLS SUP 3.7-4

WLS SUP 3.7-4 addresses the site-specific seismic modeling and analysis of Seismic Category II building structures (i.e., Turbine Building First Bay (TBFB) and Annex Building). The purpose of the modeling and analysis is to demonstrate that adverse interaction between the Seismic Category II buildings and the adjacent NI is precluded under the site-specific demands. The staff's review focused on the adequacy of the minimum seismic gap, the adequacy of the bearing capacity of the backfill under the Category II buildings, the overall global stability of the Seismic Category II structures, and the adequacy of the Seismic Category II building designs to preclude local member failure and a consequential adverse interaction with the adjacent NI.

In the application, the analyses are also discussed in detail in the, "William S. Lee Site Specific Adjacent Buildings Seismic Evaluation Report," WLG-1000-S2R-804, Revision 3, February 2014 (Reference 205). The applicant's site-specific SSI analysis of these structures is performed with site-specific and building-specific performance-based surface response spectra (PBSRS) at plant grade as input. These PBSRS were developed using the same analytical methods used in calculating the Unit 1 FIRS. Furthermore, these analyses used hazard-consistent, strain-compatible properties for the granular material supporting the Seismic Category II building structures. Three soil profiles were considered in the analyses including a best estimate (BE), lower bound (LB), and upper bound (UP) soil profiles. These analyses, consistent with the AP1000 DCD analyses of Seismic Category II structures, are 2D SSI analyses with lump mass stick models of the Seismic Category II adjacent structures.

Reference 205 provided the calculated site-specific displacements of the Seismic Category II building structures relative to the NI. As stated in Reference 205, Section 6.2, relative displacements were calculated to verify that there would be no contact between the NI and the Seismic Category II adjacent structures. As per AP1000 DCD, Revision 19, Section 3.8.5.1, buildings adjacent to the NI, such as the TBFB and Annex Building are structurally separated from NI structures by a 5-cm (2-in.) gap at and below grade and a minimum gap of 10-cm (4-in.) above grade. The applicant provided the site-specific relative displacements at four locations including, (1) TBFB foundation to NI, (2) top of the TBFB to the NI (El. 52 m (170 ft)), (3) Annex Building foundation to the NI, and (4) top of Annex Building to the NI (El. 55 m (180 ft)). The staff reviewed Reference 205, Section 6.2 and noted that the relative displacements for the TBFB bound those of the Annex Building. The staff notes that the calculated maximum relative displacements for the TBFB are 0.5 cm (0.197 in,) below grade and 1.5 cm (0.576 in.) above grade, which are significantly lower than the aforementioned minimum required gaps between the NI and adjacent structures. On this basis, the staff finds that the minimum required seismic gaps in the AP1000 DCD continue to provide adequate separation to prevent interaction between the NI and the adjacent Seismic Category II structures at the WLS site.

In RAI 120, Question 03.07.02-4, the staff requested that the applicant include in the WLS COL FSAR the site-specific values for the relative displacements between the NI and the adjacent Seismic Category II structures. In addition, the staff requested that the applicant include in the WLS COL FSAR the relative displacements for the 1.67 times the site-specific seismic demand. In an August 7, 2014, response, the applicant provided markups to WLS COL FSAR Section 3.7.2.8.4 that addressed the relative displacement information. On this basis, the staff considered the applicant's response acceptable. Accordingly, the staff considers RAI 120, Question 03.07.02-4 resolved.

In terms of the backfill bearing pressure, Reference 205 provided the site-specific dynamic bearing pressure demand vs capacity comparisons. Consistent with the AP1000 analysis of Seismic Category II structures, these pressure demands combine dead weight and seismic pressures to obtain a total bearing pressure on the soil elements under the basemat of the Seismic Category II structures. These comparisons showed that the bearing capacity of the supporting granular fill is greater than the bearing demand for the seismic Category II structures under the site-specific demands. On this basis the staff finds that the supporting granular fill is adequate for withstanding the bearing demand from the Seismic Category II structures at the WLS site.

To ensure no potential adverse interaction between the NI and the Seismic Category II structures as a result of global sliding or overturning of the Seismic Category II buildings under the effects of the site-specific seismic hazard, in RAI 121, Question 03.08.05-7, the staff requested that the applicant provide the factors of safety against sliding and overturning for the Seismic Category II structures for both the site-specific seismic demand and the 1.67 times the site-specific seismic demand. In an August 14, 2014, response, the applicant provided a qualitative assessment regarding the sliding and overturning stability of the Seismic Category II structures. The staff reviewed the applicant's response and noted that it contained insufficient details pertaining to the stability of the Seismic Category II structures. In a clarification call with the applicant, the staff requested that the applicant provide additional information including the calculation of factors of safety against sliding and overturning of the Seismic Category II structures. In an October 22, 2014, supplemental response, the applicant provided the factors of safety against sliding and overturning for the Seismic Category II structures. The staff's review of the factors of safety for the 1.67 times the site-specific demand is in Section 19.55 of this report. The smallest site-specific factors of safety for demand to resist sliding and overturning were 2.6 and 2.3, respectively. These factors of safety are greater than the

1.1 minimum required factor of safety for the NI specified in AP1000 DCD Table 3.8.5-1. Furthermore, the applicant provided markups to WLS COL FSAR Section 3.7.2.8.4, which addressed the sliding and overturning consideration in the detailed design of Seismic Category II structures. Based on the site-specific factors of safety provided by the applicant, the staff finds sufficient margin exists for Seismic Category II buildings against sliding and overturning as to preclude adverse interaction with the adjacent NI under the site-specific seismic demands. As such, the minimum seismic gap is adequate for the WLS site. Therefore, the staff finds this acceptable and, therefore, considers RAI 121, Question 03.08.05-7 resolved.

In terms of the impact of the site hazard exceedance above the CSDRS and HRHF on the design of the Seismic Category II buildings and the potential for failure leading to an adverse impact on the adjacent NI, the staff reviewed the applicant's approach to address this issue. Consistent with AP1000 DCD Section 3.7.2.8.4, the applicant computed the WLS site-specific Seismic Category II foundation response spectra (FRS) and provided comparisons of the site specific FRS with the corresponding AP1000 generic design envelop spectra. The generic design envelop spectra in AP1000 DCD Section 3.7.2.8.4, are those used in the design of Seismic Category II structures. Comparisons provided by the applicant are found in WLS COL FSAR Figures 3.7-213a and 3.7-213g for the Annex Building and 3.7-214a and 3.7-214b for the TBFB. These figures show that the site-specific horizontal FRS fall beneath the AP1000 envelope FRS for both the TBFB and the Annex Building, except for a minor exceedance between 3 Hz and 5 Hz for the TBFB. In the vertical direction, for both the TBFB and the Annex Building, there are vertical FRS exceedances between approximately 6 Hz and 25 Hz. The staff assessed the impact of the exceedance and concluded that both the horizontal and vertical exceedances have a negligible impact on the global stability of the Seismic Category II buildings, as demonstrated by sufficient margin against sliding and overturning as well as the small displacements of the Seismic Category II building structures relative to the NI. Furthermore, the applicant stated in WLS DEP 2.0-1 and in WLS COL FSAR Section 3.7.2.8.4 that the design of the Seismic Category II buildings will be performed to both the AP1000 generic design demand envelop and the WLS site-specific demands to ensure that the Seismic Category II building members are adequately designed so as not to fail and potentially negatively impact the adjacent NI SSCs. The staff finds that the design of the Seismic Category II buildings to the larger of the demands obtained from the AP1000 generic design envelop spectra and the WLS site-specific spectra to be adequate to ensure that these buildings will have adequate member capacity to preclude failure and avoid negatively impacting the safety-related functions of the adjacent NI SSCs.

Based on its review, the staff finds that the applicant has provided adequate and sufficient information to demonstrate that the Seismic Category II buildings will be supported and designed so as to preclude adverse interactions with the adjacent Seismic Category I SSCs when subjected to the WLS site-specific seismic hazard. Accordingly, WLS SUP 3.7-4 and the associated portion of DEP 2.0-1 are acceptable.

- WLS SUP 3.7-5

WLS SUP 3.7.5 addresses site-specific analyses of the NI Seismic Category I structures. The purpose of the site-specific analysis is to demonstrate that the NI Seismic Category I structures site-specific seismic demands are enveloped by the generic AP1000 DCD seismic demands. The staff's review focused on the adequacy of the modeling approach, the appropriate application of the incoherency function, and the verification of the adequacy of the AP1000 DCD demands for the design of the NI structures. WLS COL FSAR Section 3.7.2.15 describes the NI analyses that included three-dimensional incoherent SSI analysis based on the NI FIRS. The WLS NI FIRS exceeds both the AP1000 CSDRS and HRHF. As such, the applicant stated that site-specific analyses are performed to confirm that the site specific demands will not adversely affect the SSCs of the WLS. This site-specific evaluation implements the evaluation described in AP1000 DCD Appendix 3I, which describes the methodology and criteria used in the evaluation to confirm that the HRHF input is not damaging to the equipment and structures qualified by the analysis for the AP1000 CSDRS. The staff notes that while WLS COL FSAR Section 3.7.2.15 also discusses the applicant's site-specific evaluation for primary coolant loop, piping, and equipment, this report section focuses on the staff's evaluation of the structures. The staff's review of the applicant's evaluation regarding primary coolant loop, piping, and equipment is discussed in Sections 3.9, 3.12, and 3.10 of this report, respectively.

The methodology in the AP1000 DCD, Appendix 3I uses the NI20 nuclear island model described in AP1000 DCD Appendix 3G. The WLS site-specific analyses use an updated version of the NI20 model referred to as the NI20u model. The updates included in NI20u are described in Reference 206 and were discussed in detail during the May 2014 audit. The NI10 model was used in the AP1000 DCD analyses for defining the seismic response for the hard rock site. Additionally, AP1000 DCD Appendix 3G, Section 3G.2.2.2 states that the results of fixed base analyses of the NI20 model were compared to those of the NI10 model to confirm the adequacy of the NI20 model for use in the SSI analyses. In Reference 206, the applicant provided the ISRS at the six AP1000 key locations, comparing the response between the NI20u model and the NI10 model. The results provided were for fixed base analyses. During the May 2014 audit, the staff reviewed additional ISRS (at additional locations) and effective modal mass comparisons between the NI20u and NI10 models. During the review the staff noted that the comparisons included in Reference 206 and the ones presented during the May 2014 audit, demonstrate dynamic equivalence between the models. Additionally, the staff noted that the NI20u model yields responses that are consistent with the NI10 model and conservative with respect to the NI10 response. The staff also reviewed the model updates included in the NI20u model in conjunction with the NI20 model parameters provided in AP1000 DCD Appendix 3G. The staff reviewed the updates included in the NI20u model and noted that the model updates do not deviate from or alter the model parameters provided in AP1000 DCD Appendix 3G and are acceptable. Based on its consistency with the AP1000 DCD and its conservative response prediction discussed above, the staff finds the NI20u model adequate for the WLS site-specific evaluation.

As stated above, the WLS site-specific evaluation implements the three-dimensional incoherent SSI analysis based on the NI FIRS. In Reference 206, Section 5, the applicant stated that ACS SASSI (ACS SASSI by Ghiocel Predictive Technologies Inc. is a program that performs seismic soil-structure interaction (SSI) analysis) is used to perform 25 simulations with the incoherency function using the same methodology described in AP1000 DCD Appendix 3I, which, consistent with the staff guidance in ISG-1, uses the 2007 Abrahamson Hard-Rock coherency model. Additionally, the staff notes that Reference 206 Appendix A provides comparisons of WLS site-specific ISRS using both coherent and incoherent input motions. The staff noted that, while these comparisons identify the impacted frequency ranges due to incoherency and show that the incoherent response is generally lower than the coherent response, these comparisons did not explicitly indicate the percentage reduction to the coherent motion. As such, to assist the staff in its review of the incoherency function's impact on the demands, in RAI 120, Question 03.07.02-3, the staff requested that the applicant quantify the range of reductions to the coherent motion for both the site-specific evaluation and to compare it to the evaluation performed during the AP1000 certification. In an August 7, 2014, response, the applicant provided plots of reductions factors versus frequency for all the locations in Reference 206, Appendix A and for the AP1000 HRHF and the site-specific evaluations. The staff reviewed the applicant's response and noted the maximum levels of site-specific reduction to the coherent motions are consistent with the AP1000 HRHF evaluation. Therefore, based on the similar ISRS reductions of the WLS site-specific evaluation using the three-dimensional incoherent SSI analysis and the AP1000 HRHF calculations, the staff finds the applicant's reduction to coherent motion acceptable and, therefore, considers RAI 120, Question 03.07.02-3 resolved.

WLS COL FSAR Figures 3.7-209a through 3.7-211c compare the ISRS (at the AP1000 six key locations) corresponding to the AP1000 CSDRS, AP1000 HRHF (incoherent response), and the site-specific NI-FIRS (incoherent response). Furthermore, Reference 206, Figures 5.4-1 through 5.4-16 show the ISRS comparisons at additional locations consistent with the locations in the AP1000 DCD Appendix 3I evaluation. The staff reviewed these comparisons and noted that the WLS site-specific ISRS are largely enveloped by either the AP1000 CSDRS or HRHF FRS, except for a few exceedances generally in the high frequency range. As discussed in WLS COL FSAR Section 3.7.2.1.5, additional evaluations are performed to address the effect of site-specific ISRS exceedances. To assess the significance of the exceedances at the ISRS level on the NI structures, the applicant evaluated and compared the loads obtained from the site-specific NI FIRS and those obtained from the AP1000 CSDRS. These evaluations were performed for representative portions of the building structures selected by screening as being potentially sensitive to high frequency input. Moreover, as stated in Reference 206, Section 6.1, the selection of these representative portions of the building structures is based on areas that can experience high seismic demands during a seismic event. These representative portions of the building structures, consistent with AP1000 DCD Appendix 3I, include three locations in the Auxiliary Building, eight locations in the Shield Building, and three areas in the Containment Internal Structures (CIS). Reference 206, Figures 6.1-2 through 6.1-6 and Tables 6.1-1 through 6.1-6 show the locations for the screened representative portion of building structures and the member force comparisons, respectively. The staff reviewed Reference 206, Tables 6.1-1 through 6.1-6 and noted that the member forces resulting from the AP1000 CSDRS are greater than those resulting from the WLS NI FIRS for all the screened representative portions of building structures. On this basis, the staff finds that the member forces resulting from the AP1000 CSDRS envelope those resulting from the WLS NI FIRS and, therefore, the design of the WLS NI structures using the AP1000 CSDRS input is acceptable.

The staff notes that the WLS COL FSAR supplemented information from the AP1000 DCD Appendix 3I. Specifically, the applicant identified WLS DEP 2.0-1 and added information to AP1000 DCD Sections 3I.1, 3I.2, 3I.3, 3I.6, 3I.6.1, 3I.6.2, 3I.6.3, 3I.6.4, and 3I.7. This added information describes the applicant's site-specific analysis supporting WLS DEP 2.0-1.
In relation to the applicant's assessment of structures, the staff noted that the additional information in AP1000 DCD Appendix 3I is consistent with the supplemental information provided in WLS COL FSAR Section 3.7 and, as such, it is addressed by the review and conclusions in Section 3.7 of this report. The staff reviewed the site-specific SSI analyses performed by the applicant to evaluate the exceedances between the WLS NI FIRS and the CSDRS and the HRHF spectra. The staff concludes that the WLS site-specific SSI analyses demonstrate that the AP1000 standard plant structural demands are adequate to use for the design of the NI SSCs at the WLS site. On this basis, the staff finds the WLS SUP 3.7-5 and the associated portions of WLS DEP 2.0-1 acceptable.

- WLS SUP 3.7-6

WLS SUP 3.7-6 addressed the development of time histories for use in site-specific analyses. The earthquake record (CHICHI/ILA031) from the 1999 Chi-Chi earthquake was selected from NUREG/CR-6728 (Magnitude>7, Distance=50-100 km, WUS Rock bin) as the seed time history record for the site-specific analyses. The time histories in this seed record (i.e., two horizontal and one vertical time histories) are modified so as to match the site-specific NI FIRS. As stated in WLS COL FSAR Section 3.7.2.12, the time step interval for the modified time histories is no more than 0.005 seconds and the total duration is no less than 30 seconds. (WLS COL FSAR Figures 3.7-203a to 3.7-203c show a total duration of about 90 seconds.) Furthermore, WLS COL FSAR Table 3.7-201 indicates strong motion duration greater than 6 seconds and cross correlation coefficient lower than 0.16 for the 3 time history components. During the May 2014 audit, the staff reviewed the development of the time histories for use in site-specific analyses including response spectra matching and power spectral density (PSD) calculations. The staff confirmed that the development of the time histories was performed in accordance with SRP Section 3.7.1, Option 1 - Approach 2. The staff finds that the developed time histories provide an acceptable representation of the WLS NI FIRS and, as such, acceptable for use in WLS site-specific analyses. Accordingly, WLS SUP 3.7-6 and the associated portions of WLS DEP 2.0-1 are acceptable.

The following portion of this technical evaluation section is reproduced from Section 3.7.2.4 of the VEGP SER:

<u>License Conditions</u>

- *Part 10, License Condition 2, Item 3.7-3*

The applicant has proposed a license condition requiring a seismic interaction review by the licensee for as-built information. This review is performed in parallel with the seismic margin evaluation. The review is based on as-procured data, as well as the as-constructed condition. The as-built seismic interaction review is to be completed prior to fuel load. The Staff has reviewed and approved this review methodology in Section 3.7.5.3 of the AP1000 DCD. Therefore, the staff finds the proposed License Condition 2 acceptable.

- *Part 10, License Condition 2, Item 3.7-4*

The applicant has proposed a license condition requiring a seismic analysis for detail design changes, such as those due to as-procured or as-built changes in component mass, center of gravity, and support configuration based on as-procured equipment information. The reconciliation of seismic analysis of NI structures will be complete prior to fuel load.

Conducting the seismic interaction review and the seismic analysis for detail design changes based on as-procured data, as well as the as-constructed condition, does not alter the methods of seismic evaluation required to ensure the as-built design parameters are consistent with the standard design and have been reviewed by the staff as part of VEGP COL 3.7-1, as well as the information incorporated by reference from the AP1000 DCD. In addition, the NRC staff understands and agrees with the need to have as-procured data and the as-constructed condition in order to properly conduct these analyses.

3.7.2.5 *Post Combined License Activities*

The staff notes that Part 10 of the WLS application, Proposed License Conditions (including ITAAC), includes License Condition 14 and ITAAC Table 3.3-10. The staff's review determined that this license condition and ITAAC are not needed because the information sought is adequately addressed in the application. As such, License Condition 14 and the Seismic Category II ITAAC are not addressed here and should be removed from the next revision of the WLS COL FSAR. This issue is being tracked as Confirmatory Item 3.7-1.

<u>Resolution of Confirmatory Item 3.7-1</u>

Confirmatory Item 3.7-1 is an applicant commitment to revise Part 10 of the WLS application to remove License Condition 14 and the Seismic Category II ITAAC. The staff verified that Part 10 of the WLS application was appropriately revised. As a result, Confirmatory Item 3.7-1 is now closed.

The license condition language in this section has been clarified from previously considered language. In a letter dated March 22, 2016 (ADAMS Accession No. ML16084A099), the applicant did not identify any concerns with the clarified license condition language. The changes do not affect the staff's above analysis of the conditions, and therefore, for the reasons discussed in the technical evaluation section above, the staff finds the following license conditions acceptable:

- License Condition (3-2) – Before initial fuel load, the licensee shall update the seismic interaction analysis in AP1000 DCD, Rev. 19, Section 3.7.5.3 to reflect as-built information, which must be based on as-procured data, as well as the as-constructed condition.

- License Condition (3-3) – Before initial fuel load, the licensee shall reconcile the seismic analyses described in Section 3.7.2 of the AP1000 DCD, Rev. 19, to account for detailed design changes, including, but not limited to, those due to as-procured or as-built changes in component mass, center of gravity, and support configuration based on as-procured equipment information.

3.7.2.6 *Conclusion*

The staff's review confirmed that the applicant addressed the required information relating to the seismic system analysis, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the requirements of 10 CFR Part 50, Appendix A, Appendix S, and other relevant staff guidance.

- WLS DEP 2.0-1 is acceptable because the applicant's site-specific analysis demonstrated that the AP1000 DCD design is adequate for use at the WLS site and this analysis addressed the relevant information that meets the guidance in NUREG-0800, Section 3.7.2 and ISG-1. In conclusion, the applicant has provided sufficient information to satisfy the requirements in 10 CFR Part 50, Appendix A, GDC 2; 10 CFR Part 50, Appendix S; and 10 CFR 100.23.

- WLS COL 3.7-1 is acceptable because the applicant addressed the relevant information that meets the guidance in NUREG-0800, Section 3.7.2. In conclusion, the applicant has provided sufficient information to satisfy the requirements in 10 CFR Part 50, Appendix A, GDC 2; 10 CFR Part 50, Appendix S; and 10 CFR 100.23

- WLS SUP 3.7-4 is acceptable because the applicant addressed the relevant information and performed adequate analyses that meet the guidance in NUREG-0800, Section 3.7.2. In conclusion, the applicant has provided sufficient information to satisfy the requirements in 10 CFR Part 50, Appendix A, GDC 2; 10 CFR Part 50, Appendix S; and 10 CFR 100.23.

- WLS SUP 3.7-5 is acceptable because the applicant addressed the relevant information and performed adequate analyses that meet the guidance in NUREG-0800, Section 3.7.2 and ISG-1. In conclusion, the applicant has provided sufficient information to satisfy the requirements in 10 CFR Part 50, Appendix A, GDC 2; 10 CFR Part 50, Appendix S; and 10 CFR 100.23.

- WLS SUP 3.7-6 is acceptable because the applicant addressed the relevant information and performed adequate analyses that meet the guidance in NUREG-0800, Section 3.7.1. In conclusion, the applicant has provided sufficient information to satisfy the requirements in 10 CFR Part 50, Appendix A, GDC 2; 10 CFR Part 50, Appendix S; and 10 CFR 100.23.

3.7.3 Seismic Subsystem Analysis

Seismic input motion, seismic analysis methods, and modeling procedure used for the analysis and design of AP1000 SC-I subsystems are described. In particular, this review focused on such subsystems as the miscellaneous steel platforms, steel frame structures, tanks, cable trays and supports, heating, ventilation, and air conditioning (HVAC) ductwork and supports, and conduit and supports.

Specifically, the criteria and methods for the seismic analysis of safety-related SSCs and equipment include the following:

- Seismic analysis methods
- Determination of number of earthquake cycles
- Procedures used for modeling
- Basis for selection of frequencies
- Equivalent static load method of analysis
- Three components of earthquake motion
- Combination of modal responses
- Analysis procedure for piping
- Vertical static factors
- Torsional effect of eccentric mass
- Seismic Category I buried piping systems and tunnels

- Interaction of other systems with seismic Category I systems
- Seismic analysis of reactor internals
- Analysis procedure for damping
- Analysis of seismic Category I tanks
- Time history analysis of piping systems

Section 3.7 of the WLS COL FSAR, Revision 11, incorporates by reference, Section 3.7.3, "Seismic Subsystem Analysis," of Revision 19 of the AP1000 DCD. In addition, in WLS COL FSAR Section 3.7, the applicant provided the following:

Departures

- WLS DEP 6.4-2

The applicant provided additional information in Table 3.7-207 of the WLS COL FSAR about WLS DEP 6.4-2 related to design changes affecting how the temperature and humidity in the main control room are maintained within the limits for reliable human performance. This information, as well as related WLS DEP 6.4-2 information appearing in other chapters of the FSAR, is reviewed in Section 21.3 of this SER.

The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

3.7.4 Seismic Instrumentation

3.7.4.1 *Introduction*

Installation of instrumentation that is capable of adequately measuring the effects of an earthquake at the plant site is addressed. The criteria for the seismic instrumentation include a comparison with RG 1.12, "Nuclear Power Plant Instrumentation for Earthquakes," Revision 2, the location and description of instrumentation, control room operator notifications, comparison of measured and predicted responses, and tests and inspections.

3.7.4.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 3.7 incorporates by reference AP1000 DCD, Section 3.7. The applicant provided the following information in WLS COL FSAR Section 3.7.4:

AP1000 COL Information Item

- STD COL 3.7-2

The applicant provided information for STD COL 3.7-2 in WLS COL FSAR Section 3.7.4.4 to resolve COL Information Item 3.7-2 on post-earthquake procedures to compare measured and predicted ground motions. In STD COL 3.7-2, the applicant stated that post-earthquake operating procedures utilize the guidance of Electric Power Research Institute (EPRI) Reports NP-5930, "A Criterion for Determining Exceedance of the Operating Basis Earthquake";

TR-100082, “Seismic Instrumentation in Nuclear Power Plants for Response to OBE Exceedance: Guideline for Implementation”; and NP-6695, “Guidelines for Nuclear Plant Response to an Earthquake” as modified and endorsed by the NRC in RG 1.166, “Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Post-earthquake Actions”; and RG 1.167, “Restart of a Nuclear Power Plant Shut Down by a Seismic Event.” A response spectrum check up to 10 Hz will be based on the foundation instrument. The cumulative absolute velocity (CAV) will be calculated based on the recorded motions at the free field instrument. If the OBE ground motion is exceeded or significant plant damage occurs, the plant must be shutdown in an orderly manner. In addition, in WLS COL FSAR Section 3.7.4.4 addresses measurement of post-seismic event gaps between the new fuel rack and walls of the new fuel storage pit, between the individual spent fuel racks, and from the spent fuel racks to the spent fuel pool walls.

- STD COL 3.7-5

The applicant provided information in STD COL 3.7-5 in WLS COL FSAR Section 3.7.4.2.1, “Tri-axial Acceleration Sensors,” to resolve COL Information Item 3.7-5. In STD COL 3.7-5, the applicant stated that a free-field sensor will be located and installed to record the ground surface motion representative of the site. It will be located such that the effects associated with surface features, buildings, and components on the recorded ground motion will be insignificant. The “trigger value” is also described.

Supplemental Information

- STD SUP 3.7-1

The applicant provided supplemental information in WLS COL FSAR Section 3.7.4.1 to address the guidance in RG 1.12 by stating that administrative procedures define the maintenance and repair of the seismic instrumentation to keep the maximum number of instruments in service during plant operation and shutdown.

- STD SUP 3.7-2

The applicant provided supplemental information in WLS COL FSAR Section 3.7.4.4 to address the test and inspection requirements for the acceleration sensors. In this section, the applicant stated that installation and acceptance testing of the tri-axial acceleration sensors described in AP1000 DCD Section 3.7.4.2.1 is completed prior to initial startup. Installation and acceptance testing of the time-history analyzer described in AP1000 DCD Section 3.7.4.2.2 is completed prior to initial startup.

Interface Requirements

AP1000 DCD Table 1.8-1, “Summary of AP1000 Plant Interfaces with Remainder of Plant,” Items 3.3, “Site seismic sensor location and ‘trigger’ value,” and 3.12, “Earthquake response procedures,” refer to interfaces associated with AP1000 DCD Section 3.7.4. The interface requirements for staff review (associated with AP1000 DCD Section 3.7.4.2) include an onsite implementation of the site seismic sensor locations and trigger values, and development of procedures by the COL applicant for earthquake responses from the seismic instrumentation.

3.7.4.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. The acceptance criteria associated with the relevant requirements of NRC regulations for seismic instrumentation are given in NUREG-0800, Section 3.7.4.

The regulatory requirements and guidance documents for STD COLs 3.7-2 and 3.7-5 and STD SUP 3.7-1 and STD SUP 3.7-2 are 10 CFR Part 50, Appendix S, RG 1.166, RG 1.167, and RG 1.12, which provide for installation of free field tri-axial acceleration sensors and establishment of post-earthquake procedures for comparing measured and predicted responses.

3.7.4.4 *Technical Evaluation*

The staff reviewed this application section and checked the referenced DCD to ensure that the combination of the DCD and the WLS COL application represents the complete scope of information relating to this review topic. The staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the seismic instrumentation. The staff's evaluation of the information incorporated by reference in the WLS application is documented in NUREG-1793 and its supplements. Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform a technical review for each "standard issue" and use this review to evaluate subsequent COL applications. To ensure that the staff's findings on standard content for the Reference COL application FSAR (i.e., VEGP Units 3 and 4) were equally applicable to the WLS application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In the comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the application, as applicable) resulting from responses to RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed and that those changes were actualized in the WLS COL FSAR.

- The staff verified that site-specific differences, if any, did not adversely affect any previous relevant evaluation or conclusion.

The staff completed its review and concluded that the evaluation performed for the standard content is directly applicable to the WLS COL application. This standard content material is identified in this report by use of italicized, double indented formatting. Section 1.2.3 of this report provides an explanation of why the standard content material from the SER for the Reference COL application (i.e., VEGP) contains evaluation material from the SER for the BLN, Units 3 and 4 COL application. The one notable difference between the VEGP and WLS applications for these COL items is the specification in VEGP COL 3.7-5 that the free-field sensor is located on the ground surface of the engineering backfill. In the WLS COL FSAR, the exact location of the tri-axial ground surface acceleration free-field sensor is not specified, but will be installed using NRC-approved methodology, and will use the same trigger value, and the staff concludes that this minor difference does not negatively affect the conclusion reached previously by the staff. The staff reviewed the following information in the WLS COL FSAR:

The following portion of this technical evaluation section is reproduced from VEGP SER Section 3.7.4.4:

<u>AP1000 COL Information Items</u>

- *STD COL 3.7-2*

As a result of the review in Sections 9.1.1.2 and 9.1.2.2 of the AP1000 DCD, STD COL 3.7-2 in Section 3.7.4.4 of the VEGP COL FSAR was identified to clarify the measurement of the post-seismic event gaps between the new fuel rack and walls of the new fuel storage pit, between the individual spent fuel racks, and from the spent fuel racks to the spent fuel pool wall. In a letter dated October 15, 2010, the applicant committed to specify the site-specific procedures, following the guidance of EPRI Reports NP-5930, TR-10082, and NP-6695, for: 1) checking the gaps between the new fuel rack and walls of the new fuel storage pit, between the individual spent fuel racks, and from the spent fuel racks to the spent fuel pool walls following an earthquake; and 2) to take, if needed, appropriate corrective actions in the event of an earthquake such as repositioning the racks or analysis of the as-found condition. The staff considered the applicant response to be acceptable based on the applicant's commitment to use the post-earthquake procedures described in Section 3.7.5.2 of the AP1000 DCD, which comply with the requirements of Appendix S to 10 CFR Part 50. Therefore, the NRC staff considers STD COL 3.7-2 to be resolved. The incorporation of the planned VEGP COL FSAR changes will be tracked as Confirmatory Item 3.7-2.

<u>Resolution of Confirmatory Item 3.7-2</u>

Confirmatory Item 3.7-2 is an applicant commitment to revise its FSAR to adjust the left margin annotations related to STD COL 3.7-2. The staff verified that the VEGP COL FSAR was appropriately updated. As a result, Confirmatory Item 3.7-2 is now closed.

- *VEGP COL 3.7-2*

The NRC staff reviewed VEGP COL 3.7-2 related to COL Information Item 3.7-2 (COL Action Item 3.7.5-2) included under Section 3.7.4.4 of the VEGP COL FSAR.

The applicant provided additional information in VEGP COL 3.7-2 to resolve COL Information Item 3.7-2. COL Information Item 3.7-2 states:

> *Combined License applicants referencing the AP1000 certified design will prepare site-specific procedures for activities following an earthquake. These procedures will be used to accurately determine both the response spectrum and the cumulative absolute velocity of the recorded earthquake ground motion from the seismic instrumentation system. The procedures and the data from the seismic instrumentation system will provide sufficient information to guide the operator on a timely basis to determine if the level of earthquake ground motion requiring shutdown has*

> *been exceeded. The procedures will follow the guidance of EPRI Reports NP-5930, TR-100082, and NP-6695, as modified by the NRC staff.*

The commitment was also captured as COL Action Item 3.7.5-2 in Appendix F of NUREG-1793, which states:

> *The COL applicant will specify site-specific procedures for activities following an earthquake and those procedures will follow the guidance of Reports NP-5930, TR-100082, and NP-6695 promulgated by the Electric Power Research Institute (EPRI).*

In VEGP COL 3.7-2, the applicant stated the following:

> *Post-earthquake operating procedures utilize the guidance of EPRI Reports NP-5930, TR-100082, and NP-6695, as modified and endorsed by the NRC in Regulatory Guides 1.166 and 1.167. A response spectrum check up to 10Hz will be based on the foundation instrument. The cumulative absolute velocity will be calculated based on the recorded motions at the free field instrument. If the operating basis earthquake ground motion is exceeded or significant plant damage occurs, the plant must be shutdown in an orderly manner.*

The NRC staff reviewed the resolution to VEGP COL 3.7-2 related to comparison of measured and predicted seismic responses included under Section 3.7.4.4 of the VEGP COL FSAR. The applicant committed to specify site-specific procedures, which follow the guidance of EPRI Reports NP-5930, TR-10082, and NP-6695, for activities following an earthquake, which were endorsed by RGs 1.166 and 1.167. In RAI 3.7.4-1, issued to the BLN applicant, the staff asked the applicant to clarify if CAV will be used as one of the criteria to determine if a power plant should be shutdown should the OBE ground motion be exceeded or significant plant damage occurs. The BLN applicant responded by stating "As indicated in FSAR Subsection 3.7.4.4, use of the guidance of Regulatory Guide 1.166 and NP-5930 signifies that CAV is to be used as one of the post-earthquake criteria for determining whether the plant should be shutdown. In addition, BLN COL FSAR Appendix 1AA indicates conformance to the guidance of Regulatory Guide 1.166." The staff considered the applicant's response to be adequate because the BLN applicant confirmed that it will use the recommended criteria from the RG 1.166 to determine a potential plant shutdown, and the staff concludes that this RAI is closed. Furthermore, the BLN response to RAI 3.7.4-4 was endorsed as standard for VEGP by SNC letter dated December 17, 2008.

Based on the VEPG applicant's commitment to use the procedures accepted by NRC for post-earthquake activities and the clarification on the use of CAV in RAI 3.7.4-1, the NRC staff concludes that the applicant provided adequate information regarding the post earthquake activities and procedures to determine if a power plant needs to be shutdown and considers VEGP COL 3.7-2 resolved.

- *VEGP COL 3.7-5*

The applicant provided additional information in VEGP COL 3.7-5 to resolve COL Information Item 3.7-5 (COL Action Item 3.7.5-4) included under Section 3.7.4.2.1 of the VEGP COL FSAR. COL Information Item 3.7-5 states:

> *The Combined License applicant will determine the location for the free-field acceleration sensor as described in [DCD] Subsection 3.7.4.2.1.*

The commitment was also captured as COL Action Item 3.7.5-4 in Appendix F of NUREG-1793, which states:

> *The COL applicant will determine the location for the free-field acceleration sensor.*

In VEGP COL 3.7-5, the applicant stated the following:

> *A free-field sensor will be located and installed to record the ground surface motion representative of the site. To be representative of this site in regards to seismic response of structures, systems, and components, the free-field sensor is located on the ground surface of the engineered backfill. The backfill directly supports the Nuclear Island and the adjacent structures and extends out from these structures a significant distance. The free field sensor is located where the backfill vertically extends from the top of the Blue Bluff Marl to the ground surface, but horizontally at a distance where possible effects on recorded ground motion associated with surface features, buildings, and components would be minimized. The trigger value is initially set at 0.01g.*

The NRC staff reviewed the resolution to VEGP COL 3.7-5 related to triaxial acceleration sensors included under Section 3.7.4.2.1 of the VEGP COL FSAR. The applicant used the guidance in RGs 1.166 and 1.167 and supplemented information in the DCD with appropriate content, as required by Appendix S to 10 CFR Part 50. The applicant also committed to determining the location of the free field acceleration sensor and installing the sensor in a protected area. Based on the applicant's commitment to determine the location of the free-field acceleration sensor and the description of the location provided in STD COL 3.7-5, the staff concludes that the applicant presented sufficient information on the description and locations of field triaxial acceleration sensors and considers VEGP COL 3.7-5 resolved.

<u>*Supplemental information*</u>

- *STD SUP 3.7-1*

The applicant added the following supplemental information at the end of VEGP COL FSAR Section 3.7.4.1 to address RG 1.12:

> *Administrative procedures define the maintenance and repair of the seismic instrumentation to keep the maximum number of instruments inservice during plant operation and shutdown in accordance with Regulatory Guide 1.12.*

The NRC staff reviewed the resolution to STD SUP 3.7-1 using the guidance in RG 1.12 and in Appendix S to 10 CFR Part 50. Because of the equivalence of the applicant's proposed resolution to the administrative procedures, maintenance and repair plans of RG 1.12, the staff concludes the applicant has adequately resolved STD SUP 3.7-1.

- *STD SUP 3.7-2*

The applicant added the following supplemental information at the end of VEGP COL FSAR Section 3.7.4.4 to address comparison of measured and predicted responses:

> *Installation and acceptance testing of the triaxial acceleration sensors described in DCD Subsection 3.7.4.2.1 is completed prior to initial startup. Installation and acceptance testing of the time-history analyzer described in DCD Subsection 3.7.4.2.2 is completed prior to initial startup.*

The NRC staff reviewed the resolution to STD SUP 3.7-2, related to the timing of installation and acceptance testing of the triaxial acceleration sensors described in DCD Section 3.7.4.2.1 for the VEGP site. Because of the equivalence of the proposed resolution of STD SUP 3.7-2 to the general operability guidance for seismic equipment addressed in RG 1.12, RG 1.166 and RG 1.167, the staff concludes the applicant adequately resolved STD SUP 3.7-2.

3.7.4.5 *Post Combined License Activities*

There are no post COL activities related to this section.

3.7.4.6 *Conclusion*

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the WLS COL applicant addressed the required information relating to seismic instrumentation, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL application is acceptable and meets the requirements of 10 CFR Part 50, Appendix S and complies with the guidance provided in RG 1.166, RG 1.167, and RG 1.12. The staff based its conclusions on the following:

- STD COL 3.7-2 is acceptable because the applicant is committed to use the procedures endorsed by RG 1.166 and RG 1.167 and because the applicant provided sufficient information to satisfy the requirements 10 CFR Part 50, Appendix S by committing to address the measurement of the post-seismic event gaps between the new fuel rack and walls of the fuel storage pit and to take appropriate corrective actions.

- STD COL 3.7-5 is acceptable because the applicant provided sufficient information to satisfy the requirement 10 CFR Part 50, Appendix S by committing to determining the location of the free-field acceleration sensor and installing the sensor in the protected area.

- STD SUP 3.7-1 is acceptable because the applicant is committed to follow RG 1.12, to include developing administrative procedures to define the maintenance and repairing of the seismic instrumentation to keep the maximum number of instruments in service during plant operation and shutdown.

- STD SUP 3.7-2 is acceptable because the applicant provided sufficient information to satisfy the requirement of 10 CFR Part 50, Appendix S by committing to complete installation and acceptance testing of the seismic instrumentation prior to initial startup.

3.8 Design of Category I Structures

3.8.1 Concrete Containment

This section is not applicable to the WLS design, because AP1000 uses a steel containment.

3.8.2 Steel Containment

The steel containment in the AP1000 DCD provides the following information:

- Description of the containment
- Applicable codes, standard, and specifications
- Loads and load combinations
- Design and analysis procedures
- Structural acceptance criteria
- Materials, quality control, and special construction techniques
- In-service testing (IST) and inspection requirements

Section 3.8 of the WLS COL FSAR, Revision 11, incorporates by reference Section 3.8.2, "Steel Containment," of Revision 19 of the AP1000 DCD. In addition, in the WLS COL FSAR, the applicant provided the following:

Departures

- WLS DEP 6.3-1 and WLS DEP 3.2-1

The applicant provided additional information about WLS DEP 6.3-1 and WLS DEP 3.2-1 in Section 3.8.2 of the FSAR related to design modifications to the condensate return portion of the Passive Core Cooling System and quantifying the duration that the passive residual heat removal heat exchanger can maintain safe shutdown conditions, respectively. This information, as well as related WLS DEP 3.2-1 and WLS DEP 6.3-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.1 of this report.

The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements. Section 21.1 of this report evaluates the departures from the DCD provided in WLS DEP 6.3-1 and WLS DEP 3.2-1.

3.8.3 Concrete and Steel Internal Structures of Steel Containment

Structures inside the containment support the reactor coolant system components and related piping systems and equipment inside the containment and provide radiation shielding. These containment internal structures consist of the primary shield wall, reactor cavity, secondary shield walls, in-containment refueling water storage tank (IRWST), refueling cavity walls, operating floor, intermediate floors, and various platforms, and are not part of the containment pressure boundary.

The containment internal structures are constructed of concrete and structural steel. At the lower elevations, conventional concrete and reinforcing steel are used, except that permanent steel forms are used in some areas in lieu of removable forms based on constructability considerations (modular construction). These steel form modules (liners) consist of two steel plates reinforced with steel angle stiffeners and tee sections. The angles and the tee sections are on the concrete side of the plate. Welded studs, or similar embedded steel elements, are attached to the inside of the steel plates where surface attachments to the plate transfer loads into the concrete. Where these surface attachments are Seismic Category I, the portion of the steel module transferring the load into the concrete is classified as Seismic Category I.

WLS COL FSAR, Revision 11, Section 3.8, incorporates by reference, with no departures or supplements, AP1000 DCD, Revision 19, Section 3.8.3, "Concrete and Steel Internal Structures of Steel Containment." The staff reviewed the application and checked the referenced DCD to ensure that no issues relating to this section remained for review.[1] The staff's review confirmed that there are no outstanding issues related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

AP1000 COL Information Item

- STD COL 3.8-5

In an April 25, 2011, letter, the WLS COL applicant endorsed an October 1, 2010, letter from the VEGP applicant, proposing STD COL 3.8-5 and adding new WLS COL FSAR Sections 3.8.3.7, 3.8.4.7, and 3.8.5.7. The applicant provided information in STD COL 3.8-5 addressing the construction inspection program related to Seismic Category I and II structures. The staff's evaluation of STD COL 3.8-5 is included in Section 3.8.5 of this report.

3.8.4 Other Seismic Category I Structures

3.8.4.1 *Introduction*

The AP1000 DCD defines "other" Seismic Category I structures as the shield building, the auxiliary building, the containment air baffle, Seismic Category I cable tray supports, and Seismic Category I HVAC supports.

The criteria for other Category I structures include the following:

- Description of the structures
- Applicable codes, standards, and specifications
- Loads and load combinations
- Design and analysis procedures
- Structural acceptance criteria
- Materials, quality control, and special construction techniques
- In-service testing (IST) and inspection requirements
- Construction inspection

3.8.4.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 3.8, incorporates by reference Section 3.8.4, "Other Category I Structures," of the AP1000 DCD, Revision 19. In addition, in WLS COL FSAR Section 3.8.4, the applicant provided the following:

Departures

- WLS DEP 3.8-1

The applicant provided a description summary of the site-specific lateral earth pressure on the nuclear island below-grade wall departure from the AP1000 DCD. This departure affects WLS COL FSAR Section 3.8.4.4.4 and WLS COL FSAR Figures 3.8-201a, -202a, -203, and -204.

AP1000 COL Information Item

- STD COL 3.8-5

The applicant provided information in its application regarding STD COL 3.8-5 addressing the construction inspection program related to Seismic Category I and II structures. In an April 25, 2011, letter, the applicant endorsed the October 1, 2010, letter from the VEGP applicant proposing STD COL 3.8-5 and adding new Sections 3.8.3.7, 3.8.4.7, and 3.8.5.7 to the WLS COL FSAR. The staff's evaluation of STD COL 3.8-5 is included in Section 3.8.5.4 of this report.

Supplemental Information

- STD SUP 3.8-1

The applicant provided a supplement to the existing AP1000 DCD Section 3.8.4.4.4, "Below Grade Exterior Walls," under the sub-heading, "Load Combinations." The supplement provides information supporting the WLS DEP 3.8-1 related to the departure of the WLS site-specific lateral earth pressure from the AP1000 DCD.

3.8.4.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. The acceptance criteria associated with the relevant requirements of NRC regulations are given in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 3.8.4.

3.8.4.4 *Technical Evaluation*

The staff reviewed this application and checked the referenced DCD to ensure that the combination of the DCD and the WLS COL application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to other seismic Category I structures. The staff's evaluation of the information incorporated by reference in the WLS COL application is documented in NUREG-1793 and its supplements. Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform a technical review for each "standard issue" and use this review evaluate subsequent COL applications. To ensure that the staff's findings on standard content for the Reference COL application FSAR (i.e., VEGP Units 3 and 4) were equally applicable to the WLS application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In the comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the application, as applicable) resulting from responses to RAIs.
- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed and that those changes were actualized in the COL FSAR.
- The staff verified that site-specific differences, if any, did not adversely affect any previous relevant evaluation or conclusion.

The staff completed its review and concluded that the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this report by use of italicized, double indented formatting.

Departures

- WLS DEP 3.8-1

The staff's evaluation of WLS DEP 3.8-1 is accomplished through the evaluation of the supplemental information section STD SUP 3.8-1. This supplemental information is evaluated below.

Supplemental Information

- STD SUP 3.8-1

STD SUP 3.8-1 addresses WLS DEP 3.8-1 related to the departure of the WLS site-specific lateral earth pressure from the AP1000 DCD. WLS COL FSAR Section 2.5.4.10.3, “Lateral Pressures,” provides a description of the WSL site-specific lateral earth pressures. The departure is the use of a site-specific passive pressure on the below-grade NI walls that is less than the full passive pressure assumed in the analysis of the standard plant. The reason for the departure is that the conservative use of full site-specific passive earth pressure results in an earth pressure exceeding the value used in the standard design. The focus of the staff’s review is the adequacy of the applicant’s justification for the use of partial passive earth pressure and the validity of the approach used to calculate that pressure. WLS COL FSAR Section 2.5.4.10.3, “Lateral Pressures,” provides a description of the WSL site-specific lateral earth pressures. The applicant stated that lateral earth pressures are developed against the below-grade nuclear island walls because of the placement and compaction of granular backfill material and that lateral earth pressures are calculated for active, at-rest, and passive pressure conditions. The applicant further stated that “Westinghouse has evaluated the Lee Nuclear Station site-specific lateral earth pressures and has determined that they are bounded by the standard AP1000 design pressures.”

In RAI 112, Question 02.05.04-20, the staff requested that the applicant provide additional information to confirm that the site-specific lateral earth pressures are bounded by the AP1000 standard design. In an April 10, 2014, response to RAI 112, Question 02.05.04-20, the applicant provided tables that show the comparison of the WLS site-specific lateral earth pressure on the NI’s below grade walls with corresponding pressures that were used in the AP1000 standard design for all the load combinations. The applicant stated that the site-specific lateral pressures on the NI exterior walls below grade are bounded by the AP1000 design pressures for Load Combinations 1 through 6, 8, and 9 in both the east-west (E-W) and north-south (N-S) directions. The applicant noted that the site-specific lateral pressure in Load Combination 7 (LC7), which includes the summation of the full passive lateral earth pressure, the static and dynamic lateral surcharges, and the water pressure for the well-graded gravel (GW) backfill material slightly exceeds the AP1000 LC7 lateral pressure. The applicant stated that the difference in the Load Combination 7 for WLS site compared to the AP1000 generic site is attributed to the site-specific groundwater level, which is 2.4 m (8 ft) below ground surface as compared to the AP1000 groundwater level of 0.6 m (2 ft) below ground surface; a difference of 1.8 m (6 ft) of non-buoyant (heavier) soil resulting in a higher passive earth pressure.

The applicant calculated analytical estimates of the fraction of the full WLS site-specific passive earth pressure that might be mobilized in the GW backfill using a given displacement of the nuclear island assumed as 05 cm (0.2 in.) The applicant included WLS COL FSAR Table 3.8-201 that summarizes the mobilized fraction of the full WLS passive lateral earth pressure for the assumed 0.5 cm (0.2 in) displacement for the range of water-table depths at the WLS site and for the range of values used for the soil modulus and Poisson’s ratio.

AP1000 DCD Section 3.8.5.5.5, “Seismic Stability Analysis,” states that the maximum lateral displacement at the base of the nuclear island when subjected to the CSDRS is expected to be 0.31 cm (0.12 in.) neglecting buoyancy of the nuclear island and 0.48 cm (0.19 in.) considering buoyancy effects. These values are without considering passive resistance from the backfill; therefore, the applicant’s assumed 0.5 cm (0.2 in.) displacement in the analytical estimate is considered conservative. The applicant stated that development of the full passive pressure requires more displacement than the nuclear islands will experience during a seismic event and small lateral displacements such as these are not capable of developing the full passive earth pressure. Therefore, the site-specific nuclear island below-grade wall pressures resulting from the NI FIRS will be less than those used in the standard AP1000 design for this load combination.

The staff reviewed the applicant's April 10, 2014, response to RAI 112, Question 02.05.04-20, and noted that WLS COL FSAR Figures 3.8-203, "William S. Lee Nuclear Station - Mobilized Wall Pressures vs. AP1000 Design Load (E-W)," and 3.8-204, "William S. Lee Nuclear Station - Mobilized Wall Pressures vs. AP1000 Design Load (N-S)," illustrate that the WLS passive pressure in LC7 will not be fully developed and that the pressures on the below-grade walls for the AP1000 standard design continue to bound the pressures that will actually occur at the WLS site. During a May 2014 structural audit, the staff reviewed the, "Lateral Pressure on Nuclear Island Foundation Walls," calculation and found the applicant's approach for estimating the earth pressures to be consistent with the methodology described in its responses to RAI 112, Question RAI 02.05.04-20. For the reasons stated above, the staff finds the applicant's responses to RAI 112, Question RAI 02.05.04-20 acceptable. Therefore, the staff considers STD SUP 3.8-1 and WLS DEP 3.8-1 acceptable because the applicant demonstrated that the passive pressure developed by site-specific conservative maximum displacements is bounded by the AP1000 passive pressure. Accordingly, the staff considers RAI 112, Question 02.05.04-20 resolved.

3.8.4.5 *Post Combined License Activities*

There are no post COL activities related to this section.

3.8.4.6 *Conclusion*

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to other Seismic Category I structures. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements. In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the requirements of GDC 1, 2, 4, and 5 in 10 CFR Part 50, Appendix A.

- WLS DEP 3.8-1 is acceptable because the applicant adequately address the relevant information that meets the guidance in NUREG-0800, Section 3.8.4. In conclusion, the applicant has provided sufficient information to satisfy 10 CFR Part 50, Appendix A, GDC 1, GDC 2, GDC 4, and GDC 5.

- STD SUP 3.8-1 is acceptable because the applicant adequately addressed the relevant information that meets the guidance in NUREG-0800, Section 3.8.4. In conclusion, the applicant has provided sufficient information to satisfy the requirements in10 CFR Part 50, Appendix A, GDC 1, GDC 2, GDC 4, and GDC 5.

3.8.5 Foundations

3.8.5.1 *Introduction*

The NI structures consist of the Containment Building, the Shield Building, and the Auxiliary buildings located on a common 1.8 m (6 ft) thick, cast-in-place, reinforced concrete basemat foundation. Adjoining buildings, such as the Radwaste Building, Turbine Building, and Annex Building, are structurally separated from the NI structures by a 5 cm (2 in.) gap at and below grade. A 10 cm (4 in.) minimum gap is provided above grade. This provides space to prevent interaction between the NI structures and the adjacent structures during a seismic event. This space provides the required factor of safety to accommodate lateral movement under the most stringent loading conditions.

The criteria for the design of foundations include the following:

- Description of the foundations
- Applicable codes, standards, and specifications
- Loads and load combinations
- Design and analysis procedures
- Standard acceptance criteria
- Materials, quality control, and special construction techniques
- In-service testing (IST) and inspection requirements
- Construction inspection

3.8.5.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 3.8 incorporates by reference AP1000 DCD, Revision 19, Section 3.8. AP1000 DCD Section 3.8 includes Section 3.8.5. In addition, in WLS COL FSAR Section 3.8.5, the applicant provided the following:

AP1000 COL Information Items

- WLS COL 2.5-17

The applicant proposed information in WLS COL FSAR Section 3.8.4.7 addressing the type of waterproofing system to be used for the below-grade exterior walls exposed to flood and to groundwater under Seismic Category I structures. In WLS COL FSAR, Section 3.8.4.1, “Description of the Foundations,” the applicant also stated that the site-specific waterproofing approach has not been selected and the applicant will notify the NRC of its selection of system and the qualification testing thereof, on a set schedule.

- STD COL 3.8-5

In an April 25, 2011, letter, the applicant endorsed the August 17, 2010, letter from the VEGP applicant, which proposed STD COL 3.8-5,to replace AP1000 DCD Sections 3.8.3.7, 3.8.4.7, 3.8.5.7, 3.8.6.5. The applicant also provided new WLS COL FSAR sections and revised WLS COL FSAR Section 17.6. The applicant provided information in STD COL 3.8-5 addressing the construction inspection program related to Seismic Category I and II structures.

- STD COL 3.8-6

In an April 25, 2011, letter, the applicant endorsed the October 1, 2010, letter from the VEGP applicant that proposed STD COL 3.8-6 that added a new Section 3.8.6.6 to the WLS COL FSAR. The applicant provided information in STD COL 3.8-6 addressing the construction procedure program related to safety-related Seismic Category I structures. The construction procedures program addresses the pre- and post-concrete placement, and use of construction mock-ups for the steel-concrete composite shield building modules.

Supplemental Information

- STD SUP 3.8-1

The applicant provided supplemental information by adding additional text which states that the depth of overburden and depth of embedment are given in WLS COL FSAR Section 2.5.4.

License Condition

- Part 10, License Condition 6, “Operational Program Readiness,” Item 6.i

The applicant proposed to add “Item 6.i” to proposed License Condition 6 that addresses the availability to of the schedule for the implementation of operational programs, such as construction and inspection procedures, and procedures for concrete placement, use of construction mock-ups, and inspection of module concrete for concrete-filled steel plate modules to the NRC.

ITAAC

The licensee shall satisfy the non-system ITAAC described in WLS COL, Part 10, Appendix B, as new Table 3.3-9, “Waterproof Membrane Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 1 of 1),” requiring a report that documents the as-built waterproofing membrane system beneath the NI basemat has a coefficient of friction to resist sliding of ≥ 0.55, through material qualification testing.

3.8.5.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. The acceptance criteria associated with the relevant requirements of NRC regulations (GDC 1, GDC 2, GDC 4, and GDC 5 in 10 CFR Part 50, Appendix A; 10 CFR 50.55(a), and 10 CFR Part 50, Appendix B) for the foundations are given in NUREG-0800, Section 3.8.5.

3.8.5.4 *Technical Evaluation*

The staff reviewed this application section and checked the referenced DCD to ensure that the combination of the DCD and the WLS COL application represents the complete scope of information relating to this review topic. The staff’s review confirmed that the information in the application and incorporated by reference addresses the required information relating to the foundations. The staff’s evaluation of the information incorporated by reference in the WLS application is documented in NUREG-1793 and its supplements. Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform a technical review for each "standard issue" and use this review to evaluate subsequent COL applications. To ensure that the staff's findings on standard content for the Reference COL application FSAR (i.e., VEGP Units 3 and 4) were equally applicable to the WLS application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In the comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the application, as applicable) resulting from responses to RAIs.
- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed and that those changes were actualized in the WLS COL FSAR.
- The staff verified that site-specific differences, if any, did not adversely affect any previous relevant evaluation or conclusion.

The staff completed its review and concluded that the evaluation performed for the standard content is directly applicable to the WLS COL application. This standard content material is identified in this report by use of italicized, double indented formatting. The staff reviewed the following information in the WLS COL FSAR.

AP1000 COL Information Items

- *WLS COL 2.5-17*

SRP Section 3.8.5 requires confirmation that the nuclear island remains stable under design-basis demands. AP1000 DCD Section 3.4.1.1.1.1, "Waterproofing," states that (1) the waterproof membrane between the mudmat must provide adequate shear strength to transfer horizontal shear caused by seismic loading and (2) the function of the membrane is Seismic Category I. In this regard, AP1000 DCD Section 3.4.1.1.1 provides a requirement for the COL applicant to identify a waterproofing system and to demonstrate a friction coefficient greater than or equal to 0.55 with all horizontal concrete surfaces.

The staff reviewed the WLS COL FSAR and noticed that although there is a proposed ITAAC in WLS COL Part 10, Appendix B, Table 3.3-9, to address the coefficient of friction, there is no description of the selected waterproofing-membrane design. The staff notes that the regulations in 10 CFR 52.79, "Contents of applications; technical information in final safety analysis report," require the FSAR to contain information relative to materials of construction, arrangement, and dimensions sufficient to provide reasonable assurance that the design will conform to the design bases with adequate margin for safety. Therefore, in RAI 102, Question 03.08.05-6, the staff requested that the applicant describe, in an update to WLS COL FSAR Section 3.8.5.1, the proposed waterproofing approach and demonstrate compliance with the AP1000 DCD provisions. In a January 17, 2012, response to RAI 102, Question 03.08.05-6, the applicant stated that the nuclear island mudmat waterproofing system for the WLS nuclear station has not been chosen. However, the applicant confirmed that the waterproofing system selected will be chosen from one of the acceptable alternatives described in AP1000 DCD Section 3.4.1.1.1.1 and will be demonstrated to produce a friction coefficient of 0.55 or greater with the mudmat's horizontal concrete surface.

The staff reviewed the applicant's January 17, 2012, response to RAI 102, Question 03.08.05-6, and noted that the applicant committed to (1) use one of the three waterproofing membrane systems identified in the standard AP1000 design, and (2) demonstrate that the waterproofing membrane meets the waterproofing and friction requirements of greater than or equal to 0.55 as specified in the AP1000 DCD Section 3.4.1.1.1.1. For the reasons stated above, the staff finds the applicant's response to RAI 102, Question 03.08.05-6 acceptable. Accordingly, the staff considers RAI 102, Question 03.08.05-6 resolved.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 3.8.5.4:

- *STD COL 3.8-5*

In a letter dated August 17, 2010, the applicant proposed STD COL 3.8-5, adding a new Section 3.8.3.7, 3.8.4.7, and 3.8.5.7 to the VEGP COL FSAR addressing the construction inspection program related to seismic Category I and II structures. The construction inspection program will be consistent with the maintenance rule (10 CFR 50.65) and guidance in RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," in addressing maintenance requirements for the seismic Category I and seismic Category II structures. The staff concludes that the applicant has provided an acceptable construction inspection program that meets the requirement described in Section 3.8.4.8 of the AP1000 DCD. Therefore, the NRC staff considers STD COL 3.8-5 to be resolved. The incorporation of the planned VEGP COL FSAR changes will be tracked as Confirmatory Item 3.8-2.

<u>Resolution of Standard Content Confirmatory Item 3.8-2</u>

Confirmatory Item 3.8-2 is an applicant commitment to revise its FSAR Table 1.8-202, Table 1.9-201, Appendix 1AA, Section 3.8.3.7, Section 3.8.4.7, Section 3.8.5.7, Section 3.8.6.5, and Section 17.6 to address STD COL 3.8-5. The staff verified that the VEGP COL FSAR was appropriately updated. As a result, Confirmatory Item 3.8-2 is now closed.

- *STD COL 3.8-6*

In a letter dated October 1, 2010, the applicant proposed STD COL 3.8-6, adding a new Section 3.8.6.6 to the VEGP COL FSAR addressing the construction procedure program related to safety-related Category I structures. The construction procedures program addresses the pre- and post-concrete placement, and use of construction mock-ups for the SC modules. The staff concludes that the applicant has provided an acceptable construction procedures program that meets the requirement described in Section 3.8.4.8 of the AP1000 DCD. Therefore, the NRC staff considers STD COL 3.8-6 to be resolved. The incorporation of the planned VEGP COL FSAR changes will be tracked as Confirmatory Item 3.8-3.

<u>Resolution of Standard Content Confirmatory Item 3.8-3</u>

Confirmatory Item 3.8-3 is an applicant commitment to revise its FSAR Table 1.8-202 and Section 3.8.6.6 to address STD COL 3.8-6. The staff verified that the VEGP COL FSAR was appropriately updated. As a result, Confirmatory Item 3.8-3 is now closed.

Supplemental Information

- STD SUP 3.8-1

In WLS COL FSAR Section 3.8.5.1, “Description of the Foundations,” the applicant cited Section 2.5.4, “Stability of Subsurface Materials and Foundations,” which describes the depth of overburden and embedment of the WLS foundation. A foundation is a structural element that connects the superstructure and the supporting medium, such as soils or rocks. The purpose of the foundation is to hold the superstructure in place and to transmit all the loads from the superstructure to the underlying soils or rocks. The NI foundation basemat will be supported by the existing concrete foundation of Cherokee Nuclear Station Unit 1, which is underlain by continuous rock or by fill concrete supported on continuous rock; rock that is fresh to moderately weathered as determined by visual inspection, and is expected to have a Rock Quality Designation (RQD) of at least 65 percent based on the site exploration boring logs.

Sliding and Overturning

WLS COL FSAR Section 2.5.4.5.3.1, “Nuclear Island Foundation Materials,” provides a description of the overturning and sliding stability evaluation of the Seismic Category I and II structures. The applicant stated that the foundation quality rock and fill concrete provide adequate safety margins against bearing-capacity failure for both static and seismic loading of the NI and that only nominal settlement will occur. In its review of WLS COL FSAR Section 2.5.4.5.3.1, the staff noted that the applicant did not provide sufficient information concerning the factor of safety for sliding and overturning for the Seismic Category I structures.

During the May 2014 structural audit held in Cranberry Township, PA, the staff discussed with the applicant the limited information in the WLS COL FSAR related to the factor of safety for sliding and overturning for the WLS NI structures. The staff requested that the applicant provide the sliding and overturning evaluation of the NI structures subject to dynamic loads. The applicant presented the seismic analysis of the WLS NI, including the resultant shear and moment. The sliding and overturning results were conservative when compared to the AP1000 DCD minimum factor of safety values of 1.1 and 1.5 for sliding and overturning, respectively. However, to better understand as well as document the aspects and the results of the stability analysis, the staff issued post-audit RAI 121, Question 03.08.05-7, requesting that the applicant provide additional information that shows a comparison of the WLS NI maximum basemat forces and moments to the AP1000 CSDRS design-basis forces and moments.

In response to RAI 121, Question 03.08.05-7, the applicant provided WLS COL FSAR Table 2-1, “Comparison of Nuclear Island Maximum Absolute Forces and Moments: Lee vs. CSDRS,” which summarized the seismic analysis results. The table shows the maximum reaction forces and moments at the center of gravity of the WLS NI. The applicant stated that the comparison of the WLS and the AP1000 CSDRS maximum NI basement forces and moments was performed based on time-history seismic analyses using the latest NI20u ANSYS model with contact elements at the basemat interface with the rock. The results of the comparison showed that the site-specific WLS sliding and overturning forces and moments are all less than the AP1000 design-basis forces and moments.

In reviewing the applicant’s response, the staff noted that results of the comparison of the AP1000 CSDRS forces and moments enveloped the WLS forces and moments with significant margin. The comparison of the maximum forces and moments in the table shows that the factors of safety for resultant shear and moments are 2.06 and 3.68, respectively. These factors of safety are greater than the corresponding AP1000 minimum factors of safety of 1.1 for sliding and 1.5 for overturning. Moreover, the sliding and overturning analysis performed by the applicant was carried out by models and using methods acceptable to the staff. Based on the information presented above, the staff considers the applicant's response acceptable because the applicant was able to demonstrate that the WLS NI sliding and overturning factors of safety are adequate. Accordingly, the staff considers RAI 121, Question 03.08.05-7 resolved.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 3.8.5.4:

> _License Condition_
>
> - *Part 10, License Condition 6*
>
> *In its letter dated October 1, 2010, the applicant proposed to add another line item to proposed License Condition 6 addressing the availability to NRC inspectors of the schedule for the implementation of construction and inspection procedures related to concrete activities. Specifically, the applicant has proposed to add a new standard item to proposed License Condition 6 to read (where # is the next appropriate letter).*
>
> > *#. the implementation of construction and inspection procedures for concrete filled steel plate modules activities before and after concrete placement, use of construction mock-ups, and inspection of modules before and after concrete placement as discussed in DCD Subsection 3.8.4.8.*
>
> *The applicant's proposed new standard item related to concrete construction and inspection procedures will allow the staff sufficient time to inspect the procedures. Therefore, the staff finds the addition of this line item to the proposed License Condition 6 acceptable.*

3.8.5.5 *Post Combined License Activities*

The license condition language in this section has been clarified from previously considered language. In a letter dated March 22, 2016 (ADAMS Accession No. ML16084A099), the applicant did not identify any concerns with the clarified license condition language. The changes do not affect the staff's above analysis of the conditions, and therefore, for the reasons discussed in the technical evaluation section above, the staff finds the following ITAAC and license condition acceptable:

- The licensee shall perform and satisfy the non-system ITAAC described in WLS Combined License Application (COLA), Part 10, Appendix B, Table 3.3-9, "Waterproof Membrane Inspections, Tests, Analyses, and Acceptance Criteria (Sheet 1 of 1)."

- License Condition (3-4) – No later than 12 months after issuance of the COL, the licensee shall submit to the Director of NRO, or the Director's designee, a schedule for implementation of the construction and inspection procedures for steel concrete composite (SC) construction activities for seismic Category I nuclear island modules. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until each this license condition has been fully implemented. The schedule shall identify the completion of or implementation of the construction and inspection procedures for steel concrete composite (SC) construction activities for seismic Category I nuclear island modules (including shield building SC modules) described in AP1000 DCD Rev. 19, Section 3.8.4.8.

3.8.5.6 *Conclusion*

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to foundations and that no outstanding information is expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the requirements of GDC 1, GDC 2, GDC 4, and GDC 5 in 10 CFR Part 50, Appendix A. The staff based its conclusion on the following:

- STD SUP 3.8-1 is acceptable because the applicant adequately provided the relevant information that meets the guidance in NUREG-0800, Section 3.8.5 and specifically addresses WLS COL 2.5-6; WLS COL 2.5-7; WLS COL 2.5-10; WLS COL 2.5-12; and the sliding and overturning evaluation of the Seismic Category I and II structures. In conclusion, the applicant provided sufficient information to satisfy 10 CFR Part 50, Appendix A, GDC 1 GDC 2, GDC 4, and GDC 5.

- WLS COL 2.5-17 is acceptable because the applicant committed to (1) use one of the three waterproofing-membrane systems identified in AP1000 DCD Section 3.4.1.1.1.1 standard design and that were reviewed and accepted by the staff; and (2) demonstrate that the waterproofing membrane meets the waterproofing and friction requirements of greater than or equal to 0.55 as specified in the AP1000 DCD, Section 3.4.1.1.1.1.

3.9 Mechanical Systems and Components

Structural integrity and functional capability of various safety-related mechanical components are described. The design is not limited to ASME Code components and supports, but is extended to other components such as control rod drive mechanisms (CRDMs), certain reactor internals, and any safety-related piping designed to industry standards other than the ASME Code. The design includes issues such as load combinations, allowable stresses, methods of analysis, summary of results, and preoperational testing. The evaluation of this section is focused on determining whether there is adequate assurance of a mechanical component performing its safety-related function under all postulated combinations of normal operating conditions, system operating transients, postulated pipe breaks, and seismic events.

3.9.1 Special Topics for Mechanical Components

In WLS COL FSAR Section 3.9.1, "Special Topics for Mechanical Components," design transients and methods of analysis are described for all Seismic Category I components, component supports, core support (CS) structures, and reactor internals designated as Class 1, 2, 3, and CS under ASME Code, Section III, and those not covered by the ASME Code. Also included are the assumptions and procedures used for the inclusion of transients in the design and fatigue evaluation of ASME Code Class 1 and CS components and the computer programs used in the design and analysis of Seismic Category I components and their supports, as well as experimental and inelastic analytical techniques.

WLS COL FSAR, Revision 11, Section 3.9 incorporates by reference, with no departures or supplements, AP1000 DCD, Revision 19, Section 3.9.1, "Special Topics for Mechanical Components." The staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The staff's review confirmed that there is no outstanding issue related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

3.9.2 Dynamic Testing and Analysis of Systems, Structures and Components

The criteria, testing procedures, and dynamic analyses employed to ensure the structural and functional integrity of piping systems, mechanical equipment, reactor internals, and their supports (including supports for conduit and cable trays, and ventilation ducts) under vibratory loadings, are addressed in this section. The loadings include those due to fluid flow (and especially loading caused by adverse flow conditions, such as flow instabilities over standoff pipes and branch lines in the steam system) and postulated seismic events.

WLS COL FSAR, Revision 11, Section 3.9, incorporates by reference, with no departures or supplements, AP1000 DCD, Revision 19, Section 3.9.2, "Dynamic Testing and Analysis of Systems, Structures and Components." The staff reviewed the WLS COL application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] Specific to departure WLS DEP 2.0-1, which was not referenced in Section 3.9.2 of the WLS COL FSAR but is addressed further in other sections in this chapter, FSAR Reference 206 shows that the reactor vessel and internals were chosen for evaluation as representative of major equipment. From the analyses performed, the CSDRS was found to have higher loads and stresses than those from the WLS site-specific seismic response. Hence, the WLS site-specific seismic response does not impact the structural integrity of the reactor vessel and internals. The staff's review confirmed that there is no outstanding issue related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

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

3.9.3.1 *Introduction*

The structural integrity and functional capability of pressure-retaining components, their supports, and CS structures are ensured by designing them in accordance with ASME Code, Section III, or other industrial standards. The loading combinations and their respective stress limits, the design and installation of pressure-relief devices, and the design and structural integrity of ASME Code Class 1, 2, and 3 components and component supports are included.

3.9.3.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 3.9, incorporates by reference AP1000 DCD, Revision 19, Section 3.9. In addition, in WLS COL FSAR Section 3.9.3, the applicant provided the following:

Departures

- WLS DEP 6.4-2

The applicant provided additional information in Table 3.9-202 of the WLS COL FSAR about WLS DEP 6.4-2 related to design changes affecting how the temperature and humidity in the main control room are maintained within the limits for reliable human performance. This information, as well as related WLS DEP 6.4-2 information appearing in other chapters of the FSAR, is reviewed in Section 21.3 of this SER.

AP1000 COL Information Items

- STD COL 3.9-2

The applicant provided information in WLS COL FSAR Section 3.9.8.2, "Design Specifications and Reports," for STD COL 3.9-2 to address COL Information Item 3.9-2, which states that "Reconciliation of the as-built piping (verification of the thermal cycling and stratification loadings considered in the stress analysis discussed in [DCD] Section 3.9.3.1.2) is completed by the COL holder after the construction of the piping systems and prior to fuel load." Evaluation of this particular COL Information Item is provided in Section 3.12 of this report.

- STD COL 3.9-3

The applicant provided information in WLS COL FSAR Sections 3.9.3.4.4, "Inspection, Testing, Repair, and/or Replacement of Snubbers," and 3.9.8.3, "Snubber Operability Testing," for STD COL 3.9-3 to address COL Information Item 3.9-3, which describes snubber design and testing, snubber installation requirements, and snubber preservice and inservice examination and testing.

- STD COL 3.9-5

The applicant provided information in WLS COL FSAR Sections 3.9.3.1.2, "Loads for Class 1 Components, Core Support, and Component Supports," and 3.9.8.5, "Surge Line Thermal Monitoring," for STD COL 3.9-5 to address COL Information Item 3.9-5, that addresses pressurizer surge line monitoring. Evaluation of this particular COL information item is provided in Section 3.12 of this report.

- STD COL 3.9-7

In a June 21, 2011, letter, the applicant endorsed the April 23, 2010, letter from the VEGP applicant that proposed to add STD COL 3.9-7 to the WLS COL FSAR. The applicant provided information in WLS COL FSAR Section 3.9.8.7, "As-Designed Piping Analysis," to address STD COL 3.9-7. This COL item provides additional information on the process to be used to complete the piping design and to complete the ITAAC added to verify the design. Evaluation of this particular COL information item is provided in Section 3.12 of this report.

3.9.3.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. The acceptance criteria associated with the relevant requirements of NRC regulations for the ASME Code Class 1, 2, and 3 components, component supports, and CS structures are given in NUREG-0800, Section 3.9.3.

3.9.3.4 *Technical Evaluation*

The staff reviewed this application section and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic. The staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the ASME Code Class 1, 2, and 3 components, component supports, and core support structures. The staff's evaluation of the information incorporated by reference in the WLS application is documented in NUREG-1793 and its supplements. Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform a technical review for each "standard issue" and use this review to evaluate subsequent COL applications. To ensure that the staff's findings on standard content for the Reference COL application FSAR (i.e., VEGP Units 3 and 4) were equally applicable to the WLS application, the staff undertook the following reviews.

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In the comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the application, as applicable) resulting from responses to RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed and that those changes were actualized in the WLS COL FSAR.

- The staff verified that site-specific differences, if any, did not adversely affect any previous relevant evaluation or conclusion.

The staff completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this report by use of italicized, double indented formatting. Section 1.2.3 of this report provides an explanation of why the standard content material from the SER for the Reference COL application (VEGP) contains evaluation material from the SER for the BLN, Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from Section 3.9.3.4 of the VEGP SER:

> <u>*AP1000 COL Information Items*</u>
>
> - *STD COL 3.9-3*
>
> *AP1000 DCD, Section 3.9.8.3, "Snubber Operability Testing," states that COL applicants referencing the AP1000 design will develop a program to verify operability of essential snubbers as outlined in Section 3.9.3.4.3, "Snubbers Used as Component and Piping Supports," and Section 3.9.3.4.4, "Inspection,*

Testing, Repair and/or Replacement of Snubbers." In the BLN COL FSAR, the applicant states in Section 3.9.8.3, "Snubber Operability Testing," that STD COL 3.9-3 is addressed in BLN COL FSAR Section 3.9.3.4.4, which incorporates by reference AP1000 DCD Section 3.9.3.4.4, with supplemental snubber information added to the end of the existing Section 3.9.3.4.4.

As indicated in the BLN COL FSAR, STD COL 3.9-3 contains a wide range of supplemental information on snubber design and testing requirements, snubber installation requirements, and snubber preservice and inservice examination and testing. It was not clear to the staff, however, whether STD COL 3.9-3 had provided the required information called for by AP1000 DCD, Section 3.9.8.3. In RAI 3.9.3-1, the staff requested that the applicant address the following: (1) clarify what was meant by "snubber operability testing" when the applicant prepared the COL information; (2) discuss whether the entire STD COL 3.9-3 represents BLN's plant-specific, updated snubber requirements, not already covered in AP1000 DCD, Section 3.9.3; (3) clarify whether all or part of STD COL 3.9-3 is related to snubber operability testing; (4) for the portions of STD COL 3.9-3 which are not related to snubber operability testing, explain why they are included as part of the COL item; (5) discuss all the pertinent codes and standards on which STD COL 3.9-3 is based to assure snubber operability; and (6) discuss the need to modify the content and the physical placement of STD COL 3.9-3 in the BLN COL FSAR.

In its response, the applicant explained that information presented in BLN COL FSAR Section 3.9.3.4.4 regarding snubber testing includes information specific to qualification and installation tests and examinations for snubbers included in the inservice testing (IST) program and preservice examination and testing programs; and information specifically related to snubber inservice examination and testing. The applicant acknowledges, therefore, that not all information added by STD COL 3.9-3 is related specifically to snubber "operability testing." The applicant also noted that BLN COL FSAR Section 3.9.3.4.4 has been subjected to a revision responding to a separate staff RAI on snubber IST programs. Details of the applicant's responses to the RAI are provided in the following:

(1) For the purpose of STD COL 3.9-3, operability testing encompasses the preservice and inservice examinations and testing required by the ASME Code for Operation and Maintenance (OM) for Nuclear Power Plants (ASME OM Code), Subsection ISTD, "Preservice and Inservice Examination and Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Nuclear Power Plants" as described in BLN COL FSAR Section 3.9.3.4.4.c and Section 3.9.3.4.4.d (as revised in applicant's response to RAI 3.9.6-3).

(2) In order to provide a complete description of the snubber operability testing program, that is, the preservice and IST programs for snubbers, additional information was provided in BLN COL FSAR Section 3.9.3.4.4 as indicated in the applicant's letter to the NRC in response to RAI 3.9.6-3. Previously, only snubber preservice examination and testing had been described in BLN COL FSAR Section 3.9.3.4.4.c.

(3) As noted above, some of the information provided in the original BLN COL FSAR Section 3.9.3.4.4 relates to snubber qualification testing and examinations and snubber installation verification requirements. These activities are considered precursors to the snubber operability testing that will be conducted in accordance with the ASME OM Code, Subsection ISTD.

(4) The information not specifically related to STD COL 3.9-3 operability testing, i.e., Sections 3.9.3.4.4.a and 3.9.3.4.4.b, should have been labeled as standard supplemental information, using the left margin annotation STD SUP 3.9-3.

(5) Snubber operability testing is to be conducted during implementation of the preservice and ISI and testing programs in accordance with the requirements of the ASME OM Code, Subsection ISTD. As indicated in the first paragraph of BLN COL FSAR Section 3.9.3.4.4, the description of the program provided in the BLN COL FSAR is based on the 2001 Edition through the 2003 Addenda of the ASME OM Code. However, the initial IST program for snubbers will incorporate the latest Edition and Addenda of the ASME OM Code approved in 10 CFR 50.55a(f) on the date 12 months before initial fuel load.

(6) BLN COL FSAR Section 3.9.3.4.4 will be revised as indicated in the Application Revision section of this response to segregate the snubber operability testing from the remaining portions of the section (i.e., the snubber design and qualification testing, and the snubber installation requirements) and to include the appropriate left margin annotation. In addition, to maintain consistency, to the extent possible, with other industry COL applications, Section 3.9.3.4.4.a is revised to clarify and expand on snubber qualification examination and testing. Finally, minor editorial changes are made to the Section 3.9.3.4.4.c changes provided in the applicant's letter to the NRC in response to RAI 3.9.6-3. Additionally, changes will be made to the introductory (roadmap) paragraph for BLN COL FSAR Section 3.9.3.4.4 indicating it is a new subsection to follow DCD Section 3.9.3.4.3.

The staff found that above responses provided by the applicant to be adequate in clarifying that the information for snubber operability testing originally provided in STD COL 3.9-3 was primarily intended for preservice and inservice examination and testing. The staff also found that the supplemental information provided under a new STD SUP 3.9-3, for snubber design and qualification testing, and the snubber installation requirements includes a better description for snubber design and qualification testing, and is more consistent with other industry COL applications. The staff confirmed that Revision 1 has incorporated all the changes as required. RAI 3.9.3-1 is closed.

Clarification of BLN SER Standard Content

Based on the staff's review of the standard content, there were two minor changes of an editorial nature that were found not to affect the staff's conclusion.

The first paragraph discussed in Item (5) above was moved in the final VEGP COL FSAR such that it is appropriately included with the write up specific to STD COL 3.9-3. The introductory (roadmap) paragraph was not changed as described following Item (6) above because the AP1000 DCD was modified to include a paragraph numbered "3.9.3.4.4." As a result, the new text was added to an existing section as opposed to being a standalone section.

Resolution of Difference Between FSARs

In Section 3.9.3.4.4 of the BLN COL FSAR, the BLN applicant stated that a list of snubbers on systems which experience sufficient thermal movement to measure cold to hot position, is included as part of the testing program after piping analysis has been completed. In Section 3.9.3 of the VEGP COL FSAR, the VEGP applicant provides Table 3.9-201 with this list of snubbers. The addition of a list of snubbers on systems which experience sufficient thermal movement to measure cold to hot position to the VEGP COL FSAR is acceptable to the staff.

3.9.3.5 *Post Combined License Activities*

There are no post COL activities related to this section.

3.9.3.6 *Conclusion*

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to ASME Code Class 1, 2, and 3, components, component supports and CS structures, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements. In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the requirements of 10 CFR Part 52, "Licenses, certifications, and approvals for nuclear power plants." The staff based its conclusion on the following:

- WLS DEP 6.4-2, related to design changes affecting how the temperature and humidity in the main control room are maintained within the limits for reliable human performance, is reviewed and found acceptable by the staff in Section 21.3 of this SER.

- STD COL 3.9-3 is acceptable because the applicant addressed the relevant information that meets the guidance in NUREG-0800, Section 3.9.3. In conclusion, the applicant provided sufficient information to satisfy the requirements in 10 CFR Part 50, Appendix A, GDC 1 and GDC 4.

3.9.4 Control Rod Drive System

The control rod drive system (CRDS) consists of the control rods and the related mechanical components that provide the means for mechanical movement. As discussed in GDC 26, "Reactivity Control System Redundancy and Capability" and GDC 27, "Combined Reactivity Control Systems Capability," the CRDS provides one of the independent reactivity control systems. The rods and the drive mechanism are capable of reliably controlling reactivity changes either under conditions of anticipated operational occurrences, or under postulated accident conditions. A positive means for inserting the rods is always maintained to ensure appropriate margin for malfunction, such as stuck rods. Since the CRDS is a safety-related system and portions of the CRDS are a part of the RCPB, the system is designed, fabricated, and tested to quality standards commensurate with the safety-related functions to be performed. This provides an extremely high probability of accomplishing the safety-related functions either in the event of anticipated operational occurrences or in withstanding the effects of postulated accidents and natural phenomena such as earthquakes, as discussed in GDC 1; GDC 2; GDC 14, “Reactor Coolant Pressure Boundary;” GDC 29 “Protection Against Anticipated Operational Occurrences;” and 10 CFR 50.55a.

WLS COL FSAR, Revision 11, Section 3.9, incorporates by reference, with no departures or supplements, AP1000 DCD, Revision 19, Section 3.9.4, “Control Rod Drive System (CRDS).” The staff reviewed the WLS COL application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The staff’s review confirmed that there is no outstanding issue related to this section. The results of the staff’s technical evaluation of the information incorporated by reference are documented in NUREG-1793 and its supplements.

3.9.5 Reactor Pressure Vessel Internals

AP1000 reactor internals consist of two major assemblies - the lower internals and the upper internals. The reactor internals provide protection, alignment and support for the core. Control rods and gray rods provide safe and reliable reactor operation. In addition, the reactor internals help to accomplish the following: direct the main coolant flow to and from the fuel assemblies; absorb control rod dynamic loads, fuel assembly loads, and other loads and transmit these loads to the reactor vessel; support instrumentation within the reactor vessel; provide protection for the reactor vessel against excessive radiation exposure from the core; and position and support reactor vessel radiation surveillance specimens.

WLS COL FSAR, Revision 11, Section 3.9, incorporates by reference, with no departures or supplements, AP1000 DCD, Revision 19, Section 3.9.5, “Reactor Pressure Vessel Internals.” The staff reviewed the WLS COL application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The staff’s review confirmed that there is no outstanding issue related to this section. The results of the staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

3.9.6 Inservice Testing of Pumps and Valves

3.9.6.1 *Introduction*

In this section, the staff describes its review of the functional design, qualification, and inservice testing (IST) programs for pumps, valves, and dynamic restraints as required by the NRC regulations in 10 CFR Part 52 and 10 CFR 50.55a, “Conditions of Construction Permits, Early Site Permits, Combined Licenses, and Manufacturing Licenses,” for WLS. RG 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition),” discusses the Commission’s position provided in SECY-05-0197, “Review of Operational Programs in a Combined License Application and General Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria,” that operational programs should be fully described in COL applications to avoid the need to specify ITAAC for those programs. The applicant relies on the WLS COL FSAR, with its incorporation by reference of the AP1000 DCD and supplemental information, to fully describe the IST and motor-operated valve (MOV) testing operational programs in support of the COL application for WLS.

3.9.6.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 3.9, incorporates by reference AP1000 DCD, Revision 19, Section 3.9. In addition, in WLS COL FSAR Section 3.9.6, the applicant provided the following

Departures

- WLS DEP 6.4-2

The applicant provided additional information in Table 3.9-203 of the WLS COL FSAR about WLS DEP 6.4-2 related to design changes affecting how the temperature and humidity in the main control room are maintained within the limits for reliable human performance. This information, as well as related WLS DEP 6.4-2 information appearing in other chapters of the FSAR, is reviewed in Section 21.3 of this SER.

AP1000 COL Information Item

- STD COL 3.9-4

The applicant provided information in several sections of WLS COL FSAR Section 3.9.6 in response to STD COL 3.9-4 to supplement the AP1000 DCD provisions to fully describe the IST and MOV testing programs for WLS. For example, the WLS COL FSAR supplements the provisions in the AP1000 DCD with respect to the Edition and Addenda of the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) applicable to the description of the IST program for WLS, determination of the MOV testing frequency, operability testing of power-operated valves (POVs) other than MOVs, performance of check valve exercise tests, and plans to apply alternatives to the ASME OM Code.

The AP1000 DCD addresses the functional design and qualification of mechanical equipment to be used at an AP1000 nuclear power plant in several AP1000 DCD sections. For example, AP1000 DCD Section 3.9.3.2, “Pump and Valve Operability Assurance,” states that criteria are developed to assess the functional capability of required components to operate. AP1000 DCD Section 3.9.3.2.2, “Valve Operability,” indicates that operational tests will be performed to verify that valves open and close prior to installation. This section also specifies cold hydro tests, hot functional tests, periodic inservice inspections (ISIs), and periodic inservice operations to be performed in situ to verify the functional capability of the valves. AP1000 DCD Section 5.4.8, “Valves,” includes provisions regarding design and qualification, and preoperational testing of valves within the scope of those systems, and refers to these activities for other safety-related valves. AP1000 DCD Section 5.4.8.3, “Design Evaluations,” specifies that the requirements for qualification testing of power-operated active valves are based on ASME Standard QME-1-2007, “Qualification of Active Mechanical Equipment Used in Nuclear Power Plants.” AP1000 DCD Section 5.4.9, “Reactor Coolant System Pressure Relief Devices,” includes provisions for design, testing, and inspection of relief devices in the reactor coolant system. AP1000 DCD Section 5.4.10, “Component Supports,” includes provisions for design, testing, and inspection of component supports in the reactor coolant system. The WLS COL FSAR incorporates by reference these specific sections in the AP1000 DCD.

With respect to flow-induced vibration (FIV) of plant components, AP1000 DCD Section 3.9.2, "Dynamic Testing and Analysis," describes tests to confirm that piping, components, restraints, and supports have been designed to withstand the dynamic effects of steady-state FIV and anticipated operational transient conditions. AP1000 DCD Section 14.2.9.1.7, "Expansion, Vibration and Dynamic Effects Testing," states that the purpose of the expansion, vibration and dynamic effects testing is to verify that the safety-related, high-energy piping and components are properly installed and supported such that, in addition to other factors, vibrations caused by steady-state or dynamic effects do not result in excessive stress or fatigue to safety-related plant systems. The WLS COL FSAR incorporates by reference these sections in the AP1000 DCD.

AP1000 DCD, Section 3.9.3.4.4, "Inspection, Testing, Repair, and/or Replacement of Snubbers," specifies that a program for inservice examination and testing of dynamic supports (snubbers) will be prepared in accordance with the requirements of the ASME OM Code, Subsection ISTD, "Preservice and Inservice Examination and Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Nuclear Power Plants." AP1000 DCD Section 3.9.3.4.4 indicates that details of the snubber inservice examination and testing program, including test schedules and frequencies, will be reported in the ISI and testing plan included in the IST program required by AP1000 DCD Section 3.9.8.3, "Snubber Operability Testing." AP1000 DCD Section 3.9.8.3 states that COL applicants referencing the AP1000 design will develop a program to verify operability of essential snubbers. The WLS COL FSAR provides supplemental information for AP1000 DCD Section 3.9.3.4.4 regarding snubbers. For example, WLS COL FSAR Section 3.9.3.4.4, includes provisions for snubber design and testing with specifications that snubber qualification and production testing will satisfy the applicable sections of the ASME Boiler and Pressure Vessel Code (BPV Code); the ASME OM Code; and ASME Standard QME-1-2007. WLS COL FSAR Section 3.9.3.4.4 also describes the inservice examination and testing of safety-related snubbers in accordance with the requirements of the ASME OM Code, Subsection ISTD. The description includes specifications for initial and subsequent examination intervals, visual examination attributes, IST methods and intervals, establishment of snubber test groups, response to examination and test results, snubber repair and replacement, post-maintenance examination and testing, and establishment and monitoring of snubber service life. WLS COL FSAR Table 3.9-201, "Safety Related Snubbers," provides a list of safety-related snubbers to be installed, including the snubber identification number and the associated system or component.

AP1000 DCD Section 3.9.6, "Inservice Testing of Pumps and Valves," provides a general description of the IST Program to be developed for AP1000 reactors. AP1000 DCD Table 3.9-16, "Valve Inservice Test Requirements," lists valves within the scope of the IST program provided in support of the AP1000 DC, and indicates the valve tag number, valve and actuator type, safety-related missions, safety functions, ASME Code class and IST category, and IST type and frequency. WLS COL FSAR Section 3.9.6 incorporates by reference AP1000 DCD Section 3.9.6 with supplemental information in several areas. For example, the applicant states that the description of the IST program for WLS is based on the ASME OM Code, 2001 Edition through 2003 Addenda. The applicant also indicates that the initial IST program will incorporate the latest edition and addenda of the ASME OM Code approved in 10 CFR 50.55a(f) on the date 12 months before initial fuel load. In the WLS COL FSAR, the applicant described the periodic testing program for POVs other than MOVs that incorporates lessons learned based on nuclear power plant operating experience and research programs for MOV performance. The applicant also indicated its plan to apply Revision 1 to ASME OM Code Case OMN-1, "Alternative Rules for the Preservice and Inservice Testing of

Certain Electric Motor-Operated Valve Assemblies in Light Water Reactor Power Plants," as an alternative to the quarterly MOV stroke-time testing provisions in the ASME OM Code, and to satisfy the supplemental requirements specified in 10 CFR 50.55a(b)(3)(ii) to ensure that MOVs continue to be capable of performing their design-basis safety functions. The WLS COL FSAR does not identify any additional plant-specific valves to be included in the IST program beyond those listed in AP1000 DCD Table 3.9-16.

License Conditions

- Part 10, License Condition 3, "Operational Program Implementation," Items G.2 and G.5

The applicant proposed a license condition providing the implementation milestones for the Preservice Testing Program and MOV Testing Program.

- Part 10, License Condition 6, "Operational Program Readiness," Item 6.d

The applicant proposed a license condition to provide a schedule to support the NRC inspection of operational programs, including the Preservice Testing Program and MOV Testing Program.

3.9.6.3 *Regulatory Basis*

The regulatory basis of the design related information incorporated by reference is addressed in NUREG-1793 and its supplements. The regulatory basis for the staff's review of the WLS COL FSAR is provided by 10 CFR Parts 50 and 52. Specifically, the NRC regulations in 10 CFR 52.79(a) require that the COL application include information at a level sufficient to enable the Commission to reach a final conclusion on all safety matters that must be resolved by the Commission before COL issuance. For example, paragraph (4) in 10 CFR 52.79(a) requires that a COL application include the design of the facility with specific reference to the GDC in 10 CFR Part 50, Appendix A which establish the necessary design, fabrication, construction, testing, and performance requirements for SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. 10 CFR 52.79(a), Paragraph (11) requires that a COL application provide a description of the programs and their implementation necessary to ensure that the systems and components meet the requirements of the ASME BPV Code and the ASME OM Code in accordance with 10 CFR 50.55a. 10 CFR 52.79(a), Paragraph (29)(i) requires that a COL application provide plans for conduct of normal operations, including maintenance, surveillance, and periodic testing of SSCs. 10 CFR 52.79(a), Paragraph (37) requires that a COL application provide the information necessary to demonstrate how operating experience insights have been incorporated into the plant design.

RG 1.206 provides guidance for a COL applicant in preparing and submitting its COL application in accordance with NRC regulations. For example, RG 1.206, Section C.IV.4 discusses the requirement in 10 CFR 52.79(a) for descriptions of operational programs that need to be included in the FSAR for a COL application to allow a reasonable assurance finding of acceptability. In particular, a COL applicant should fully describe the IST, MOV testing, and other operational programs as defined in Commission Paper SECY-05-0197 to avoid the need for ITAAC for the implementation of those programs. The term "fully described" for an operational program should be understood to mean that the program is clearly and sufficiently described in terms for scope and level of detail to allow a reasonable assurance finding of acceptability. Further, operational programs should be described at a functional level and an increasing level of detail where implementation choices could materially and negatively affect the program effectiveness and acceptability. The Commission approved the use of a license condition for operational program implementation milestones that are fully described or referenced in the FSAR as discussed in the SRM for SECY-05-0197, February 22, 2006.

The staff followed NUREG-0800, Section 3.9.6, “Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints,” in its review of the WLS COL application. The staff also evaluated the WLS COL FSAR information against the guidance provided in RG 1.206. Appendix 1AA, “Conformance with Regulatory Guides,” and confirms that the COL application conforms to RG 1.206 without exceptions related to the IST program. In addition, WLS COL FSAR Table 1.9-202, “Conformance with SRP Acceptance Criteria,” conforms to NUREG-0800, Section 3.9.6.

3.9.6.4 *Technical Evaluation*

The staff reviewed WLS COL FSAR Section 3.9.6 and checked the referenced DCD to ensure that the combination of the DCD and the WLS COL application represents the complete scope of information relating to this review topic.[1] The staff’s review confirmed that the information in the application and incorporated by reference addresses the required information relating to functional design, qualification and IST programs for pumps, valves, and dynamic restraints. The results of the staff’s evaluation of the design-related information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements. The results of the staff’s review of the material in the AP1000 DCD related to the IST operational program for pumps, valves, and dynamic restraints are in this section of the report.

Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform a technical review for each “standard issue” and use this review to evaluate subsequent COL applications. To ensure that the staff’s findings on standard content that were documented in the SER for the reference COL application (i.e., VEGP Units 3 and 4) were equally applicable to the WLS COL application, the staff undertook the following reviews.

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from responses to RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed and that those changes were included in the WLS COL FSAR.

- The staff verified that site-specific differences did not adversely affect any previous relevant evaluation or conclusion.

The staff completed its review and concluded that the evaluation performed for the standard content is directly applicable to the WLS COL application. This standard content material is identified in this report by use of italicized, double-indented formatting. Section 1.2.3 of this report provides an explanation of why the standard content material from the SER for the Reference COL application (i.e., VEGP) contains evaluation material from the SER for the BLN, Units 3 and 4 COL application. The staff reviewed the following information in the WLS COL FSAR:

The following portion of this technical evaluation section is reproduced from VEGP SER Section 3.9.6.4:

> *In its letter dated December 17, 2008, Southern Nuclear Operating Company (SNC) listed the RAIs prepared by the NRC staff on the BLN Units 3 and 4 COL application. In that letter, SNC endorsed the responses, including proposed changes to the FSAR, submitted by the Tennessee Valley Authority (TVA) on 16 RAIs related to the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints as applicable to the VEGP COL application. In letters dated December 14, 2009, and January 12, March 1, and May 14, 2010, SNC described its plans to resolve open items identified in the "SER with open items on the standard content information" prepared by the NRC staff on the description of the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints in the BLN Units 3 and 4 COL application. The NRC staff has reviewed the SNC letters and Revision 2 to the VEGP COL FSAR to determine whether the description of the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints in the VEGP COL application with its incorporation by reference of the AP1000 DCD meets the regulatory requirements to provide reasonable assurance that those components at VEGP will be capable of performing their safety functions if these programs are developed and implemented consistent with the description in the VEGP COL FSAR and AP1000 DCD.*
>
> *The staff reviewed the information in the VEGP COL FSAR, and the staff's review of the standard content open item is provided:*
>
> *<u>AP1000 COL Information Item</u>*
>
> - *STD COL 3.9-4*
>
> *The NRC staff reviewed STD COL 3.9-4 related to COL Information Item 3.9-4 included in AP1000 DCD Tier 2, Section 3.9.8.4. COL Information Item 3.9-4 states:*
>
>> *Combined License applicants referencing the AP1000 design will develop an inservice test program in conformance with the valve inservice test requirements outlined in subsection 3.9.6 and Table 3.9-16. For power-actuated valves, the requirements for operability testing shall be based on subsection 3.9.6.2.2. This program will include provisions for nonintrusive check valve testing methods and the program for valve disassembly and inspection outlined in subsection 3.9.6.2.3. The Combined License applicant will complete an evaluation as identified in subsection 3.9.6.2.2 to determine the frequency of power-operated valve operability testing.*
>
> *The information item for COL applicants to develop an IST Program was specified as COL Action Item 3.9.6.4-1 in Appendix F of NUREG-1793, which states:*

> *The COL applicant will provide an inservice test (IST) program that complies with the inservice testing requirements for valves.*

In STD COL 3.9-4, the applicant states that this COL item is addressed in Sections 3.9.6, 3.9.6.2.2, 3.9.6.2.3, 3.9.6.2.4, 3.9.6.2.5, and 3.9.6.3 for the VEGP COL application.

In this section of the SER, the NRC staff describes its review of the VEGP COL FSAR with the incorporation by reference of the AP1000 DCD for an acceptable description of the functional design, qualification, and IST programs, including the MOV Testing Program, for VEGP Units 3 and 4 to provide reasonable assurance that the safety-related components within the scope of the VEGP IST Program will be capable of performing their safety functions in accordance with the NRC regulations and the ASME Code requirements.

AP1000 DCD Tier 2, Section 3.9.6.1, "Inservice Testing of Pumps," specifies that the AP1000 reactor design does not include pumps with safety functions with the exception of the coastdown of the reactor coolant pumps. As determined in NUREG-1793, the NRC staff considers the IST Program scope for the AP1000 design with respect to pumps to be acceptable. Therefore, the NRC staff did not include pumps in the review of the IST Program for safety-related components at VEGP Units 3 and 4.

VEGP COL FSAR Section 3.9.6 states that the description of the IST Program for VEGP Units 3 and 4 is based on the ASME OM Code, 2001 Edition through 2003 Addenda, and that the limitations and modifications set forth in 10 CFR 50.55a will be incorporated. The NRC regulations in 10 CFR 50.55a incorporate by reference the ASME OM Code, 2001 Edition through 2003 Addenda, with certain limitations and modifications. Therefore, the NRC staff considers the application of the ASME OM Code, 2001 Edition through 2003 Addenda, as incorporated by reference in the NRC regulations with applicable limitations and modifications, to be acceptable for the VEGP IST Program description in support of the VEGP COL application. As specified in 10 CFR 50.55a, a COL licensee is required to incorporate in its IST Program the latest Edition and Addenda of the ASME OM Code approved in 10 CFR 50.55a(f) on the date 12 months before initial fuel load.

The VEGP COL FSAR incorporates by reference AP1000 DCD Tier 2, Table 3.9-16, "Valve Inservice Test Requirements," that includes the valve type, safety-related missions, safety functions, the ASME Code IST category, and IST type and frequency. The NRC staff considers this table to be sufficient in describing the IST Program in support of the VEGP COL application. Following the issuance of the VEGP COL, the guidance in NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants," can be used to develop the VEGP IST Program, including the specific information to be included in the IST Program documentation and tables for NRC inspection.

On March 26 and 27, 2008, the NRC staff held a public meeting to discuss the NRC's review of the description of the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints in COL applications *referencing the AP1000 certified design and the AP1000 DC amendment application. At the public meeting, Westinghouse stated that it would make information available on the functional design and qualification of safety-related valves and dynamic restraints within the scope of the AP1000 DCD in design and procurement specifications that will be applicable to AP1000 COL applications. On October 14 and 15, 2008, the NRC staff conducted an audit of design and procurement specifications for pumps, valves, and dynamic restraints to be used for the AP1000 reactor at the Westinghouse office in Monroeville, Pennsylvania. In a memorandum dated November 6, 2008, the NRC staff documented the results of the onsite review with specific open items. For example, the staff found that Westinghouse had included ASME Standard QME-1-2007 in its design and procurement specifications for AP1000 components. ASME QME-1-2007 incorporates lessons learned from valve testing and research programs performed by the nuclear industry and the NRC Office of Nuclear Regulatory Research. Also, AP1000 DCD Tier 2 has been revised in Section 5.4.8.3 to specify that the provisions for qualification testing of power-operated active valves will be based on ASME QME-1-2007. In September 2009, the NRC issued RG 1.100, "Seismic Qualification of Electric and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," Revision 3, which accepts the use of ASME QME-1-2007, with certain staff positions, for the functional design and qualification of safety-related pumps, valves, and dynamic restraints. In a letter dated January 26, 2010, Westinghouse provided its planned response to the audit follow-up items. In a letter dated December 14, 2009, SNC stated, in response to Standard Content Open Item 3.9-1 in the "SER with open items" on the BLN COL application, that it had not identified any specific actions for the VEGP COL application based on the audit open items. The NRC staff discussion of the audit of the design and procurement specifications for pumps, valves, and dynamic restraints to be used for the AP1000 reactor is in the SER on the AP1000 DC amendment application. Therefore, the staff considers Standard Content Open Item 3.9-1 resolved.*

The VEGP COL FSAR incorporates by reference AP1000 DCD Tier 2, Section 3.9.3.4, "Component and Piping Supports," and adds a new Section 3.9.3.4.4, "Inspection, Testing, Repair and/or Replacement of Snubbers." VEGP COL FSAR Section 3.9.3.4.4 specifies that snubber design and testing will satisfy the applicable sections of the ASME BPV Code, ASME OM Code, and ASME QME-1-2007. Further, VEGP COL FSAR Section 3.9.3.4.4 describes the snubber inservice examination and testing program for VEGP Units 3 and 4. For example, the FSAR specifies that the inservice examination and testing of safety-related snubbers will be conducted in accordance with the requirements of the ASME OM Code, Subsection ISTD. The inservice visual examination will be performed to identify physical damage, leakage, corrosion, degradation, indication of binding, misalignment or deformation, and potential defects generic to a particular design. Snubbers will be tested in service to determine operational readiness during each fuel cycle, beginning no sooner than 60 days before the start of the refueling outage. Defined test plan groups will be established and snubbers in each group will be tested each fuel cycle according to an established sampling plan. Unacceptable snubbers will be adjusted, modified, or replaced. Service life for snubbers will be established, monitored, *and adjusted in accordance with ASME OM Code, ISTD-6000, "Service Life Monitoring," and ASME OM Code, Appendix F, "Dynamic Restraints (Snubbers) Service Life Monitoring Methods." In addition, VEGP COL FSAR Table 3.9-201 provides a list of safety-related snubbers to be installed at VEGP, including the snubber identification number and the associated system or component. Revision 3 to RG 1.100 accepts with certain conditions the use of ASME QME-1-2007 for the functional design and qualification of dynamic restraints. The NRC staff finds that the provisions in the VEGP COL FSAR, together with the AP1000 DCD, provide an acceptable description of the inservice examination and testing program for dynamic restraints that support a finding that the program, when developed and implemented, will satisfy the 10 CFR 50.55a regulatory requirements.*

The VEGP COL FSAR incorporates by reference AP1000 DCD Tier 2, Section 3.9.6.2.2, "Valve Testing," with supplemental information. Table 3.9-16 in AP1000 DCD lists the valves in the IST Program for the AP1000 design. VEGP COL FSAR Section 3.9.6.2.2 includes provisions for (a) the establishment of reference values; (b) the prohibition of preconditioning that undermines the purpose of IST activities; (c) comparison of stroke time to the reference value except for fast-acting valves for which a stroke-time limit of 2 seconds is assigned; (d) determination of valve obturator movement during valve exercise tests; (e) testing of solenoid-operated valves; (f) preoperational testing of check valves; (g) acceptance criteria for check valve tests; (h) use of nonintrusive techniques for check valve tests; (i) test conditions for check valve tests; (j) post-maintenance testing for check valves; (k) check valve disassembly and testing; and (l) re-establishment of reference values following maintenance. The VEGP COL FSAR also includes provisions for valve disassembly and inspection; valve preservice tests; and valve replacement, repair, and maintenance in Sections 3.9.6.2.3 to 3.9.6.2.5. The NRC staff finds that these provisions in the VEGP COL FSAR are consistent with Subsection ISTC of the ASME OM Code incorporated by reference in 10 CFR 50.55a, and therefore, are acceptable.

In its letter dated March 1, 2010, SNC provided its planned response for VEGP to Standard Content Open Item 3.9-2 on POV operability tests discussed in the "SER with open items" on the BLN COL application. The NRC staff review of the response by SNC to the three issues in this open item is discussed below.

First, SNC states in its letter dated March 1, 2010, that TVA had indicated in its response to BLN RAI 3.9.6-8 that the BLN COL FSAR would be revised to indicate that MOV testing will apply the provisions of ASME OM Code Case OMN-1 (Revision 1) and the guidance in the Joint Owners Group (JOG) MOV Periodic Verification Program including the applicable NRC safety evaluation (and its supplement) for periodic verification of the design-basis capability of safety-related MOVs. SNC did not consider additional changes to the VEGP COL FSAR to be necessary. The NRC staff finds that the VEGP COL FSAR with its incorporation by reference of the AP1000 DCD (including the planned DCD changes) will address the use of JOG MOV Periodic Verification Program. As the AP1000 IST Program applies the JOG MOV Periodic Verification Program, SNC will need to confirm that MOVs provided by the valve supplier and their application at VEGP Units 3 and 4 are within the scope of the JOG program. The *planned use of ASME OM Code Case OMN-1 (Revision 1) is addressed below in this SER section.*

Second, SNC provides in its letter dated March 1, 2010, a planned revision to the VEGP COL FSAR that specifies the use of Revision 1 to ASME OM Code Case OMN-1 as an alternative to the quarterly MOV stroke-time testing provisions in the ASME OM Code. In the letter, SNC notes that RG 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code," accepts the use of Revision 0 to ASME OM Code Case OMN-1 with three conditions. SNC considers Revision 1 to ASME OM Code Case OMN-1 to represent a superior alternative to Revision 0 to ASME OM Code Case OMN-1 by addressing the conditions on the use of the Code case specified in RG 1.192. In a telephone discussion on April 13, 2010, the NRC staff requested that SNC address the specific provisions in RG 1.192 in justifying the use of Revision 1 to ASME OM Code Case OMN-1 as an alternative to the MOV stroke-time provisions in the ASME OM Code pursuant to 10 CFR 50.55a(a)(3)(i).

In a letter dated May 14, 2010, SNC modified its response to Standard Content Open Item 3.9-2 to provide a planned revision to the VEGP COL FSAR in Section 3.9.6.3 in support of the request to apply Revision 1 to Code Case OMN-1 as an alternative to the quarterly IST stroke-time provisions in the ASME OM Code. The NRC staff has accepted the application of ASME OM Code Case OMN-1 (Revision 0) in RG 1.192 with certain conditions. In the planned VEGP COL FSAR revision, SNC has addressed those conditions as they apply to the requested use of ASME OM Code Case OMN-1 (Revision 1) at VEGP Units 3 and 4. In particular, the VEGP COL FSAR revision specifies that the IST Program will incorporate the provisions in RG 1.192 by providing that the adequacy of the diagnostic test interval for each MOV will be evaluated and adjusted as necessary, but not later than 5 years or three refueling outages (whichever is longer) from the initial implementation of the Code case. The planned VEGP COL FSAR revision also states that the potential increase in core damage frequency (CDF) and risk associated with extending high-risk MOV test intervals beyond quarterly will be determined to be small and consistent with the intent of the Commission's Safety Goal Policy Statement. The VEGP COL FSAR also specifies this provision as consistent with the conditions specified in RG 1.192 for application of ASME OM Code Case OMN-11, "Risk-Informed Testing of Motor-Operated Valves," which has been incorporated into Revision 1 to ASME OM Code Case OMN-1. The planned VEGP COL FSAR revision specifies that risk insights will be applied using MOV risk ranking methodologies accepted by the NRC on a plant-specific or industry-wide basis, consistent with the conditions in the applicable safety evaluations. The planned VEGP COL FSAR revision also indicates that the benefits for performing any particular test will be balanced against the potential adverse effects placed on the valve or system caused by this testing. The VEGP COL FSAR indicates that use of Revision 1 to ASME OM Code Case OMN-1 will be appropriate for the ASME OM Code 2001 Edition with the 2003 Addenda that is the basis for the description of the VEGP Units 3 and 4 IST Program in support of the COL application. The NRC staff finds that the provisions to be specified in the VEGP COL FSAR for the use of Revision 1 to ASME OM Code Case OMN-1 satisfy the conditions specified in RG 1.192 for the use of Revision 0 to ASME OM Code

Case OMN-1. The staff considers Revision 1 in ASME OM Code Case OMN-1 to continue to provide an acceptable technical approach for MOV diagnostic testing as an alternative to quarterly MOV stroke-time testing, and that the changes from Revision 0 to Revision 1 reflect improvements for user application and incorporation of ASME OM Code Case OMN-11. Pursuant to 10 CFR 50.55a(a)(3)(i), the staff authorizes the use of ASME OM Code Case OMN-1 (Revision 1) requested by SNC as an alternative to the quarterly MOV stroke-time testing provisions in the ASME OM Code for VEGP Units 3 and 4 on the basis that the proposed alternative provides an acceptable level of quality and safety and therefore, Standard Content Open Item 3.9-2 is resolved. The incorporation of the planned VEGP COL FSAR changes will be tracked as Confirmatory Item 3.9-1.

<u>Resolution of Standard Content Confirmatory Item 3.9-1</u>

Confirmatory Item 3.9-1 is an applicant commitment to revise its FSAR Table 1.9-201, Section 3.9.6.3, Section 3.9.6.2.2, and Section 3.9.9, to address IST of valves. The staff verified that the VEGP COL FSAR was appropriately updated. As a result, Confirmatory Item 3.9-1 is now closed.

Third, SNC in its March 1, 2010, submittal provides several planned changes to the VEGP COL FSAR to clarify the provisions that would be redundant when combined with the valve testing provisions in the AP1000 DCD. The NRC staff considers the proposed changes to the VEGP COL FSAR to be acceptable because these provisions are incorporated by reference as part of the AP1000 DCD. The incorporation of the planned VEGP COL FSAR changes will be tracked as part of Confirmatory Item 3.9-2.

<u>Resolution of Standard Content Confirmatory Item 3.9-2</u>

Confirmatory Item 3.9-2 is an applicant commitment to revise its FSAR. The staff verified that the VEGP COL FSAR was appropriately updated. As a result, Confirmatory Item 3.9-2 is now closed.

In light of the weaknesses in the IST provisions in the ASME OM Code for quarterly MOV stroke-time testing, the NRC issued Generic Letter (GL) 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves," to request that nuclear power plant licensees establish programs to assure the capability of safety-related MOVs to perform their design-basis functions on a periodic basis. Further, the NRC revised 10 CFR 50.55a to require that nuclear power plant licensees supplement the quarterly MOV stroke-time testing provisions specified in the ASME OM Code with a program to ensure that MOVs continue to be capable of performing their design-basis safety functions. In its letter dated March 1, 2010, SNC provided its response to Standard Content Open Item 3.9-3 related to MOV testing in the "SER with open items" on the BLN COL application. The NRC staff review of the response by SNC to the six issues in this open item is discussed below:

First, SNC notes the planned use of Revision 1 to ASME OM Code Case OMN-1 as part of the IST Program to be developed for VEGP. As discussed above in *this SER section, the NRC staff authorized the use of Revision 1 to ASME OM Code Case OMN-1 at VEGP Units 3 and 4.*

Second, SNC states that the MOV Testing Program at VEGP will implement the JOG MOV Periodic Verification Program as described in the VEGP COL FSAR and AP1000 DCD. As indicated above, the NRC staff finds that the VEGP COL FSAR with its incorporation by reference of the AP1000 DCD (including the planned DCD changes) will address the use of the JOG MOV Periodic Verification Program. Other necessary changes to the VEGP COL FSAR regarding MOV testing are discussed in this SER section.

Third, SNC indicates that MOV output capability will be determined using the provisions of ASME OM Code Case OMN-1. The NRC staff has reviewed ASME OM Code Case OMN-1 as part of its acceptance in RG 1.192, and has determined that the Code case provides acceptable provisions for diagnostic testing to determine the output capability of MOVs.

Fourth, SNC describes MOV testing using the guidance in the JOG MOV Periodic Verification Program and Revision 1 to ASME OM Code Case OMN-1 to periodically determine the capability of MOVs to perform under design-basis conditions. The NRC staff has reviewed the JOG MOV Periodic Verification Program as part of its acceptance in an NRC safety evaluation dated September 25, 2006 with a supplement dated September 18, 2008, and has reviewed ASME OM Code Case OMN-1 as part of its acceptance in RG 1.192. From those evaluations, the staff has determined that the JOG MOV Periodic Verification Program and ASME OM Code Case OMN-1 will demonstrate continued MOV capability to open and close under design-basis conditions. As discussed above in this SER section, the NRC staff authorized the use of Revision 1 to ASME OM Code Case OMN-1 at VEGP Units 3 and 4.

Fifth, SNC notes that the initial test frequency of POVs will be based on the ASME OM Code or applicable ASME OM Code cases. For example, the VEGP COL FSAR specifies that the IST frequency will be determined as specified by ASME OM Code Case OMN-1. Further, the JOG MOV Periodic Verification Program with the NRC safety evaluation and its supplement includes provisions for MOV test frequencies based on risk ranking and functional margin with a maximum diagnostic test interval of 10 years. The staff considers these provisions in the VEGP COL FSAR and the AP1000 DCD for POV test frequency to incorporate lessons learned from MOV testing and research programs, and therefore, to be acceptable.

Sixth, SNC describes provisions for successful completion of MOV testing at VEGP in its March 1, 2010, letter, and provides several planned changes to the VEGP COL FSAR. For example, SNC provides a planned FSAR change to specify the use of ASME OM Code Case OMN-1, Revision 1. SNC also plans to revise the FSAR to specify that the design-basis capability testing of MOVs will apply guidance from GL 96-05 and the JOG MOV Periodic Verification Program. SNC will revise the FSAR to note the need to consider degraded voltage, control switch repeatability, and load-sensitive MOV behavior in ensuring that MOVs have adequate capability margin, in addition to the consideration of age-related *degradation. SNC provides a proposed addition to the description of the MOV test frequency determination in the FSAR that will specify that maximum torque and/or thrust (as applicable) achieved by the MOV (allowing sufficient margin for diagnostic equipment inaccuracies and control switch repeatability) must not exceed the allowable structural and undervoltage motor capability limits for the individual parts of the MOV. SNC provides a proposed addition to the description of POV operability testing that specifies that successful completion of the preservice testing and IST of MOVs, in addition to MOV testing as required by 10 CFR 50.55a, will demonstrate that the following criteria are met for each valve tested: (i) valve fully opens and/or closes as required by its safety function; (ii) adequate margin exists and includes consideration of diagnostic equipment inaccuracies, degraded voltage, control switch repeatability, load-sensitive MOV behavior, and margin for degradation; and (iii) maximum torque and/or thrust (as applicable) achieved by the MOV (allowing sufficient margin for diagnostic equipment inaccuracies and control switch repeatability) does not exceed the allowable structural and undervoltage motor capability limits for the individual parts of the MOV. In its letter dated May 14, 2010, SNC provided an additional planned revision to the VEGP COL FSAR that clarifies the application of the JOG MOV Periodic Verification Program (including the applicable NRC safety evaluation and its supplement on the JOG program) in response to NRC staff comments provided during the telephone discussion on April 13, 2010. The NRC staff considers the planned changes to the VEGP COL FSAR to resolve Standard Content Open Item 3.9-3. The incorporation of the planned changes to the VEGP COL FSAR will be tracked as Confirmatory Item 3.9-3.*

<u>Resolution of Standard Content Confirmatory Item 3.9-3</u>

Confirmatory Item 3.9-3 is an applicant commitment to revise its FSAR Section 3.9.6.2.2 to address MOV testing. The staff verified that the VEGP COL FSAR was appropriately updated. As a result, Confirmatory Item 3.9-3 is now closed.

In addition to incorporating by reference AP1000 DCD Tier 2 Section 3.9.6.2.2, the VEGP COL FSAR includes a paragraph titled "Other Power-Operated Valve Operability Tests," that states that POVs other than active MOVs are exercised quarterly in accordance with ASME OM Code, Subsection ISTC, unless justification is provided in the IST Program for testing these valves at other Code-mandated frequencies. Lessons learned from the resolution of weaknesses in the design, qualification, and testing of MOVs are also applicable to other POVs used at nuclear power plants. In discussing the MOV lessons learned applicable to other POVs in Regulatory Issue Summary (RIS) 2000-03, "Resolution of Generic Safety Issue 158: Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions," the NRC staff determined that the current regulations provide adequate requirements to ensure design-basis capability of safety-related POVs. For example, the staff noted that licensees are required by 10 CFR 50.65 (Maintenance Rule) to monitor the performance of SSCs in a manner sufficient to provide reasonable assurance that the SSCs are capable of fulfilling their intended functions. VEGP COL FSAR Section 3.9.6.2.2 provides a description of operability testing for POVs other than MOVs to be implemented at VEGP. For example, the FSAR states that *subsequent to verification of the design-basis capability of POVs as part of the design and qualification program, POVs that perform an active safety function will be tested after installation to ensure valve setup is acceptable to perform their required functions consistent with valve qualification. This testing will document the baseline performance of the valves and will include measurement of critical parameters with consideration of uncertainties associated with the performance of these tests and use of the test results. Additional periodic testing will be performed as part of the air-operated valve (AOV) program based on the JOG AOV program discussed in RIS 2000-03 with specific reference to NRC staff comments on that program. The AOV program will also include the attributes for a successful POV periodic verification program described in RIS 2000-03 by incorporating lessons learned from nuclear power plant operations and research programs as they apply to the periodic testing of AOVs and other POVs in the IST Program. The FSAR specifies AOV program attributes including valve categorization based on safety significance and risk ranking, AOV setpoints based on current vendor information or valve qualification diagnostic testing, periodic static testing to identify potential degradation, use of sufficient diagnostics to collect relevant data to verify that the valve meets functional requirements, specification of test frequency and evaluation based on data trends, post-maintenance procedures to ensure baseline testing will be re-performed as necessary when high-risk valve performance could be affected, inclusion of lessons learned from other valve programs, and retention and periodic evaluation of AOV test documentation.*

The NRC staff has reviewed the VEGP COL FSAR, including the incorporation by reference of the AP1000 DCD, to determine whether it addresses the lessons learned from MOV operating experience and research programs in describing the program for the periodic verification of the design-basis capability of POVs other than MOVs. In its letters dated December 14, 2009, and March 1, 2010, SNC provided a response to Standard Content Open Item 3.9-4 related to other POV operability testing in the "SER with open items" on the BLN COL application. In particular, SNC provided planned changes to the VEGP COL FSAR to clarify the potential need for periodic dynamic testing of POVs other than MOVs based on the design qualification results or valve operating experience. The planned FSAR change will also clarify that post-maintenance procedures will be implemented for all safety-related POVs consistent with the QA requirements in 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," regardless of their specific risk ranking. SNC also provided a proposed change to the VEGP COL FSAR specifying that the attributes of the AOV testing program, to the extent that they apply to and can be implemented on other safety-related POVs (such as electro-hydraulic valves) will be applied to those other POVs. The NRC staff considers that the planned revision to the VEGP COL FSAR, when combined with the AP1000 DCD provisions incorporated by reference, will adequately describe the periodic testing program for POVs other than MOVs to be used at VEGP and resolves Standard Content Open Item 3.9-4. The incorporation of the planned changes to the VEGP COL FSAR will be tracked as Confirmatory Item 3.9-4.

<u>Resolution of Standard Content Confirmatory Item 3.9-4</u>

Confirmatory Item 3.9-4 is an applicant commitment to revise its FSAR Section 3.9.6.2.2, to address POV testing. The staff verified that the VEGP COL FSAR was appropriately updated. As a result, Confirmatory Item 3.9-4 is now closed.

The VEGP COL FSAR incorporates by reference AP1000 DCD Tier 2, Section 3.9.6.3, "Relief Requests," with a discussion of the planned use of ASME OM Code Case OMN-1, Revision 1. The applicant stated that use of Revision 1 to ASME OM Code Case OMN-1 will require request for relief, unless it is approved by the NRC in RG 1.192 or incorporated into the ASME OM Code on which the IST Program is based and that Code Edition is incorporated by reference in 10 CFR 50.55a. As discussed above in this SER section, the NRC staff authorized the use of Revision 1 to the ASME OM Code Case OMN-1 at VEGP Units 3 and 4.

AP1000 DCD Tier 2, Section 3.9.2, "Dynamic Testing and Analysis," describes tests to confirm that piping, components, restraints, and supports have been designed to withstand the dynamic effects of steady-state FIV and anticipated operational transient conditions. Section 14.2.9.1.7, "Expansion, Vibration and Dynamic Effects Testing," in AP1000 DCD Tier 2, Chapter 14, "Initial Test Program," states that the purpose of the expansion, vibration and dynamic effects testing is to verify that safety-related, high energy piping and components are properly installed and supported such that, in addition to other factors, vibrations caused by steady-state or dynamic effects do not result in excessive stress or fatigue to safety-related plant systems. Nuclear power plant operating experience has revealed the potential for adverse flow effects from vibration caused by hydrodynamic loads and acoustic resonance on reactor coolant, steam, and feedwater systems. In its letter dated January 12, 2010, SNC provided its response for VEGP to Standard Content Open Item 3.9-5 related to FIV in the "SER with open items" on the BLN COL application. In its response, SNC stated that it intended to use the overall Initial Test Program to demonstrate that the plant has been constructed as designed and the systems perform consistent with design requirements. SNC referenced the provisions in the AP1000 DCD for vibration monitoring and testing to be implemented at VEGP. For example, the applicant notes that AP1000 DCD Tier 2, Section 3.9.2.1, "Piping Vibration, Thermal Expansion and Dynamic Effects," specifies that the preoperational test program for ASME BPV Code, Section III, Class 1, 2, and 3 piping systems simulates actual operating modes to demonstrate that components comprising these systems meet functional design requirements and that piping vibrations are within acceptable levels. SNC indicates that the planned vibration testing program described in AP1000 DCD Tier 2, Sections 14.2.9 and 14.2.10, with the preservice and IST programs described in AP1000 DCD Tier 2, Sections 3.9.3.4.4 and 3.9.6, will confirm component installation in accordance with design requirements, and address the effects of steady-state (flow-induced) and transient vibration to ensure the operability of valves and dynamic restraints in the IST Program. The NRC staff considers the response by SNC clarifies its application of the provisions in the AP1000 DCD to ensure that potential adverse flow effects will be addressed at VEGP. Therefore, *the staff considers Standard Content Open Item 3.9-5 to be resolved for the VEGP COL application.*

Subsection ISTC-5260, "Explosively Actuated Valves," in the ASME OM Code specifies that at least 20 percent of the charges in explosively actuated valves shall be fired and replaced at least once every 2 years. If a charge fails to fire, the ASME OM Code states that all charges with the same batch number shall be removed, discarded, and replaced with charges from a different batch. In light of the updated design and safety significance of squib valves in new reactors, the need for improved surveillance activities for squib valves is being considered by the nuclear industry, ASME, and U.S. and international nuclear regulators. In RAI 3.9.6-1, the NRC staff requested that SNC describe its plans for addressing the surveillance of squib valves that will provide reasonable assurance of the operational readiness of those valves to perform their safety functions in support of the VEGP COL application. In a letter dated May 27, 2010, SNC submitted a planned revision to VEGP COL FSAR Section 3.9.6 to specify that industry and regulatory guidance will be considered in the development of the IST Program for squib valves. The FSAR will also state that the IST Program for squib valves will incorporate lessons learned from the design and qualification process for these valves such that surveillance activities provide reasonable assurance of the operational readiness of squib valves to perform their safety functions. The NRC staff finds that the planned changes to the VEGP COL FSAR are sufficient to describe the IST Program for squib valves for incorporating the lessons learned from the design and qualification process in developing surveillance activities that will provide reasonable assurance of the operational readiness for squib valves to perform their safety functions. Therefore, the NRC staff considers the planned changes to the VEGP COL FSAR to resolve this RAI acceptable. The incorporation of the planned changes to the VEGP COL FSAR will be tracked as Confirmatory Item 3.9-5.

<u>Resolution of Standard Content Confirmatory Item 3.9-5</u>

Confirmatory Item 3.9-5 is an applicant commitment to revise its FSAR Section 3.9.6.2.2 to address squib valve testing. The staff verified that the VEGP COL FSAR was appropriately updated. As a result, Confirmatory Item 3.9-5 is now closed.

<u>Technical Specifications</u>

In its letter dated December 14, 2009, SNC provided a response to an open item related to Part 4, "Technical Specifications," (Standard Content Open Item 3.9-6) in the "SER with open items" on the BLN COL application. In its response, SNC stated that Part 4 of the VEGP COL application will be revised to ensure that Technical Specifications and Technical Specification Bases are consistent with the ASME OM Code, 2001 Edition through the 2003 Addenda. Therefore, the NRC staff considers the planned changes to the VEGP COL application in Part 4 to resolve Standard Content Open Item 3.9-6. The incorporation of the planned changes to the VEGP COL FSAR will be tracked as Confirmatory Item 3.9-6.

<u>Resolution of Standard Content Confirmatory Item 3.9-6</u>

Confirmatory Item 3.9-6 is an applicant commitment to revise its FSAR Section 3.9.6.2.2 to address the ASME OM Code. The staff verified that the VEGP COL FSAR was appropriately updated. As a result, Confirmatory Item 3.9-6 is now closed.

<u>License Conditions</u>

- *Part 10, License Condition 3, Items G.2 and G.5*

The applicant proposed a license condition providing the implementation milestones for the Preservice Testing Program and MOV Testing Program.

- *Part 10, License Condition 6*

The applicant proposed a license condition to provide a schedule to support the NRC's inspection of operational programs including the Preservice Testing Program and MOV Testing Program. These license conditions are consistent with the policy established in SECY-05-0197 and are, thus, acceptable.

<u>Squib Valves</u>

During the uncontested hearing for the VEGP Units 3 and 4 COL application, the Commission discussed issues associated with the inservice testing and inspection program for squib valves to be used to perform safety functions at VEGP Units 3 and 4. Tier 1 of the AP1000 DCD requires squib valves to undergo tests or type tests to demonstrate their operational capability under design conditions. Additionally, the Commission asked the staff questions on this topic after the VEGP and V.C. Summer Nuclear Station (VCSNS) COL uncontested hearings. For these COL applications, the Commission concluded that, although it found that the staff's review of the squib valve issues was rigorous, it had a concern similar to that initially raised by the Advisory Committee on Reactor Safeguards (ACRS) regarding the status of the inservice testing and inspection program for this component. As such, the Commission imposed a license condition for each COL that directs the implementation of a surveillance program for squib valves at VEGP Units 3 and 4 and VCSNS Units 2 and 3, with the specific requirements described in the Commission orders authorizing issuance of the VEGP and VCSNS COLs.

The squib valves subject to the surveillance program license condition under the VEGP and VCSNS COLs are part of the AP1000 certified design, and the same squib valves are specified in the Levy COL application. Therefore, the staff determined that it was appropriate to apply the same surveillance program license condition to the WLS Units 1 and 2 squib valves.

The surveillance program is established to provide reasonable assurance that the WLS squib valves are operational and ready to perform their safety function. The staff-proposed license condition follows the precedent set in the VEGP and VCSNS COLs (ADAMS Accession Nos. ML113540620 and ML113420105) to require such a surveillance program.

3.9.6.5 *Post Combined License Activities*

The license condition language in this section has been clarified from previously considered language. In a letter dated March 22, 2016 (ADAMS Accession No. ML16084A099), the applicant did not identify any concerns with the clarified license condition language. The changes do not affect the staff's above analysis of the conditions, and therefore, for the reasons discussed in the technical evaluation section above, the staff finds the following license conditions acceptable:

- License Condition (3-5) – Before initial fuel load, the licensee shall implement (1) the Preservice Testing Program and (2) the Motor-Operated Valve Testing Program.

- License Condition (3-6) – No later than 12 months after issuance of the COL, the licensee shall submit to the Director of NRO a schedule that supports planning for and conduct of NRC inspections of the IST program (including preservice and MOV testing). The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until the inservice testing program (including preservice testing and the MOV testing) has been fully implemented or the plant has been placed in commercial service, whichever comes first.

- License Condition (3-7) – Before initial fuel load, the licensee shall implement a surveillance program for explosively actuated valves (squib valves) that includes the following provisions in addition to the requirements specified in the edition of the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) as incorporated by reference in 10 CFR 50.55a.

    a. Preservice Testing

    All explosively actuated valves shall be preservice tested by verifying the operational readiness of the actuation logic and associated electrical circuits for each explosively actuated valve with its pyrotechnic charge removed from the valve. This must include confirmation that sufficient electrical parameters (voltage, current, resistance) are available at the explosively actuated valve from each circuit that is relied upon to actuate the valve. In addition, a sample of at least 20% of the pyrotechnic charges in all explosively actuated valves shall be tested in the valve or a qualified test fixture to confirm the capability of each sampled pyrotechnic charge to provide the necessary motive force to operate the valve to perform its intended function without damage to the valve body or connected piping. The sampling must select at least one explosively actuated valve from each redundant safety train. Corrective action shall be taken to resolve any deficiencies identified in the operational readiness of the actuation logic or associated electrical circuits, or the capability of a pyrotechnic charge. If a charge fails to fire or its capability is not confirmed, all charges with the same batch number shall be removed, discarded, and replaced with charges from a different batch number that has demonstrated successful 20% sampling of the charges.

    b. Operational Surveillance

    Explosively actuated valves shall be subject to the following surveillance activities after commencing plant operation:

    (1) At least once every 2 years, each explosively actuated valve shall undergo visual external examination and remote internal examination (including evaluation and removal of fluids or contaminants that may interfere with operation of the valve)

to verify the operational readiness of the valve and its actuator. This examination shall also verify the appropriate position of the internal actuating mechanism and proper operation of remote position indicators. Corrective action shall be taken to resolve any deficiencies identified during the examination with post-maintenance testing conducted that satisfies the preservice testing requirements.

(2) At least once every 10 years, each explosively actuated valve shall be disassembled for internal examination of the valve and actuator to verify the operational readiness of the valve assembly and the integrity of individual components and to remove any foreign material, fluid, or corrosion. The examination schedule shall provide for both of the two valve designs used for explosively actuated valves at the facility to be included among the explosively actuated valves to be disassembled and examined every 2 years. Corrective action shall be taken to resolve any deficiencies identified during the examination with post-maintenance testing conducted that satisfies the preservice testing requirements.

(3) For explosively actuated valves selected for test sampling every 2 years in accordance with the ASME OM Code, the operational readiness of the actuation logic and associated electrical circuits shall be verified for each sampled explosively actuated valve following removal of its charge. This must include confirmation that sufficient electrical parameters (voltage, current, resistance) are available for each valve actuation circuit. Corrective action shall be taken to resolve any deficiencies identified in the actuation logic or associated electrical circuits.

(4) For explosively actuated valves selected for test sampling every 2 years in accordance with the ASME OM Code, the sampling must select at least one explosively actuated valve from each redundant safety train. Each sampled pyrotechnic charge shall be tested in the valve or a qualified test fixture to confirm the capability of the charge to provide the necessary motive force to operate the valve to perform its intended function without damage to the valve body or connected piping. Corrective action shall be taken to resolve any deficiencies identified in the capability of a pyrotechnic charge in accordance with the preservice testing requirements.

This license condition shall expire upon (1) incorporation of the above surveillance provisions for explosively actuated valves into the facility's inservice testing program, or (2) incorporation of inservice testing requirements for explosively actuated valves in new reactors (i.e., plants receiving a construction permit, or combined license for construction and operation, after January 1, 2000) to be specified in a future edition of the ASME OM Code as incorporated by reference in 10 CFR 50.55a, including any conditions imposed by the NRC, into the facility's inservice testing program.

3.9.6.6 *Conclusion*

The staff reviewed the WLS COL application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to the IST Program, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the design-related information incorporated by reference in the COL application are documented in NUREG-1793 and its supplements. The results of the staff's review of the material in the AP1000 DCD related to the IST operational program for pumps, valves, and dynamic restraints are in this section of the report. In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the guidance in NUREG-0800, Section 3.9.6 and in RG 1.206. The staff based its conclusion on the following:

- WLS DEP 6.4-2, related to design changes affecting how the temperature and humidity in the main control room are maintained within the limits for reliable human performance, is reviewed and found acceptable by the staff in Section 21.3 of this SER.

- STD COL 3.9-4, regarding the operational program for pumps, valves, and dynamic restraints is acceptable because the requirements of 10 CFR 52.79(a) are satisfied.

3.9.7 Integrated Head Package

AP1000 DCD Section 3.9.7 describes the integrated head package (IHP). The IHP combines several components in one assembly to simplify refueling the reactor. The IHP includes a lifting rig, seismic restraints for CRDM, support for reactor head vent piping, cable bridge, power cables, cables for in-core instrumentation, cable supports, and shroud assembly. The IHP provides the ability to rapidly disconnect cables, including the CRDM power cables, digital rod position indication cables, and in-core instrument cables from the components.

WLS COL FSAR, Revision 11, Section 3.9, incorporates by reference, with no departures or supplements, AP1000 DCD, Revision 19 Section 3.9.7, "Integrated Head Package." The staff reviewed the WLS COL application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The staff's review confirmed that there is no outstanding issue related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

3.10 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

3.10.1 Introduction

Seismic Category I equipment includes the following types (1) safety-related active mechanical equipment that performs a mechanical motion while accomplishing a system safety-related function. Examples include pumps, valves, and valve operators, (2) safety-related, nonactive mechanical equipment whose mechanical motion is not required while accomplishing a system safety-related function, but whose structural integrity must be maintained to fulfill its design safety-related function (3) safety-related instrumentation and electrical equipment and certain monitoring equipment.

Mechanical and electrical equipment (including instrumentation and controls), and where applicable, their supports classified as Seismic Category I must demonstrate that they are capable of performing their intended safety-related functions under the full range of normal and accident (including seismic) loadings. The equipment includes devices associated with systems essential to safe shutdown, containment isolation, reactor core cooling, containment and reactor heat removal, or equipment otherwise essential to prevent significant release of radioactive material to the environment or in mitigating the consequences of accidents.

3.10.2 Summary of Application

WLS COL FSAR, Revision 11, Section 3.10, incorporates by reference AP1000 DCD, Revision 19, Section 3.10. This section of the WLS COL FSAR does not include any COL information items or supplemental information related to AP1000 DCD Section 3.10.

3.10.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. The acceptance criteria associated with the relevant requirements of NRC regulations for the seismic and dynamic qualification of mechanical and electrical equipment are given in NUREG-0800, Section 3.10.

3.10.4 Technical Evaluation

The staff reviewed this application and checked the referenced DCD to ensure that the combination of the DCD and the WLS COL application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the seismic and dynamic qualification program. The staff's evaluation of the information incorporated by reference in the WLS COL application is documented in NUREG-1793 and its supplements. Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform a technical review for each "standard issue" and use this review to evaluate subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the Reference COL application (i.e., VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In the comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from responses to RAIs.
- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed and that those changes were actualized in the WLS COL FSAR.
- The staff verified that site-specific differences, if any, did not adversely affect any previous relevant evaluation or conclusion,

The staff completed its review and concluded that the evaluation performed for the standard content is directly applicable to the WLS COL application. This standard content material is identified in this report by use of italicized, double-indented formatting.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 3.10.4:

<u>Implementation Program</u>

In RAI 3.10-1, dated August 7, 2008, the applicant was requested to provide an implementation program, including milestones and completion dates with appropriate information submitted with sufficient time for staff review and approval prior to installation of the equipment, not prior to fuel loading, in accordance with Section C.I.3.10.4 of RG 1.206.

In its response, the applicant stated that details of the implementation milestones for the seismic and dynamic qualification program are not currently available, and are not expected to be available until after a detailed construction schedule of the plant has been developed. Appropriate scheduling information will be provided, when available, to the NRC as necessary to support timely completion of their inspection and audit functions. Additionally, seismic and dynamic qualification is the subject of ITAAC, and 10 CFR 52.99(a) does not require that a schedule for implementing ITAAC be provided to the NRC until one year after issuance of the COL.

*The NRC staff determined that the applicant's response to RAI 3.10-1 is not adequate because, in accordance with Section C.I.3.10.4 of RG 1.206, if the results of seismic and dynamic qualification is not available at the time of the COL application, the applicant is expected to submit the following before the issuance of the combined license: (1) descriptions of the implementation program such as identification of seismic qualification methods (Testing or Analysis) for each type of equipment; and (2) milestones for when the different aspects of the seismic qualification program will be complete - dates or condition should be such that the NRC staff will be able to audit the qualification results prior to the installation of the equipment (not before fuel loading as part of the ITAAC program). This is **Open Item 3.10-1.***

<u>Resolution of Open Item 3.10-1</u>

In its responses dated February 5, 2010 and April 2, 2010, the VEGP applicant submitted a table providing the planned methods of seismic qualification for safety-related, seismic Category I equipment types listed in AP1000 DCD, Chapter 3, Table 3.2-3. Furthermore, the applicant stated that the seismic qualification packages will be available to the NRC as necessary to support timely completion of its inspection and audit functions. Because not all packages are expected to be completed within a year of the issuance of the COL (or at the start of construction as defined in 10 CFR 50.10(a), whichever is later), a schedule for the availability of the seismic qualification packages will be included with the schedule information for closure of ITAAC (as required by 10 CFR 52.99(a)). The staff finds the applicant's response acceptable, and Open Item 3.10-1 is closed. The incorporation of the planned changes to the VEGP COL FSAR is complete.

On April 25, 2012, the staff issued RAI 105, Questions 01.05-1 through 01.05-4 on the NRC Fukushima NTTF Recommendation 2.1, Recommendation 9.3, spent fuel instrumentation, and mitigation strategies for beyond-design-basis events. The staff completed the review of the Duke Energy's January 30, 2014, response as supplemented by the February 28, 2014, response to the questions in RAI 105. In the Duke Energy's January 30, 2014, response as supplemented by the February 28, 2014, the applicant stated that the only equipment potentially affected by the identified high frequency (HF) exceedance is tested to levels higher than those imposed by those equipment ISRS. In RAI 114, Question 03.10-1, the staff questioned how the exceedances for those affected equipment were resolved if the seismic qualification by analysis was used. In a June 5, 2014, response to RAI 114, Question 03.10-1, Duke Energy stated that AP1000 DCD Chapter 3, Appendix 3I identifies equipment that is potentially sensitive to HF excitation. No such HF sensitive equipment is qualified by analysis, and all such HF sensitive equipment is qualified by testing, thereby confirming that equipment is adequate for the site-specific demands. The staff finds the applicant's response acceptable and, therefore, considers RAI 114, Question 03.10-1 resolved.

In RAI 114, Question 03.10-2, the staff requested that the applicant provide technical explanation for the fact that the AP1000 combined CSDRS & HRHF equipment Required Response Spectra (RRS) exceed the WLS RRS, even though both the horizontal and vertical design ground motion response spectra for WLS-CEUS spectra exceed the AP1000 CSDRS and AP1000 HRHF ground spectra (RRS) for some equipment. In a June 5, 2014, response to RAI 114, Question 03.10-2, the applicant stated that the differences in determining ISRS are discussed in WLG-GW-GLR-815, Section 5.3 and the ISRS differences are predominately attributed to (1) model refinements and (2) rock profile differences between the AP1000 generic hard rock profile and the site-specific WLS concrete/rock profiles (i.e., lower shear wave velocity (Vs) in the 15 m (50 ft) directly below basemat. The staff finds the applicant's response acceptable and, therefore, considers RAI 114, Question 3.10-2 resolved.

In RAI 114, Question 03.10-3, the staff requested that the applicant provide the source of test data for protection and monitoring systems, nuclear instrumentation source- and intermediate-range systems, and main control room/remote shutdown panels, shown on the Duke Energy January 20, 2014, submittal (WLG-GW-GLR-815, Letter WLG 2014.01-02) Enclosure 4, Figures 6.4-1 through 6.4-8. In a June 5, 2014, response to RAI 114, Question 03.10-3, the applicant stated that Westinghouse calculation WLG-1000-SC2-702 documents the source of AP1000 TRS/RRS testing references and the site-specific Lee RRS plots in WLG-GW-GLR-815, Figures 6.4-1 through 6.4-8 for some equipment cited in WLG-GW-GLR-815, specifically, Westinghouse calculations, APP-PMS-VPR-006, APP-PMS-VPR-004, and APP-JW03-VBR-001. The staff reviewed those calculations and verified the source of the TRS shown in WLG-GW-GLR-815, Figures 6.4-1 through 6.4-8. The staff finds the applicant's response acceptable and, therefore, considers RAI 114, Question 03.10-3 resolved.

3.10.5 Post Combined License Activities

There are no post COL activities related to this section.

3.10.6 Conclusion

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to the seismic and dynamic qualification program, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements. The staff compared the information in the application to relevant NRC regulations, the acceptance criteria in NUREG-0800, Section 3.10. The staff's review confirmed that the applicant has adequately addressed the COL information relating to the seismic qualification of equipment in accordance with the requirements of GDC 2, GDC 4, and GDC 14.

3.11 Environmental Qualification of Mechanical and Electrical Equipment

3.11.1 Introduction

The objective of environmental qualification (EQ) is to reduce the potential for common failure due to specified environmental and seismic events and to demonstrate that equipment within the scope of the EQ program is capable of performing its intended design safety function under all conditions including environmental stresses resulting from design bases events. The information presented includes identification of the equipment required to be environmentally qualified and, for each item of equipment, the designated functional requirements, definition of the applicable environmental parameters, and documentation of the qualification process employed to demonstrate the required environmental capability. During plant operation, the licensee implements the EQ program. This specifies the replacement frequencies of affected safety-related equipment in harsh environments, and non-safety-related equipment whose failure under the postulated environmental conditions could prevent satisfactory performance of the safety functions of the safety-related equipment, and certain post-accident monitoring equipment. The seismic qualification of mechanical and electrical equipment is presented in Section 3.10 of this report. The portions of post-accident monitoring equipment required to be environmentally qualified are identified in AP1000 DCD Table 7.5-1.

RG 1.206 discusses the Commission's position provided in SECY-05-0197 stating that operational programs should be fully described in COL applications to avoid the need to specify ITAAC for those programs. The applicant relies on the WLS COL application with its incorporation by reference of the AP1000 DCD and supplemental information to fully describe the EQ program and other related operational programs.

3.11.2 Summary of Application

WLS COL FSAR, Revision 11, Section 3.11 incorporates by reference AP1000 DCD, Revision 19, Section 3.11. AP1000 DCD Section 3.11 describes the EQ Program for electrical and mechanical equipment to be used in the AP1000 certified design.

Departures

- WLS DEP 3.11-1

In WLS COL FSAR Table 3.11-201 (Sheet 14 of 51), "Environmentally Qualified Electrical and Mechanical Equipment," the applicant added three spent fuel pool level instruments related to the Fukushima Lessons Learned report. The staff addressed the departure in the technical evaluation section below.

- WLS DEP 6.4-2

The applicant provided additional information in Tables 3.11-202, 3I-201, and 3I-202 and in Figure 3D-201 of the WLS COL FSAR about WLS DEP 6.4-2 related to design changes affecting how the temperature and humidity in the main control room are maintained within the limits for reliable human performance. This information, as well as related WLS DEP 6.4-2 information appearing in other chapters of the FSAR, is reviewed in Section 21.3 of this SER.

AP1000 COL Information Item

- STD COL 3.11-1

In WLS COL FSAR Section 3.11.5, "Combined License Information Item For Equipment Qualification File," the applicant provided information to address COL Information Item 3.11-1 regarding administrative control and milestones for implementation of the EQ Program for WLS.

License Conditions

- Part 10, License Condition 3, "Operational Program Implementation," Item G.1

The applicant proposed a license condition requiring the submittal of a schedule to the NRC to aid in the planning for and conduct NRC inspections of operational programs including the EQ program.

- Part 10, License Condition 6

The applicant proposed a license condition to provide a schedule to support the NRC's inspection of operational programs including the EQ Program.

3.11.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. The acceptance criteria associated with the relevant requirements of NRC regulations for the environmental qualification of mechanical and electrical equipment are given in NUREG-0800, Section 3.11.

The applicable regulatory requirements for the Operational EQ program are as follows:
10 CFR 52.79(a)(10) requires that a COL application provide a description of the program, and its implementation, required by 10 CFR 50.49(a) for the EQ of electric equipment important to safety and the list of electric equipment important to safety that is required by 10 CFR 50.49(d).
10 CFR 52.79(a)(29)(I) requires that a COL application provide plans for conduct of normal operations, including maintenance, surveillance, and periodic testing of SSCs. RG 1.206 provides guidance for a COL applicant in preparing and submitting its COL application in accordance with the NRC regulations. For example, RG 1.206, Section C.IV.4 discusses the requirement in 10 CFR 52.79(a) for descriptions of operational programs that need to be included in the FSAR for a COL application to allow a reasonable assurance finding of acceptability. In particular, a COL applicant should fully describe EQ and other operational programs as defined in Commission Paper SECY-05-0197 to avoid the need for ITAAC for the implementation of those programs. The term "fully described" for an operational program should be understood to mean that the program is clearly and sufficiently described in terms of

scope and level of detail to allow a reasonable assurance finding of acceptability. Further, operational programs should be described at a functional level and an increasing level of detail where implementation choices could materially and negatively affect the program effectiveness and acceptability. The Commission approved the use of a license condition for operational program implementation milestones that are fully described or referenced in the FSAR as discussed in the February 22, 2006, SRM for SECY-05-0197.

3.11.4 Technical Evaluation

Mechanical and electrical equipment (including instrumentation and controls), its supports (classified as Seismic Category I) must demonstrate that it is capable of performing its intended safety-related functions under the full range of normal and accident (including seismic) loadings. This equipment includes devices associated with systems essential to safe shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal, or equipment otherwise essential in preventing significant release of radioactive material to the environment or in mitigating the consequences of accidents.

The staff reviewed this application section and checked the referenced DCD to ensure that the combination of the DCD and the WLS COL application represents the complete scope of information relating to this review topic. The staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the environmental qualification of mechanical and electrical equipment. The staff's evaluation of the information incorporated by reference in the WLS application is documented in NUREG-1793 and its supplements. Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform a technical review for each "standard issue" and use this review to evaluate subsequent COL applications. To ensure that the staff's findings on standard content for the Reference COL application FSAR (i.e., VEGP Units 3 and 4) were equally applicable to the WLS application, the staff undertook the following reviews.

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In the comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the application, as applicable) resulting from responses to RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed and that those changes were actualized in the COL FSAR.

- The staff verified that site-specific differences, if any, did not adversely affect any previous relevant evaluation or conclusion.

The staff completed its review and concluded that the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this report by use of italicized, double indented formatting. Section 1.2.3 of this report provides an explanation of why the standard content material from the SER for the Reference COL application (i.e., VEGP) contains evaluation material from the SER for the BLN, Units 3 and 4 COL application.

Departures

- WLS DEP 3.11-1

In WLS COL FSAR Table 3.11-201, the applicant included a departure of "Environmental Zone" for three spent fuel pool level instruments (SFS-JE-LT 019A, SFS-JE-LT 019B, and SFS-JE-LT 019C) from AP1000 DCD Table 3.11-1, "Environmentally Qualified Electrical and Mechanical Equipment," (Sheet 14 of 51) to correct the location of those instruments. This change updates DCD Table 3.11-1 and addresses the spent fuel pool level instruments concern related to the Fukushima Lessons Learned report. All the aforementioned instruments currently shown in an Environmental Zone (number) 11 will change (i.e., SFS-JE-LT 019A to Environmental Zone 6, SFS-JE-LT 019B to Environmental Zone 7, and SFS-JE-LT 019C to Environmental Zone 6) in the proposed DCD Table 3.11-1.

The staff reviewed the departure that corrects the location of three spent fuel pool level instruments (i.e., Environmental Zone from 11 to 6 and 7). The staff finds that the above corrections do not result in any changes in the environmental qualification requirements (i.e., environment, "Function," "Operating Time Required," and "Qualification Program." Thus, the staff concludes the departure is acceptable.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 3.11.4:

> *<u>AP1000 COL Information Item</u>*
>
> - *STD COL 3.11-1*
>
> *The COL information item for the EQ file in Section 3.11.5 of the AP1000 DCD, states:*
>
> > *Westinghouse Electric Company LLC will act as the agent for the COL holder during the equipment design phase, equipment selection and procurement phase, equipment qualification phase, plant construction phase, and ITAAC inspection phases.*
> >
> > *The COL holder will define the process and procedures for which the equipment qualification files will be accepted from Westinghouse and how the files will be retained and maintained in an auditable format for the period that the equipment is installed and/or stored for future use in the nuclear power plant.*
>
> *This commitment was also captured as COL Action Item 3.11.2-1 in the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:*
>
> > *Pursuant to 10 CFR 50.49(j), the COL applicant shall keep the list and information in the file current and retain the file in auditable form for the entire period during which the covered item is installed in the nuclear power plant or is stored for the future use to permit verification that each item of electrical equipment important to safety (1) is qualified for its application, and (2) meets its specified performance requirements. To conform with 10 CFR 50.49, electrical equipment for PWRs referencing the AP1000 design should be qualified according to the criteria in Category I of NUREG-0588 and Revision 1 of RG 1.89.*

This commitment was also listed as COL Action Item 3.11.2-1 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:

> *The COL applicant is responsible for maintaining the equipment qualification file during the equipment selection and procurement phase.*

In STD COL 3.11-1, the applicant describes under "Combined License Information Item for Equipment Qualification File," that the COL holder is responsible for the maintenance of the equipment qualification file. The NRC staff reviewed STD COL 3.11-1 related to equipment qualification file included under Section 3.11.5 of the BLN COL. The NRC staff's evaluation is as follows.

Section 3.11.5 of the BLN COL FSAR states that the COL holder is responsible for the maintenance of the equipment qualification file upon receipt from the reactor vendor. EQ files developed by the reactor vendor are maintained as applicable for equipment and certain post-accident monitoring devices that are subject to a harsh environment. The files are maintained for the operational life of the plant.

The Environmental Qualification Master Equipment List (EQMEL) identifies the electrical and mechanical equipment or components that must be environmentally qualified for use in a harsh environment. The BLN COL FSAR states that the EQMEL and a summary of equipment qualification results are maintained as part of the equipment qualification file for the operational life of the plant. Administrative programs are in place to control revision to the EQ files and the EQMEL. When adding or modifying components in the EQ Program, EQ files are generated or revised to support qualification. The EQMEL is revised to reflect these new components. Plant modifications and design basis changes are subject to change process reviews, e.g., reviews in accordance with 10 CFR 50.59 or Section VIII of Appendix D to 10 CFR Part 52, in accordance with appropriate plant procedures. Any changes to the EQMEL that are not the result of a modification or design basis change are subject to a separate review that is accomplished and documented in accordance with plant procedures.

Based on the above, the NRC staff concludes that the COL applicant would keep the equipment qualification file and information in the file current and retain the file in an auditable form for the entire period during which the covered item is installed in the nuclear power plant or is stored for the future use to permit verification that each item of electrical equipment important to safety: (1) is qualified for its application; and (2) meets its specified performance requirements. This is consistent with 10 CFR 50.49(j) and acceptable.

In addition, the staff requested additional information related to specific implementation of this program, which is discussed below.

BLN COL FSAR Section 3.11 incorporates by reference AP1000 DCD Tier 2, Section 3.11.2.2, "Environmental Qualification of Mechanical Equipment," in the AP1000 DCD, which references Appendix 3D, "Methodology for Qualifying AP1000 Safety-Related Electrical and Mechanical Equipment." In RAI 3.11-1, the NRC staff requested that the applicant describe in more detail the EQ

Program for safety-related mechanical equipment to be used at BLN Units 3 and 4. In its response, the applicant stated that the EQ Program will be performed as described in Section 3.11 and Appendix 3D of the AP1000 DCD, by reference as stated in the BLN COL FSAR. The EQ Program will be implemented through design specifications, equipment procurement documents, and equipment qualification procedures. Equipment qualification specifications and equipment design specifications will be developed based on the AP1000 EQ requirements. The incorporation of the AP1000 DCD, Section 3.11 and Appendix 3D into the BLN COL FSAR also includes future maintenance, surveillance, and replacement activities to maintain EQ over the life of the BLN plant through operational programs and procedures. AP1000 DCD, Table 3.11-1 provides a listing of the safety-related mechanical equipment, its location, and the environment to be considered in the EQ Program. AP1000 DCD, Appendix 3D, describes: (1) qualification methodology for the critical safety-related nonmetallic sub-components; (2) thermal and radiation information for the nonmetallic components used in safety-related mechanical equipment; (3) plant normal, abnormal, and accident environmental parameters; and (4) documentation requirements. On October 14 and 15, 2008, the NRC staff conducted an onsite review of design and procurement specifications, including EQ, for pumps, valves, and dynamic restraints to be used for the AP1000 reactor at the Westinghouse offices in Monroeville, PA. The staff found that Westinghouse had included ASME Standard QME-1-2007, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants," in its design and procurement specifications for AP1000 components, including ASME QME-1, Appendix QR-B, "Guide for Qualification of Nonmetallic Parts." At the conclusion of the onsite review, the staff provided comments on the AP1000 design procurement specifications, and Westinghouse indicated that those comments would be addressed in a future revision to the specifications. The staff also identified several items that remain open from the onsite review that are specified in Section 3.9.6 of the SER on the AP1000 DCD revision. As noted in Section 3.9.6 of the BLN COL FSAR, the NRC staff documented the results of the on-site review with follow-up items in a memorandum dated November 6, 2008, (ML083110154). This is Open Item 3.11-1.

Section 3D.6.2.3, "Analysis of Safety-Related Mechanical Equipment," in the AP1000 DCD, Appendix 3D, summarizes the EQ of safety-related mechanical equipment by analysis methods, but does not discuss implementation of the EQ approach. In RAI 3.11-2, the NRC staff requested that the applicant discuss the implementation of the EQ approach, including the application of industry standards, prescribed in Section 3D.6.2.3 in Appendix 3D to Chapter 3 in the AP1000 DCD. In its response to this RAI, the applicant stated that equipment qualification specifications and equipment design specifications have been developed based on the AP1000 DCD EQ requirements. The applicant stated that these procurement documents reference ASME QME-1 and Institute of Electrical and Electronic Engineers (IEEE) Standard 323 for the EQ of active safety-related mechanical equipment. As noted above, the NRC staff conducted an onsite review of the Westinghouse design and procurement specifications for the AP1000 components on October 14 and 15, 2008. The issues in this RAI are being addressed under Open Item 3.11-1. Therefore, RAI 3.11-2 is closed.

AP1000 DCD, Appendix 3D, Section 3D.6.3, "Operating Experience in the Equipment Qualification Program," states that the COL applicant will provide documentation of the EQ methodology where seismic experience data are used. In RAI 3.11-3, the NRC staff requested that the applicant discuss the documentation of the EQ methodology where seismic experience data are used. In its response to this RAI, the applicant stated that Westinghouse would revise the AP1000 DCD to resolve this issue. Revision 17 to the AP1000 DCD, Appendix 3D, Section 3D.6.3 specifies that qualification by experience is not employed in the AP1000 equipment qualification program as a method of qualification. The applicant revised the BLN COL FSAR to reflect the revision to the AP1000 DCD. Therefore, RAI 3.11-3 is resolved.

The section titled "In-Service Vibration" in Section B.4.5, "External Stresses," in Attachment B, "Aging Evaluation Program," to Appendix 3D to Chapter 3 in the AP1000 DCD, states that inservice pipe and FIV may be significant for line-mounted equipment. As a consequence, the section states that an additional vibration aging step is included in the aging sequence. Operating experience has revealed that FIV from acoustic resonance and hydraulic loading can adversely impact safety-related mechanical equipment at nuclear power plants. The COL applicant will demonstrate the performance of this additional vibration aging step specified in the AP1000 DCD in the EQ of safety-related mechanical equipment to be used at BLN Units 3 and 4. This technical issue is addressed in Section 3.9.6 of this SER.

License Conditions

Section 3, "Operational Program Implementation," in Part 10 of the BLN COL application provides proposed license conditions for operational program implementation. One specified license condition is that the EQ Program will be implemented prior to initial fuel loading. In addition, Section 6 in Part 10 provides a proposed license condition for operational program readiness that requires the licensee to submit a schedule no later than 12 months after COL issuance that supports planning and conducting NRC inspections of operational programs with periodic updating. These license conditions are consistent with the policy established in SECY-05-0197 and are, thus, acceptable.

Resolution of Standard Content Open Item 3.11-1

Standard Content Open Item 3.11-1 resulted from the identification of items that remained open from the October 14 and 15, 2008, onsite review at Westinghouse offices of design and procurement specifications, including EQ, for pumps, valves, and dynamic restraints to be used for the AP1000 reactor. As noted in Section 3.9.6.4 of the BLN COL FSAR, the NRC staff documented the results of the onsite review with follow-up items in a memorandum dated November 6, 2008. In a letter dated December 14, 2009, the VEGP applicant stated that it had not identified any specific actions for the VEGP COL application based on the audit open items. The NRC staff's discussion of the audit of the EQ specifications, which includes the issues in RAI 3.11-2 addressed to the BLN applicant, is in NUREG-1793 and its supplements. Therefore, Standard Content Open Item 3.11-1 is resolved for the VEGP COL application.

<u>*Supplemental Review of Operational Aspects of the EQ Program*</u>

As discussed in RG 1.206 and Commission Paper SECY-05-0197, COL applicants must fully describe their operational programs to avoid the need for ITAAC regarding those programs. In addition to the initial EQ of electrical and mechanical equipment, the NRC staff reviewed the VEGP COL FSAR Section 3.11 with its incorporation by reference of the AP1000 DCD and supplemental information for operational aspects of the EQ Program. For example, AP1000 DCD Tier 2, Appendix 3D, Section 3D.7, "Documentation," states that information regarding maintenance, refurbishment, or replacement of the equipment will be included in the equipment qualification package if necessary to provide confidence in the equipment's capability to perform its safety function. Further, Section 3D.7.1, "Equipment Qualification Data Package," states that equipment qualification data packages will specify preventive maintenance that is required to support qualification or the qualified life, including maintenance or periodic activities assumed as part of the qualification program or necessary to support qualification. With respect to safety-related mechanical equipment, AP1000 DCD Tier 2, Section 3D.6.2.3.8, "Equipment Qualification Maintenance Requirements," specifies that maintenance requirements resulting from EQ activities will be based on: (1) qualification evaluation results (for example, periodic replacement of age-susceptible parts before the end of their qualified life); (2) equipment qualification-related maintenance activities derived from the qualification report; and (3) vendor recommended equipment qualification maintenance, if required, in order to maintain qualification. The staff finds that the VEGP COL applicant provides an acceptable description of the transition from the initial to the operational aspects of the EQ Program in support of the VEGP COL application through the VEGP COL FSAR with its incorporation by reference of the AP1000 DCD Tier 2, Section 3.11. The NRC staff will evaluate the implementation of the EQ Program through inspections conducted during plant construction and operation. The NRC inspection activities will include consideration of: (1) evaluation of EQ results for design life to establish activities to support continued EQ; (2) determination of surveillance and preventive maintenance activities based on EQ results; (3) consideration of EQ maintenance recommendations from equipment vendors; (4) evaluation of operating experience in developing surveillance and preventive maintenance activities for specific equipment; (5) development of plant procedures that specify individual equipment identification, appropriate references, installation requirements, surveillance and maintenance requirements, post-maintenance testing requirements, condition monitoring requirements, replacement part identification, and applicable design changes and modifications; (6) development of plant procedures for reviewing equipment performance and EQ operational activities, and for trending the results to incorporate lessons learned through appropriate modifications to the EQ Program; and (7) development of plant procedures for the control and maintenance of EQ records.

Based on the above discussion, the NRC staff finds the information added to the VEGP COL application as part of STD COL 3.11-1 to be acceptable.

<u>License Conditions</u>

- *Part 10, License Condition 3, Item G.1*

The applicant proposed a license condition providing the implementation milestone for the EQ Program.

- *Part 10, License Condition 6*

The applicant proposed a license condition to provide a schedule to support the NRC's inspection of operational programs including the EQ Program.

These license conditions are consistent with the policy established in SECY-05-0197 and are, thus, acceptable.

3.11.5 Post Combined License Activities

The license condition language in this section has been clarified from previously considered language. In a letter dated March 22, 2016 (ADAMS Accession No. ML16084A099), the applicant did not identify any concerns with the clarified license condition language. The changes do not affect the staff's above analysis of the conditions, and therefore, for the reasons discussed in the technical evaluation section above, the staff finds the following license conditions acceptable:

- License Condition (3-8) – Before initial fuel load, the licensee shall implement the Environmental Qualification Program.

- License Condition (3-9) – No later than 12 months after issuance of the COL, the licensee shall submit to the Director of NRO a schedule that supports planning for and conduct of NRC inspections of the Environmental Qualification Program. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until the Environmental Qualification Program has been fully implemented.

3.11.6 Conclusion

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to the environmental qualification program, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff compared the information in the application to relevant NRC regulations, the acceptance criteria in NUREG-0800, Section 3.11. The staff's review confirmed that the applicant has adequately addressed the COL information relating to the environmental qualification of equipment in accordance with the requirements of GDC 1, GDC 2, GDC 4, and GDC 23.

- WLS DEP 3.11-1, regarding a correction to the Environmental Zone designation for three level instruments for the spent fuel pool, is acceptable because the correction does not result in any changes in the environmental qualification requirements applicable to the instruments.

- WLS DEP 6.4-2, related to design changes affecting how the temperature and humidity in the main control room are maintained within the limits for reliable human performance, is reviewed and found acceptable by the staff in Section 21.3 of this SER.

- STD COL 3.11-1, regarding the administrative control of the EQ program for WLS, is acceptable because the requirements of 10 CFR 52.79(a)(10) and 10 CFR 52.79(a)(29)(I) are satisfied.

3.12 Piping Design

3.12.1 Introduction

This section covers the design of piping systems and supports for Seismic Category I and non-seismic systems. It also discusses the adequacy of the structural integrity, as well as the functional capability, of the safety-related piping systems, piping components, and their associated supports. The design of piping systems should ensure that they perform their safety-related functions under all postulated combinations of normal operating conditions, system operating transients, postulated pipe breaks, and seismic events. This includes pressure-retaining piping components and their supports, buried piping, instrumentation lines, and the interaction of Non-seismic Category I piping and associated supports with Seismic Category I piping and associated supports. This section covers the design transients and resulting loads and load combinations with appropriate specified design and service limits for Seismic Category I piping and piping support, including those designated as ASME Code Class 1, 2, and 3.

3.12.2 Summary of Application

WLS COL FSAR, Revision 11, Chapter 3, incorporates by reference AP1000 DCD, Revision 19, Chapter 3. NUREG-0800, Sections 3.7 and 3.9 address Section 3.12, “ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and their Associated Supports.” In addition, in WLS COL FSAR Sections 3.7 and 3.9, the applicant provided the following:

Departures

- WLS DEP 2.0-1

WLS DEP 2.0-1 provides updated seismic hazards and updated site-specific foundation response spectra (i.e., GMRS, FIRS, and NI FIRS (envelope of GMRS and FIRS)) for WLS that exceed the AP1000 CSDRS. These spectra consider the newly released model described in NUREG-2115, “Central and Eastern United States Seismic Source Characterization for Nuclear Facilities,” as well as local and regional refinements. This updated information affects the seismic inputs to the piping analysis, the methodology for which is described in the AP1000 DCD.

AP1000 COL Information Items

- STD COL 3.9-2

The applicant provided information in STD COL 3.9-2 to address COL Information Item 3.9-2, which states that design specifications and design reports for the ASME Code, Section III piping will be available for the staff review and that reconciliation of these documents is completed after construction and prior to fuel load.

- STD COL 3.9-5

The applicant provided information in STD COL 3.9-5 to address COL Information Item 3.9-5, which provides a description for pressurizer surge line monitoring.

- STD COL 3.9-7

In a November 4, 2010, letter, the applicant endorsed the April 23, 2013, letter from the VEGP applicant, which proposed to add STD COL 3.9-7 to the FSAR. This COL item provides additional information on the process to be used to complete the piping design and ITAAC added to verify the design.

License Condition

- Part 10, License Condition 2, "COL Holder Items" Item 3.9-7

In a November 4, 2010, letter, the applicant endorsed an April 23, 2010, letter from the VEGP applicant, which proposed a license condition addressing the as-designed piping analysis reconciliation schedule.

ITAAC

In a November 4, 2010, letter, the applicant endorsed the April 23, 2010, letter from the VEGP applicant, proposing ITAAC requiring the completion of a design report referencing the as-designed piping calculation packages, including the ASME Code, Section III piping analysis, support evaluations and piping component fatigue analysis for Class 1 piping using the methods and criteria outlined in AP1000 DCD Table 3.9-19.

3.12.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. The acceptance criteria associated with the relevant requirements of NRC regulations for the pipe and support analysis are given in NUREG-0800, Section 3.12.

3.12.4 Technical Evaluation

The staff reviewed WLS COL FSAR Section 3.9 and checked the referenced DCD to ensure that the combination of the DCD and the WLS COL application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the piping design review. The staff's evaluation of the information incorporated by reference in the WLS application is documented in NUREG-1793 and its supplements. Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform a technical review for each "standard issue" and use this review to evaluate subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the Reference

COL application (i.e., VEGP Units 3 and 4) were equally applicable to the WLS application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In the comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the application, as applicable) resulting from responses to RAIs.
- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed and that those changes were actualized in the COL FSAR.
- The staff verified that the site-specific differences, if any, did not adversely affect any previous relevant evaluation or conclusion.

The staff completed its review and concluded that the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this report by use of italicized, double-indented formatting.

Departures

- WLS DEP 2.0-1

The WLS COL Part 7, "Departures and Exemptions Requests," Revision 9, identifies in departure WLS DEP 2.0-1 that the WLS FIRS exceeds the AP1000 DCD CSDRS and HRHF spectra. The applicant's June 11, 2014, response to RAI 115, Question 03.12-1, shows that safety-related piping (ASME Class 1, Class 2, and Class 3) is designed for both CSDRS and HRHF spectra, as described in AP1000 DCD Appendix 3I. 10 CFR Part 50, Appendix A, GDC 2 and 10 CFR Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," state that SSCs important to safety shall be designed to withstand the effects of earthquakes. In the applicant's June 11, 2014, response to RAI 115, Question 03.12-1, Westinghouse, as documented in its report WLG-GW-GLR-815, which is cited as WLS COL FSAR Section 3.7, Reference 206, performed a detailed review of all WLS site-specific in-structure floor response spectra (FRS) X-, Y-, and Z-direction exceedances of both the AP1000 CSDRS design spectra and the HRHF design spectra. The applicant's response also stated that selected piping packages in Reference 206 showed that resulting pipe stresses due to WLS FRS are bounded by the CSDRS or HRHF design basis analysis results.

The staff reviewed Reference 206 for the effects of the WLS FIRS exceedances on piping. Forty ASME Class 1, 2 and 3 piping layout packages were reviewed. Reference 206, Figures 5.4-15 and 5.4-16, state that the horizontal WLS FRS for the hot legs and pressurizer bottom has exceedances at lower frequencies. On this basis, two piping packages inside containment were selected for analysis. These two packages are the automatic depressurization system (ADS) 4th stage east compartment and passive residual heat removal (RHR) supply and the pressurizer surge line. A third package, the spent fuel cooling system (SFS) from the auxiliary building steel containment vessel (SCV) to the auxiliary building SFS pumps was also selected for review because it is potentially sensitive to high frequency response. These three piping systems were analyzed using the PIPESTRESS computer program with seismic FRS input loadings from CSDRS, HRHF and WLS FIRS. The staff reviewed a comparison of moment stresses from the seismic analyses due to these seismic input loadings was performed. For all three piping systems, the WLS spectrum resulting stresses were bounded either by the AP1000 DCD CSDRS or the AP1000 DCD HRHF spectra. The Reference 206 report also determined that piping support loads due to WLS FRS are enveloped by CSDRS loads. Furthermore, the applicant, in a revised January 13, 2015, response to RAI 116, Question 03.12-2, the applicant stated that when piping reanalysis is required to reconcile as-built piping, in addition to the AP1000 DCD CSDRS and HRHF spectra, the as-built piping system will also be qualified using the Lee site-specific spectra to confirm that configuration changes during construction have not affected the piping system qualification for site-specific demands. The staff also notes that the applicant revised the WLS COLA, Part 2, and WLS COL FSAR, Chapter 3, Appendix 3I to identify WLS DEP 2.0-1 and provide additional information to the end of AP1000 DCD Sections 3I.1, 3I.2, 3I.3, 3I.6, 3I.6.1, 3I.6.2, 3I.6.3, 3I.6.4, and 3I.7. Specifically, the revised Appendix 3I, Section 3I.6.3 describes the applicant's site-specific analyses for piping, supporting WLS DEP 2.0-1. The staff's concluded that the additional information in AP1000 Appendix 3I.6.3 regarding piping systems is consistent with the information provided in Reference 206.

Based on its review above, the staff finds that the applicant adequately evaluated the effects of the exceedances of the WLS site-specific spectra to the AP1000 DCD CSDRS and HRHF spectra on piping and has provided reasonable assurance that the WLS site-specific seismic exceedances will not adversely affect the structural integrity of the WLS AP1000 safety related piping.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 3.12.4:

> *Due to the significant amount of new information provided by both the VEGP applicant and Westinghouse on the piping design issues since the development of the BLN SER for Section 3.12, the NRC staff decided not to use the BLN SER material as a starting point for the evaluation of these issues.*
>
> *<u>AP1000 COL Information Items</u>*
>
> - *STD COL 3.9-2*
>
> *COL Information Item 3.9-2 states that design specifications and design reports for the ASME Code, Section III piping will be available for the NRC's review and that reconciliation of the piping is completed prior to fuel load in accordance with an ITAAC in AP1000 DCD Tier 1, Section 2. The discussion on STD COL 3.9-7 below addresses design specifications and design reports.*
>
> *The staff acknowledged that an ITAAC in the AP1000 DCD Tier 1 addresses verification of this aspect of the design and that COL Information Item 3.9-2 has been addressed.*
>
> - *STD COL 3.9-5*
>
> *The staff reviewed STD COL 3.9-5 (surge line thermal monitoring) and determined that the proposed program did not provide sufficient information for the staff to determine reasonable assurance for safety. The staff issued RAI 3.12-2 to ask the applicant to provide additional information including a test abstract including stating the standard operating conditions in Chapter 14 that identifies the objective, prerequisites, test method, data required, and acceptance*

criteria for surge line thermal monitoring that complies with NRC Bulletin 88-11. In this RAI, the staff also noted that

> *For subsequent SCOLs, the design is such that assumptions are made that the layout will be the same such that monitoring of the follow-on plants is not required. However, all plants are required to comply with NRC Bulletin 88-11. Given that the heatup and cooldown procedures have not been developed and the affect on the plant, even with similar layout, will be different depending on the procedures used, subsequent plants will need to verify that they will be using the same heatup and cooldown procedures as the monitored plant to comply with NRC Bulletin 88-11.*

In a letter dated July 2, 2010, the applicant provided its response to address the staff's concern. In the response, the applicant stated that VEGP COL FSAR Section 3.9.3.1.2 would be revised to add the following paragraph:

> *Subsequent AP1000 plants (after the first AP1000 plant) confirm that the heatup and cooldown procedures are consistent with the pertinent attributes of the first AP1000 plant surge line monitoring. In addition, changes to the heatup and cooldown procedures consider the potential impact on stress and fatigue analyses consistent with the concerns of NRC Bulletin 88-11.*

In this letter, the applicant also added a new Section 14.2.9.2.22 to provide a test abstract. The test abstract included the purpose, prerequisites, general test methods, and acceptance criteria.

In a subsequent letter dated August 6, 2010, the applicant provided additional information for the location of test instruments. In the response, the applicant stated that VEGP COL FSAR Section 3.9.3.1.2 would be revised to add the following paragraph:

> *In addition to the existing permanent plant temperature instrumentation, temperature and displacement monitoring will be included at critical locations on the surge line. The additional locations utilized for monitoring during the hot functional testing and the first fuel cycle (see Subsection 14.2.9.2.22) are selected based on the capability to provide effective monitoring.*

The staff reviewed the RAI responses and concluded the position is acceptable to comply with NRC Bulletin 88-11. On this basis, the proposed program for surge line thermal monitoring is acceptable. The incorporation of the planned changes to the VEGP COL FSAR detailed in the applicant's July 2, 2010, letter will be tracked as Confirmatory Item 3.12-1.

Resolution of Standard Content Confirmatory Item 3.12-1

Confirmatory Item 3.12-1 is an applicant commitment to revise its FSAR Table 1.9-204 and Sections 3.9.3.1.2 and 3.9.8.5 for surge line monitoring testing. The *staff verified that the VEGP COL FSAR was appropriately updated. As a result, Confirmatory Item 3.12-1 is now closed.*

- *STD COL 3.9-7*

In letter dated April 23, 2010, the applicant proposes that the as-designed piping analysis is made available for NRC review. Additionally in this letter, License Condition 2, Item 3.9-7, proposed by the applicant, calls for the design to be made available for review prior to installation of the piping and adding a site-specific ITAAC in Table 3.8-# [where # is the next sequential number] of Part 10 of the VEGP COL application for verification of the ASME Code design reports. In this letter, the applicant also proposed adding Section 14.3.3 to the VEGP COL FSAR, describing the process to be followed to address closure of the piping DAC during the construction period, to complete the review of the piping design including an ITAAC to review the design, and an ITAAC to review reconciliation of the design after it is built.

The staff reviewed the applicant's proposed approach of including ITAAC for verification of the design and reconciliation of the design, and a license condition to address timing of when the initial design verification would occur. The approach, including the ITAAC and the license condition, is acceptable to the staff as it allows verification that the methodology described in the AP1000 DCD and VEGP COL FSAR and the general requirements of the ASME Code, as specified in 10 CFR 50.55a, were met.

Proposed VEGP COL FSAR Section 14.3.3.# [where # is the next sequential number] also states that "The piping design completed for the first standard AP1000 plant will be available to subsequent standard AP1000 plants under the "one issue, one review, one position" approach for closure." Westinghouse letter dated August 17, 2010, as supplemented by letter dated August 23, 2010, stated that the ASME Code Class 1, 2 and 3 piping systems will be evaluated as part of the piping DAC for hard rock site to address hard rock site seismic issue. The standard AP1000 plant will have analysis that addresses both CSDRS and HRHF GMRS effect. Therefore, the one issue, one review, one position approach applies and the staff finds this acceptable for piping analysis.

*The incorporation of the planned changes to the VEGP COL application detailed in the applicant's April 23, 2010, letter and in response to hard rock seismic issues will be tracked as **Confirmatory Item 3.12-2**.*

<u>*Resolution of Standard Content Confirmatory Item 3.12-2*</u>

Confirmatory Item 3.12-2 is an applicant commitment to revise its FSAR Table 1.8-202, Section 3.9.8.2, Section 3.9.8.7, and Section 14.3.3.3 for pipe analysis and add an ITAAC (Table 3.8-2) for verification of the ASME Code design reports. The staff verified that the VEGP COL FSAR and Part 10 of the application (ITAAC Table 3.8-2) were appropriately updated. As a result, Confirmatory Item 3.12-2 is now closed.

3.12.5 Post Combined License Activities

The license condition language in this section has been clarified from previously considered language. In a letter dated March 22, 2016 (ADAMS Accession No. ML16084A099), the applicant did not identify any concerns with the clarified license condition language. The changes do not affect the staff's above analysis of the conditions, and therefore, for the reasons discussed in the technical evaluation section above, the staff finds the following ITAAC and license condition acceptable:

- The licensee shall perform and satisfy the piping design analysis ITAAC in requiring the completion of a design report referencing the as-designed piping calculation packages, including the ASME Code, Section III piping analysis, support evaluations and piping component fatigue analysis for Class 1 piping using the methods and criteria outlined in AP1000 DCD Table 3.9-19.

- License Condition (3-10) – Before commencing installation of individual piping segments identified in AP1000 DCD, Rev. 19, Section 3.9.8.7, and connected components in their final locations in the facility, the licensee shall complete the analysis of the as-designed individual piping segments and shall inform the Director of NRO, or the Director's designee, in writing, upon the completion of these analyses and the availability of the design reports for the selected piping packages.

3.12.6 Conclusion

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to piping design, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL application is acceptable and meets NRC regulations.

- WLS DEP 2.0-1 is acceptable because the applicant provided sufficient information to satisfy the requirements of 10 CFR Part 50, Appendix A, GDC 2 of Appendix A and 10 CFR Part 50, Appendix S, "Earthquake engineering criteria for nuclear power plants."

- STD COL 3.9-2 is acceptable because it meets the general requirements of the ASME Code, as specified in 10 CFR 50.55a.

- STD COL 3.9-5 is acceptable because it is consistent with pressurizer surge line monitoring discussed in 10 CFR Part 52, Appendix D, "Design Certification Rule for the AP1000 Design."

- STD COL 3.9-7 is acceptable because it meets the general requirements of the ASME Code, as specified in 10 CFR 50.55a.

- WLS DEP 2.0-1 is acceptable because the applicant provided sufficient information to satisfy the requirements of 10 CFR Part 50, Appendix A, GDC 2 and 10 CFR Part 50, Appendix S, "Earthquake engineering criteria for nuclear power plants."

5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 Introduction

The reactor coolant system (RCS) consists of two heat transfer circuits, each with a steam generator (SG), two reactor coolant pumps (RCPs) and a single hot leg and two cold legs for circulating reactor coolant. In addition, the system includes the pressurizer, interconnecting piping/valves and instrumentation for operational control and safeguards actuation. All RCS equipment is located in the reactor containment. The RCS is designed to transfer heat generated by the reactor core, located in the reactor vessel (RV), to the secondary side of the steam generators for plant power generation.

Section 5.1 of the William States Lee III Nuclear Station (WLS) combined license (COL) Final Safety Analysis Report (FSAR), Revision 11, incorporates by reference, with no departures or supplements, Section 5.1 of Revision 19 of the AP1000 Design Control Document (DCD). The U.S. Nuclear Regulatory Commission (NRC) staff (the staff) reviewed the COL application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The staff's review confirmed that there is no outstanding issue related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," and its supplements.

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1.1 *Compliance with 10 CFR 50.55a*

5.2.1.1.1 Introduction

Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a incorporates by reference the American Society of Mechanical Engineers (ASME) *Boiler & Pressure Vessel Code* (B&PV Code) and the ASME Code for Operation and Maintenance for Nuclear Power Plants (OM Code), including Editions and Addenda for ASME Class 1, 2, and 3 components, required for component design, construction, inservice inspection (ISI), and inservice testing (IST).

AP1000 DCD Tier 2, Table 3.2-1 classifies the pressure-retaining components of the reactor coolant pressure boundary (RCPB) as ASME BPV Code, Section III, Class 1 components. These Class 1 components are designated quality group (QG) A in conformance with Regulatory Guide (RG) 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Revision 4.

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[1] See Section 1.2.2 of this safety evaluation report (SER) for a discussion of the staff's review related to verification of the scope of information to be included in a COL application that references a design certification (DC).

5.2.1.1.2 Summary of Application

WLS COL FSAR, Revision 11, Section 5.2 incorporates by reference AP1000 DCD, Revision 19, Section 5.2 with departures and/or supplements. AP1000 DCD Section 5.2 includes Section 5.2.1.1.

To address the departures and/or supplements in WLS COL FSAR Section 5.2.1.1, the applicant provided the following additional information:

AP1000 COL Information Item

- Standard (STD) COL 5.2-1

The applicant provided additional information in STD COL 5.2-1 to address COL Action Item 5.2.1.1-1 identified in NUREG-1793, Appendix F, “Combined License Action Items,” and COL Information Item 5.2-1 discussed in AP1000 DCD Section 5.2.6.1, “ASME Code and Addenda.” The portion of STD COL 5.2-1 evaluated here applies to ASME B&PV Code reconciliation. The portion applicable to Code cases is reviewed in Section 5.2.1.2 of this SER.

Specifically, WLS COL FSAR Section 5.2.1.1 states:

> If a later Code edition/addenda than the Design Certification Code edition/addenda is used by the material and/or component supplier, then a code reconciliation to determine acceptability is performed as required by the ASME Code, Section III, NCA-1140. The later Code edition/addenda must be authorized in 10 CFR 50.55a or in a specific authorization as provided in 50.55a(a)(3). Code Cases to be used in design and construction are identified in the DCD; additional Code Cases for design and construction beyond those for the design certification are not required.
>
> Inservice inspection of the reactor coolant pressure boundary is conducted in accordance with the applicable edition and addenda of the ASME Boiler and Pressure Vessel Code Section XI, as described in Subsection 5.2.4. Inservice testing of the reactor coolant pressure boundary components is in accordance with the edition and addenda of the ASME OM Code as discussed in Subsection 3.9.6 for pumps and valves, and as discussed in Subsection 3.9.3.4.4 for dynamic restraints.

5.2.1.1.3 Regulatory Basis

The regulatory basis of the information incorporated by reference in the COLA is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of NRC regulations for the ASME B&PV Code reconciliation are given in NUREG-0800, Section 5.2.1, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition.”

The applicable regulatory requirements for the staff’s review of STD COL 5.2-1 are provided in 10 CFR 50.55a, as it relates to the establishment of the minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of RCPB components and other safety-related fluid systems of pressurized-water reactor (PWR) nuclear power plants by compliance with appropriate editions of published industry codes and standards. The regulatory basis is also provided in 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 1, “Quality Standards and Records,” as it relates to requirements that nuclear power plant structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

5.2.1.1.4 Technical Evaluation

The staff reviewed WLS COL FSAR Section 5.2.1.1 and the referenced DCD to ensure that the combination of the DCD information incorporated by reference and the information in the COL application represents the complete scope of information relating to this review topic.[1] The staff’s review confirmed that the information in the application and that incorporated by reference addressed the required information relating to integrity of the RCPB. The results of the staff’s evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff’s findings on standard content that were documented in the SER for the reference COL application (Vogtle Electric Generating Plant (VEGP), Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from requests for additional information (RAIs).

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff’s review determined the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) contains evaluation material from the SER for the Bellefonte Nuclear Plant (BLN), Units 3 and 4 COL application. There was a change to the AP1000 DCD and NUREG-1793 referenced in the standard content material. This change is discussed in this SER.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 5.2.1.1.4:

AP1000 COL Information Item

- *STD COL 5.2-1*

The NRC staff reviewed STD COL 5.2-1 related to ASME BPV Code reconciliation included under Section 5.2.1.1 of the BLN COL FSAR.

The regulations in 10 CFR 50.55a(a)(3) provide requirements to authorize alternatives to the regulations in 10 CFR 50.55a, while 10 CFR 50.55a(f)(6)(i) and 10 CFR 50.55(g)(6)(i) provide requirements to grant requests for relief from impractical ASME Code requirements. In addition, NUREG-1793, Section 5.2.1.1 provides a discussion on the need for allowing changes to the ASME Code Edition and Addenda during plant construction to ensure consistency between design and construction requirements.

Section 5.2.1.1 of the NRC staff's NUREG-1793 states:

> *DCD Tier 2, Section 5.2.1.1, states that the baseline code used to support the AP1000 DCD is ASME Code, Section III, 1998 Edition, up to and including the 2000 Addenda. However, the ASME Code, Section III, 1989 Edition, 1989 Addenda will be used for Articles NB-3200, NB-3600, NC-3600, and ND-3600 in lieu of the later edition and addenda. The use of these editions and addenda meets the requirements of 10 CFR 50.55a(b) and the associated modifications in 10 CFR 50.55a(b)(1)(iii) and is, thus, acceptable. Any proposed change to the use of the ASME Code editions or addenda by a Combined License (COL) applicant will require NRC approval prior to implementation.*

The issue was also captured as COL Action Item 5.2.1.1-1 in Appendix F of NUREG-1793. The NRC staff states in Section 5.2.1.1 of NUREG-1793.

> *The COL applicant should ensure that the design is consistent with the construction practices (including inspection and examination methods) of the ASME Code edition and addenda, as endorsed in 10 CFR 50.55a. DCD Tier 2, Section 5.2.6.1, "ASME Code and Addenda," contains a commitment that the COL applicant will address consistency of the design with the construction practices (including inspection and examination methods) of the later ASME Code edition and addenda. The staff finds this to be an acceptable commitment. This is COL Action Item 5.2.1.1-1.*

Specifically, the AP1000 DCD in Section 5.2.6.1 identified a COL information item stating:

> *The Combined License applicant will address in its application the portions of later Code editions and addenda to be used to construct components that will require NRC staff review and approval. The Combined License applicant will address consistency of the design with the construction practices*

> *(including inspection and examination methods) of the later ASME Code edition and addenda added as part of the Combined License application. The Combined License applicant will address the addition of ASME Code cases approved subsequent to design certification.*

The staff reviewed conformance of BLN's resolution to COL Action Item 5.2.1.1-1 to the guidance in NUREG-0800, Section 5.2.1.1, "Compliance with the Codes and Standards Rule, 10 CFR 50.55a." ASME Code, Section III, NCA-1140, "Use of Code Editions, Addenda, and Cases," states that specific provisions within an Edition or Addenda later than those established in the design specifications may be used, provided that all the related requirements are met. NCA-1140(a)(1) also states:

> *Under the rules of this Section [Section III], the Owner or his designee shall establish the Code Edition and Addenda to be included in the Design Specifications. All items of a nuclear power plant may be constructed to a single Code Edition and Addenda, or each item may be constructed to individually specified Code Editions and Addenda.*

Accordingly, a COL applicant should establish whether it plans to use a single Code Edition and Addenda consistent with the certified design or to use individually specified Code Editions and Addenda. If individually specified Code Editions and Addenda are used, then differences between those Editions and Addenda are required to be reconciled consistent with requirements in the ASME BPV Code, Section III, NCA-1140.

The NRC staff found that Revision 0 to the BLN COL FSAR did not address NCA-1140 in describing the use of later Code Editions and Addenda. Therefore, in request for additional information (RAI) 5.2.1.1-1, the staff requested that the applicant explain the methodology for the ASME BPV Code reconciliation consistent with NCA-1140.

In its response to RAI 5.2.1.1-1 (this also applies to RAI 5.2.1.2-1 and RAI 5.2.1.1-3), the COL applicant described a revision to the FSAR to address this issue. Revision 1 to BLN COL FSAR Section 5.2.1.1, specifies that the methodology used to ensure consistency of design and construction practices when using later Section III Code Editions and Addenda would conform to the provisions of NCA-1140, and that all related requirements of the Code case(s) would be met. The use of NCA-1140 addresses the provisions to be followed for reconciliation of later Editions/Addenda of the ASME BPV Code. As a result, RAI 5.2.1.1-1 and RAI 5.2.1.2-1 are closed.

Revision 0 of the BLN COL FSAR referred to the use of ASME BPV Code, Section XI, as part of the reconciliation process if a later-Code year/Addenda than the DC Code year/Addenda is used by the material and/or component supplier. In RAI 5.2.1.1-3, the staff requested that the applicant provide justification for the use of ASME BPV Code, Section XI, which addresses ISI at *operating nuclear power plants, in the reconciliation process for new reactor designs.*

In its response to RAI 5.2.1.1-3 (referring to the response to RAI 5.2.1.1-1), the applicant noted that ASME BPV Code, Section III components are being designed using the baseline ASME BPV Code defined in DCD Section 5.2.1.1. Design specifications for component and material procurement will specify the ASME BPV Code to be used for design and construction to be that identified in the DCD. The applicant also noted that the reference in FSAR Section 5.2.1.1 to the ASME BPV Code, Section XI reconciliation process for repair and replacement was inappropriate for the original design and construction. Therefore, the applicant stated that this reference would be corrected. Revision 1 to the BLN COL FSAR in Section 5.2.1.1 removes the reference to ASME BPV Code, Section XI, and states, if a later Code Edition/Addenda than the DC Code Edition/Addenda is used by the material and/or component supplier, then a Code reconciliation to determine acceptability is performed as required by the ASME Code, Section III, NCA-1140. The staff finds that Revision 1 to the BLN COL FSAR meets the requirements of 10 CFR 50.55a. As a result, RAI 5.2.1.1-3 is closed.

Revision 0 of the BLN COL FSAR referenced Revision 16 of the AP1000 DCD. AP1000 DCD, Revision 16 required the use of the 1989 Edition, 1989 Addenda for NB-3200, NB-3600, NC-3600 and ND-3600 for construction of components and piping. In RAI 5.2.1.1-5, the NRC staff requested that the applicant identify components that are designed and constructed using the 1989 ASME BPV Code and discuss whether these components will meet the requirements of the 1998 Edition through and including the 2000 Addenda ASME BPV Code, which is the Code of record for the AP1000 DCD. In its response to RAI 5.2.1.1-5, the applicant indicated that in a letter dated May 16, 2008, Westinghouse submitted a document (APP-GW-GLE-005) to address the limitation on the use of ASME Section III Code for seismic design in accordance with 10 CFR 50.55a(b)(1)(iii) as related to the use of the above four articles. The AP1000 DCD was accordingly changed in Revision 17 to limit the use of the 1989 Edition, 1989 Addenda to piping design only. Since BLN COL FSAR, Revision 1 incorporated by reference Revision 17 of AP1000 DCD, no components will be constructed using the 1989 Edition, 1989 Addenda Code and they will be used for piping design only. As a result, RAI 5.2.1.1-5 is closed.

AP1000 DCD, Section 5.2.1.1 discusses the application of ASME BPV Code, Section III, for the design and fabrication of RCPB components. In RAI 5.2.1.1-2, the NRC staff requested that the applicant discuss the application of other sections of the ASME BPV Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) not specified in the AP1000 DCD, Section 5.2.1.1. In its response to RAI 5.2.1.1-2, provided in a letter dated July 25, 2008, the applicant discussed other sections in the AP1000 DCD and the BLN COL FSAR that reference the ASME BPV Code and the ASME OM Code. In response to RAI 5.2.1.1-2, the applicant stated that BLN COL FSAR Section 5.2.1.1 would be revised to address this issue. Revision 1 to the BLN COL FSAR in Section 5.2.1.1, specifies that ISI of the RCPB will be conducted in accordance with the applicable Edition and Addenda of the ASME

BPV Code, Section XI, as described in BLN COL FSAR Section 5.2.4, "Inservice Inspection and Testing of Class 1 Components." The BLN COL FSAR, Revision 1 also specifies that IST of the RCPB components will be performed in accordance with the applicable Edition and Addenda of the ASME OM Code as discussed in BLN COL FSAR Section 3.9.6, "Inservice Testing of Pumps and Valves," and as discussed in BLN COL FSAR Section 3.9.3.4.4, "Inspection, Testing, Repair and/or Replacement of Snubbers." Revision 1 to the BLN COL FSAR clarified the application of other sections of the ASME BPV Code and the ASME OM Code in the design, construction, and operation of BLN Units 3 and 4. As a result, RAI 5.2.1.1-2 is closed.

As discussed in NUREG-1793, use of the ASME BPV Code for the AP1000 reactor is Tier 1 information while the specific Edition and Addenda are designated Tier 2 because of the continually evolving design and construction practices (including inspection and examination techniques) of the ASME BPV Code. The NRC staff finds that the design and construction of ASME BPV Code Class 1, 2, and 3 components and their supports will conform to the appropriate ASME BPV Code Editions and Addenda and, thus, meet the relevant NRC regulations governing the use of codes and standards. The use of Editions and Addenda of the ASME BPV Code, Section III issued subsequent to the AP1000 design code of record may be used provided the Edition and Addenda are incorporated by reference in the regulations, and NRC staff approval is obtained as required for Tier 2* changes to the AP1000 DC information. Generic NRC approval of the Tier 2* changes related to use of later Editions and Addenda during construction may be obtained by a COL applicant through NCA-1140(a)(1) for components other than piping. Further, the staff finds that quality standards used will be commensurate with the importance of the safety function of all safety-related components because the ASME BPV Code, Section III that is incorporated by reference into the NRC regulations will be used by the COL licensee to ensure consistency with design, construction, and inspection requirements. The staff finds this to be an acceptable basis for satisfying the requirements of GDC 1. Finally, STD COL 5.2-1 states that any proposed alternatives to the ASME BPV Code must be authorized by the NRC pursuant to 10 CFR 50.55a(a)(3). This meets the regulations and is, therefore, acceptable.*

<u>Correction to the Standard Content Evaluation Text</u>

The section of the technical evaluation above, which discusses the Tier 2 information is no longer valid. Westinghouse, in a proposed revision of its DCD, changed the Edition and Addenda of the ASME BPV Code from a Tier 2* designation to Tier 2. This change is evaluated in a supplement to NUREG-1793.*

This change does not impact the conclusions of the BLN or VEGP evaluations.

In a February 5, 2009, letter, Duke Energy endorsed BLN standard content RAIs 05.02.01.01-01 through 05.02.01.01-05, and 05.02.01.02-01. Therefore, the above discussion related to the BLN standard content RAIs is equally applicable to WLS COL application.

5.2.1.1.5 Post Combined License Activities

There are no post COL activities related to this section.

5.2.1.1.6 Conclusion

The staff reviewed the COL application and the referenced DCD. The staff's review confirmed that the applicant adequately addressed the required information relating to codes and standards, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the requirements of 10 CFR 50.55a and GDC 1. The staff based its conclusion on the following:

- STD COL 5.2-1, as related to ASME Code reconciliation, is acceptable because the design and construction of ASME BPV Code Class 1, 2, and 3 components and their supports will conform to the appropriate ASME B&PV Code Editions and Addenda and, thus, meet the relevant NRC regulations in 10 CFR 50.55a governing the use of codes and standards. Further, the staff finds that quality standards used will be commensurate with the importance of the safety function of all safety-related components that is an acceptable basis for satisfying the requirements of GDC 1. Also, STD COL 5.2-1 states that any proposed alternatives to the ASME B&PV Code must be authorized by the NRC pursuant to 10 CFR 50.55a(a)(3).

5.2.1.2 *Applicable Code Cases (Related to RG 1.206, Section C.III.1, Chapter 5, C.I.5.2.1.2, "Compliance with Applicable ASME Code Cases")*

5.2.1.2.1 Introduction

This section addresses the ASME Code cases to be used at WLS. In general, a Code case is developed by ASME based on inquiries from the nuclear industry associated with Code clarification, modification or alternative to the Code. All Code cases will remain valid and available for use until annulled by the ASME B&PV Standards Committee. ASME Code cases acceptable to the staff are published in RG 1.84, "Design and Fabrication Code Case Acceptability, ASME Section III, Division 1"; RG 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1"; and RG 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code"; in accordance with the requirements of 10 CFR 50.55a(b)(4), 10 CFR 50.55a(b)(5) and 10 CFR 50.55a(b)(6).

5.2.1.2.2 Summary of Application

WLS COL FSAR, Revision 11, Section 5.2 incorporates by reference AP1000 DCD, Revision 19, Section 5.2 with departures and/or supplements. AP1000 DCD Section 5.2 includes Section 5.2.1.2.

WLS COL FSAR Section 5.2 does not include supplemental information in the incorporation by reference of AP1000 DCD Section 5.2.1.2. However, WLS COL FSAR Section 5.2 specifies supplementary information in STD COL 5.2-1 that relates to applicable Code cases.

To address the departure and/or supplements in WLS COL FSAR Section 5.2.1.1, the applicant provided the following additional information:

AP1000 COL Information Item

- STD COL 5.2-1

The applicant provided additional information in STD COL 5.2-1 to address COL Action Item 5.2.1.1-1 identified in NUREG-1793 and COL Information Item 5.2-1 discussed in AP1000 DCD Section 5.2.6.1, “ASME Code and Addenda.” The portion of STD COL 5.2-1 evaluated in this section applies to the applicable Code cases.

5.2.1.2.3 Regulatory Basis

The regulatory basis of the AP1000 DCD information incorporated by reference in the WLS COL application is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of NRC regulations for the applicable Code cases are given in NUREG-0800, Section 5.2.1.2.

The applicable regulatory requirements for the staff’s review of the WLS COL application are as follows.

GDC 1 in 10 CFR Part 50, Appendix A and 10 CFR 50.55a, as it relates to the establishment of the minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of nuclear power plant components, requires conformance to appropriate editions of published industry codes and standards.

As one means of meeting the applicable NRC regulations, RG 1.84 lists ASME B&PV Code, Section III Code cases oriented to design, fabrication, materials, and testing, which are acceptable with applicable conditions for implementation at nuclear power plants. RG 1.147 lists ASME B&PV Code, Section XI Code cases, which are acceptable with applicable conditions for use in the ISI of nuclear power plant components and their supports. RG 1.192 lists Code cases related to the ASME OM Code oriented to operation and maintenance of nuclear power plant components, which are acceptable with applicable conditions for implementation at nuclear power plants.

5.2.1.2.4 Technical Evaluation

The staff reviewed WLS COL FSAR Section 5.2 and the referenced DCD to ensure that the combination of the DCD information incorporated by reference in COL application and the information in the COL application represents the complete scope of information relating to this review topic.[1] The staff’s review confirmed that the information in the COL application and that incorporated by reference from the DCD addressed the required information relating to applicable Code cases. The results of the staff’s evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In NUREG-1793, Section 5.2.1.2, the NRC staff states that the COL applicant may submit, with its COL application, future Code cases that are endorsed in RG 1.84 at the time of the application, provided that they do not alter the staff's safety findings on the AP1000 certified design. The staff also states that the COL applicant should submit those Code cases that are in effect at the time of the COL application and apply to operational programs involving ISI and IST. The supplement to NUREG-1793 describes the staff's technical evaluation of modifications to the list of ASME Code cases in AP1000 DCD, Revision 19, Table 5.2-3.

The staff followed the guidance provided in NUREG-0800, Section 5.2.1.2, "Applicable Code Cases," and RG 1.206, Section C.III.1, Chapter 5, C.I.5.2.1.2, in evaluating WLS COL FSAR Section 5.2.1.2 for compliance with NRC regulations.

Section 1.2.3 of this SER provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the DC and to use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.
- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.
- The staff verified that the site-specific differences were not relevant.

The staff's review determined the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) contains evaluation material from the SER for the BLN Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 5.2.1.2.4:

> *<u>AP1000 COL Information Item</u>*
>
> - *STD COL 5.2-1*
>
> *Revision 0 to the BLN COL FSAR in Section 5.2.1.1 had referenced ASME BPV Code, Section XI, as part of the reconciliation process for the use of ASME Code cases other than those included in AP1000 DCD Table 5.2-3. In RAI 5.2.1.1-4, the staff requested that the applicant explain how this met 10 CFR 50.55a(a)(3), 10 CFR 50.55a(b)(4), 10 CFR 50.55a(b)(5), and10 CFR 50.55a(b)(6).*
>
> *In its response to RAI 5.2.1.1-4, the applicant noted that no Code cases other than those included in the DCD have been identified as necessary at this time. Code cases approved by the NRC in RG 1.147 may be used, and if so, they will*

be identified in a revision to the FSAR. The applicant also indicated that the FSAR statement regarding reconciliation of Code cases was incorrect and would be revised. Revision 1 to the BLN COL FSAR in Section 5.2.1.1 specifies that Code cases to be used in design and construction are identified in the DCD and that additional Code cases for design and construction beyond those for the DC are not required. The staff considers Revision 1 to the BLN COL FSAR Section 5.2.1.1 to be acceptable. As a result, RAI 5.2.1.1-4 is closed.

AP1000 DCD, Revision 17, Section 5.2.1.2 indicated that use of Code cases approved in revisions of the RGs issued subsequent to the DC may be used as discussed in Section 5.2.6.1 by using the process outlined for updating the ASME Code Edition and Addenda. Section 5.2.6.1 stated that the COL applicant will address in its application, the addition of ASME Code cases approved subsequent to DC. Similar to the Section III Code cases listed in DCD Table 5.2-3, in RAI 5.2.1.2-2, the staff requested that the applicant identify the ASME BPV Code, Section XI ISI and the ASME OM Code cases that are used for BLN design and construction. The applicant was also requested to confirm whether these Code cases are approved by the NRC as documented in RGs 1.147 and 1.192. If not, these Code cases must be submitted to the NRC for authorization pursuant to 10 CFR 50.55a(a)(3).

In its response to RAI 5.2.1.2-2, the applicant referred to its response to RAI 5.2.1.1-4 and noted that there are no additional Code cases used for design and construction beyond those identified in the DCD. In its RAI response, the applicant stated that the IST Program described in BLN COL FSAR Section 3.9.6 will utilize Code Case OMN- 1, Revision 1, "Alternative Rules for the Preservice and In-service Testing of Certain Electric Motor-Operated Valve Assemblies in Light Water Reactor Power Plants," which establishes alternate rules and requirements for preservice and IST to assess the operational readiness of certain motor operated valves. The staff notes that the current revision to RG 1.192 at the time of this COL review conditionally accepts the use of Code Case OMN-1, Revision 0, and does not address Revision 1 to Code Case OMN-1. The applicant will need to submit a request under 10 CFR 50.55a for authorization to apply Revision 1 to Code Case OMN-1, if RG 1.192 is not updated to accept this revision to the Code case prior to development of the IST Program for BLN. The NRC staff's review of the use of OMN-1, Revision 1, for BLN is discussed in Section 3.9.6 of this SER. In its response to RAI 5.2.1.2-2, the applicant stated that no code cases other than those included in the DCD are used for BLN and the FSAR would be revised as indicated in response to RAI 5.2.1.1-4. As noted above, Revision 1 to the BLN COL FSAR resolved RAI 5.2.1.1-4. Therefore, RAI 5.2.1.2-2 is also closed.

Based on its review, the NRC staff has determined that BLN COL FSAR Section 5.2 appropriately incorporates by reference AP1000 DCD, Section 5.2.1.2, in satisfying the NRC regulations for the design, fabrication, erection, testing, and inspection of plant SSCs commensurate with the importance of the safety function to be performed by referencing the use of accepted ASME Code cases. As a result, the staff concludes that compliance by the applicant with the provisions of the ASME Code cases accepted in RGs 1.84, 1.147, and 1.192, or individually reviewed and accepted in NUREG-1793 or its

> *supplements, will result in component quality that is commensurate with the importance of the safety functions of the components at BLN Units 3 and 4. This satisfies the requirements of GDC 1, and, therefore, is acceptable.*
>
> *AP1000 DCD, Section 5.2.6.1 states, in part, that the COL applicant will address the addition of ASME Code cases approved subsequent to the DC. As noted above, the applicant has not identified any Code cases other than those included in the AP1000 DCD as necessary at this time for the design and construction of BLN Units 3 and 4. If the applicant determines the need to apply other ASME Code cases in the future, it may apply those ASME Code cases in accordance with their acceptance in RG 1.84, RG 1.147, or RG 1.192, including any applicable conditions, or must request NRC authorization to use those Code cases.*

In a February 5, 2009, letter, Duke Energy endorsed BLN standard content RAIs 05.02.01.01-04 and 05.02.01.02-02. Therefore, the above discussion related to the BLN standard content RAIs is equally applicable to WLS COL application. In addition, the staff also verified that the equivalent changes are incorporated in the WLS COL FSAR.

5.2.1.2.5 Post Combined License Activities

There are no post-COL activities related to this section.

5.2.1.2.6 Conclusion

The staff reviewed the COL application and the referenced DCD. The staff's review confirmed that the applicant adequately addressed the required information relating to ASME Code cases, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the requirements of 10 CFR 50.55a and GDC 1, and complies with the provisions of the ASME Code cases accepted in RGs 1.84, 1.147, and 1.192. The staff based its conclusion on the following:

- STD COL 5.2-1, as related to applicable ASME Code cases, is acceptable because the staff concluded that WLS COL FSAR Section 5.2.1.2 appropriately incorporated by reference AP1000 DCD Section 5.2.1.2, to satisfy NRC regulations for the design, fabrication, erection, testing, and inspection of plant SSCs commensurate with the importance of the safety function to be performed by referencing the use of accepted ASME Code cases. As a result, the staff concludes that compliance by the applicant with the provisions of the ASME Code cases accepted in RG 1.84, RG 1.147, and RG 1.192, or individually reviewed and accepted in NUREG-1793 or its supplements, will result in component quality that is commensurate with the importance of the safety functions of the components at WLS Units 1 and 2. This satisfies the requirements of GDC 1 and, therefore, is acceptable.

5.2.1.3 *Alternate Classification*

In the standard plant design, Westinghouse applies an alternate classification for the chemical and volume control system (CVCS).

WLS COL FSAR, Revision 11, Section 5.2 incorporates by reference, with no departures or supplements, AP1000 DCD, Revision 19, Section 5.2.1.3, “Alternate Classification.” The staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The staff’s review confirmed that there is no outstanding issue related to this section. The results of the staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

5.2.2 Overpressure Protection

RCS and steam system overpressure protection during power operation is provided by the pressurizer safety valves and the steam generator safety valves, in conjunction with the action of the reactor protection system. In addition, a relief valve in the suction line of the normal residual heat removal system (RNS) provides low-temperature overpressure protection (LTOP) for the RCPB during low-temperature operation of the plant (startup, shutdown).

WLS COL FSAR, Revision 11, Section 5.2 incorporates by reference, with no departures or supplements, AP1000 DCD, Revision 19, Section 5.2.2, “Overpressure Protection.” The staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The staff’s review confirmed that there is no outstanding issue related to this section. The results of the staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

5.2.3 Reactor Coolant Pressure Boundary Materials

5.2.3.1 *Introduction*

Materials selected for 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 RCS components must maximize the corrosion resistance and fracture toughness of the components.

5.2.3.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 5.2 incorporates by reference AP1000 DCD, Revision 19, Section 5.2 of the with departures and supplements. AP1000 DCD Section 5.2 includes Section 5.2.3.

To address the departures and/or supplements in WLS COL FSAR Section 5.2.3.2.1, the applicant provided the following additional information:

<u>*Supplemental Information*</u>

- STD SUP 5.2-1

The applicant provided supplemental information to describe the monitoring program for primary water chemistry to be implemented at the plant during plant operation.

5.2.3.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of NRC regulations for the RCPB materials are identified in NUREG-0800, Section 5.2.3.

The applicable regulatory requirements for acceptance of the supplementary information on water chemistry monitoring is established in GDC 14, "Reactor Coolant Pressure Boundary," which 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.

5.2.3.4 *Technical Evaluation*

The staff reviewed WLS COL FSAR Section 5.2 and checked the referenced DCD to ensure that the combination of the DCD information incorporated in the COL application and that included in the COL application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to RCPB materials. The results of the staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff completed its review and concluded that the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference

COL application (VEGP) contains evaluation material from the SER for the BLN Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 5.2.3.4:

Supplemental Information

- *STD SUP 5.2-1*

The NRC staff reviewed the standard supplementary information on water chemistry as discussed in Section 5.2.3.2.1 of the BLN COL FSAR. In its review of the supplemental information the staff used the applicable sections of NUREG-0800 and RG 1.206 as guidance. However, Section 5.2.3 of NUREG-0800 does not directly address PWR reactor coolant chemistry, but, rather, refers the reviewer to NUREG-0800, Section 9.3.4, "Chemical and Volume Control System (PWR) Including Boron Recovery." Section 9.3.4 of NUREG-0800 recommends that the Chemical and Volume Control System (CVCS) ensure that RCS chemistry meets GDC 14, by maintaining acceptable purity levels in the reactor coolant through the removal of insoluble corrosion products and dissolved ionic material by filtration and ion exchange. In addition, Section 9.3.4 of NUREG-0800 recommends that the CVCS maintain proper RCS chemistry by controlling total dissolved solids, pH, oxygen concentration, and halide concentrations within the acceptable ranges. RG 1.206, Section C.III.1, Chapter 5, C.I.5.2.3.2 recommends that COL applications referencing PWR standard designs describe the chemistry of the reactor coolant and the additives (such as inhibitors), the water chemistry, including maximum allowable content of chloride, fluoride, sulfate, and oxygen and permissible content of hydrogen and soluble poisons, the methods to control water chemistry, including pH, the industry-recommended methodologies to be used to monitor water chemistry, and provide appropriate references. Additionally, RG 1.206, Section C.III.1, Chapter 5, C.I.5.2.3.2 also states that "this section may reference the Electric Power Research Institute (EPRI) water chemistry guidelines to support the plant-specific program. However, this section should fully describe and discuss the plant-specific water coolant chemistry control program and its compatibility with the RCPB materials."

The supplementary information in the BLN COL FSAR states that monitoring of water chemistry is implemented using the guidance of EPRI TR-1002884, "Pressurized Water Reactor Primary Water Chemistry Guidelines: Volume 1," Appendix F (Revision 5, dated October 2003). The cited appendix pertains specifically to sampling of soluble and insoluble corrosion products from the RCS. Use of this appendix is consistent with the recommendation in NUREG-0800 that the CVCS system maintains acceptable purity levels in the reactor coolant through the removal of insoluble corrosion products and dissolved ionic material by filtration and ion exchange, and must maintain proper RCS chemistry by controlling total dissolved solids, pH, oxygen concentration, and halide concentrations within the acceptable ranges. Accurate sampling of corrosion products supports this recommendation.

Appendix F of the Primary Water Chemistry Guidelines only provides a recommended methodology for sampling RCS corrosion products, and does not provide acceptance criteria or methods for reducing/controlling RCS corrosion products. Further, other primary water chemistry parameters that NUREG-0800 and RG 1.206 recommend be addressed in the FSAR are not addressed by Appendix F, such as pH, oxygen, and halide concentrations. These parameters are addressed in DCD Section 5.2.3 and DCD Table 5.2.2, which provides maximum values of primary water chemistry parameters including oxygen, pH and halide concentration for the various plant operating modes. Referencing Appendix F only of the Primary Water Chemistry Guidelines does not add any more detail or specificity for these other parameters. Therefore, in a letter dated April 10, 2008, the staff requested additional information (RAI 5.2.3-1) from the applicant to address these items.

Specifically, the NRC staff requested that the applicant explain the rationale for referencing only Appendix F to the "Pressurized Water Reactor Primary Water Chemistry Guidelines" rather than referencing the entire guidelines document.

The applicant responded to RAI 5.2.3-1, in a letter dated May 23, 2008, stating that "the AP1000 Design Control Document (DCD) describes, in Section 5.2.3.2.1, the RCS chemistry specifications and the methods to control water chemistry. In addition, DCD Table 5.2-2 summarizes these specifications for conductivity, pH, oxygen, chloride, hydrogen, suspended solids (corrosion product particulates), pH control agent, boric acid, silica, aluminum, calcium, magnesium, and zinc."

The applicant's response further stated that FSAR Section 5.2 incorporates the aforementioned DCD section by reference and refers to Appendix F of EPRI TR-1002884 as the industry recommended methodology to be used to monitor water chemistry. As noted by the question, Appendix F of the EPRI document is limited to corrosion products and as such, is insufficient to address the remaining details of the program. As such, the text of FSAR Section 5.2.3.2.1 will be revised to reference the complete EPRI document which does address the requested program attributes not covered by the DCD.

The applicant also proposed changes to the BLN COL FSAR Chapter 5, Section 5.2.3.2.1. The following information is to replace the previous supplemental information:

> *The water chemistry program is based on industry guidelines as described In EPRI TR-1002884, "Pressurized Water Reactor Primary Water Chemistry." The program includes periodic monitoring and control of chemical additives and reactor coolant impurities listed in DCD Table 5.2-2. Detailed procedures implement the program requirements for sampling and analysis frequencies, and corrective actions for control of reactor water chemistry. The frequency of sampling water chemistry varies (e.g., continuous, daily, weekly, or as needed) based on plant operating conditions and the EPRI water chemistry guidelines. Whenever corrective actions are taken to address an abnormal*

> *chemistry condition, increased sampling is utilized to verify the effectiveness of these actions. When measured water chemistry parameters are outside the specified range, corrective actions are taken to bring the parameter back within the acceptable range and within the time period specified in the EPRI water chemistry guidelines. Following corrective actions, additional samples are taken and analyzed to verify that the corrective actions were effective in returning the concentrations of contaminants.*

The staff finds the applicant's response, and the proposed COL application changes, acceptable because it meets the acceptance criteria in Section 9.3.4 of NUREG-0800 related to the evaluation of the proposed chemistry program using the latest version in the EPRI report series, "PWR Primary Water Guidelines." The staff verified that Revision 1 of the FSAR (STD SUP 5.2-1) adequately incorporates the above. As a result, RAI 5.2.3-1 is closed.

Additionally, the staff finds that the BLN FSAR meets the recommendation in RG 1.206, Section C.III.1, Chapter 5, C.I.5.2.3.2 to fully describe the primary water chemistry control program in the FSAR by referencing the most recent version of the "EPRI PWR Primary Water Guidelines" in its entirety. Although Section 5.2 of the AP1000 DCD, Revision 17, provides maximum values (and in some cases, normal ranges) for the key primary water chemistry parameters, referencing the EPRI PWR Primary Water Guidelines provides a more detailed description of the chemistry control program because various action levels (at which varying levels of corrective action are required) are specified for the key parameters for different reactor operating modes, as well as the required periodicity for sampling the various parameters.

Although the staff does not formally review or issue a safety evaluation of the revisions to the EPRI water chemistry guidelines (including the PWR Primary Water Chemistry Guidelines), the guidelines are recognized as representing industry best practices in water chemistry control. Extensive experience in operating reactors has demonstrated that following the EPRI guidelines minimizes the occurrence of corrosion related failures. Further, the EPRI guidelines are periodically revised to reflect evolving knowledge with respect to best practices in chemistry control. Therefore, the staff accepts the use of the EPRI PWR Primary Water Chemistry Guidelines as a basis for a primary water chemistry program for a COL referencing a standard reactor design.

In a February 5, 2009, letter, Duke Energy endorsed BLN standard content RAI 05.02.03-01. Therefore, the above discussion related to the BLN standard content RAIs is equally applicable to WLS COL application. In addition, the staff also verified that WLS COL FSAR, Section 5.2.3.2.1, has been adequately updated to include similar changes.

5.2.3.5 *Post Combined License Activities*

There are no post COL activities related to this section.

5.2.3.6 *Conclusion*

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant adequately addressed the required information relating to RCPB materials, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the requirements of GDC 14. The staff based its conclusion on the following:

- STD SUP 5.2-1 meets the relevant guidance in NUREG-0800, Section 9.3.4 with respect to developing a water chemistry program consistent with the latest EPRI guidelines and is acceptable. Conformance to these guidelines provides an acceptable basis to satisfy, in part, the requirements of GDC 14.

5.2.4 Inservice Inspection and Testing of Class 1 Components (Related to RG 1.206, Section C.III.1, Chapter 5, C.I.5.2.4, "Inservice Inspection and Testing of Reactor Coolant Pressure Boundary")

5.2.4.1 *Introduction*

Components that are part of the RCPB must be designed to permit periodic inspection and testing of important areas and features to assess their structural and leaktight integrity. ISI programs are based on the requirements of 10 CFR 50.55a, "Codes and Standards," in that Code Class 1 components, as defined in Section III of the ASME B&PV Code, meet the applicable inspection requirements set forth in the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components,"

5.2.4.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 5.2 incorporates by reference AP1000 DCD, Revision 19, Section 5.2 with departure and/or supplements. AP1000 DCD Section 5.2 includes Section 5.2.4.

To address departure and/or supplements in WLS COL FSAR Section 5.2.4, the applicant provided the following additional information:

AP1000 COL Information Item

- STD COL 5.2-2

The applicant provided additional information in STD COL 5.2-2 to address COL Information Item 5.2-2. The information relates to plant-specific preservice inspection (PSI) and ISI programs.

- STD COL 5.3-7

The applicant provided additional information in STD COL 5.3-7 to address COL Information Item 5.3-7. The information relates to the ISI program for the Quickloc weld buildup on the reactor vessel head.

Supplemental Information

- STD SUP 5.2-2

The applicant provided supplemental information regarding guidance for inspecting the integrity of bolting and threaded fasteners.

License Condition

- Proposed License Condition 6, regarding PSI/ISI program readiness

5.2.4.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of NRC regulations for ISI are given in NUREG-0800, Section 5.2.4.

The applicable regulatory requirements for acceptance of the resolution to COL Information Items 5.2-2 and 5.3-7 and supplementary information on ISI and testing of Class 1 components are established in GDC 32, "Inspection of Reactor Coolant Pressure Boundary," found in 10 CFR Part 50, Appendix A, as it relates to periodic inspection and testing of the RCPB, and 10 CFR 50.55a, as it relates to the requirements for inspecting and testing ASME Code Class 1 components of the RCPB.

The applicable policy for acceptance of COL Information Item 5.2-2, as it relates to fully describing an operational program, is found in SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," October 28, 2005.

5.2.4.4 *Technical Evaluation*

The staff reviewed WLS COL FSAR Section 5.2.4 and the referenced DCD to ensure that the combination of the DCD information incorporated by reference and the COL application information represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and that incorporated by reference from the DCD addressed the required information relating to the RCPB ISI and testing. The results of the staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In NUREG-1793, Section 5.2.4, the staff concluded that the AP1000 ISI program for Code Class 1 components is acceptable and meets the requirements of 10 CFR 50.55a with regard to the preservice and inservice inspectability of these components. The specific version of the ASME Code, Section XI used as the baseline Code in the AP1000 certified design is the

1998 Edition up to and including the 2000 Addenda. It should be noted that the staff did not identify any portions of the AP1000 ISI program for Class 1, 2, and 3 components that were excluded from the scope of the staff's review of the AP1000 DC (as the staff did for IST of valves in AP1000 Final Safety Evaluation Report (FSER) Section 3.9.6.4). Therefore, the staff's conclusions regarding the acceptability of the AP1000 ISI program based on the 1998 Edition up to and including the 2000 Addenda of the ASME Code, Section XI with regard to preservice and inservice inspectability of Class 1 components remain unchanged with Revision 19 of the AP1000 DCD. Accordingly, the staff's evaluation of this section focused on the acceptability of the COL applicant's supplemental information and responses to AP1000 COL information items and action items. The staff's evaluation in this section also addresses the operational program aspects of the ASME Code Class 1, 2, and 3 PSI and ISI programs.

Section 1.2.3 of this SER provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff completed its review and determined the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) contains evaluation material from the SER for the BLN Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 5.2.4.4:

> *<u>AP1000 COL Information Item</u>*
>
> - *STD COL 5.2-2*
>
> *The COL applicant added the following after the first paragraph in DCD Section 5.2.4:*
>
> > *The initial inservice inspection program incorporates the latest edition and addenda of the ASME Boiler and Pressure Vessel Code approved in 10 CFR 50.55a(b) on the date 12 months before the initial fuel load. Inservice examination of components and system pressure tests conducted during successive 120-month inspection intervals must comply with the requirements*

of the latest edition and addenda of the Code incorporated by reference in 10 CFR 50.55a(b) 12 months before the start of the 120-month inspection interval (or the optional ASEM [sic] *Code cases listed in NRC Regulatory Guide 1.147, that are incorporated by reference in 10 CFR 50.55a(b), subject to the limitations and modifications listed in 10 CFR 50.55a(b).*

10 CFR 50.55a(g) requires that inservice examinations of components and system pressure tests conducted during the initial 120-month inspection interval must comply with the requirements in the latest edition and addenda of the Code incorporated by reference in paragraph (b) of 10 CFR 50.55a on the date 12 months before the date scheduled for initial loading of fuel under a combined license under 10 CFR Part 52. The staff concludes that the supplemental information provided by the COL applicant meets the NRC's regulations and is, therefore, acceptable.

The COL applicant added the following at the end of DCD Section 5.2.4.1:

The Class 1 system boundary for both preservice and inservice inspection programs and the system pressure test program include those items within the Class 1 and Quality Group A (Equipment Class A [in accordance with] DCD Section 3.2.2 and DCD Table 3.2-3 boundary). Based on 10 CFR Part 50 and Regulatory Guide 1.26, the Class 1 boundary includes the following:

- *reactor pressure vessel;*
- *portions of the reactor system(RXS);*
- *portions of the chemical and volume control system (CVS);*
- *portion of the in core instrumentation system (IIS);*
- *portions of the passive core cooling system (PXS);*
- *portions of the reactor coolant system;*
- *portions of the normal residual heat removal system.*

Those portions of the above systems within the Class 1 boundary are those items that are part of the RCPB as defined in Section 5.2 of the Bellefonte COL FSAR.

Exclusions

Portions of the systems within the reactor coolant pressure boundary [RCPB], as defined above, that are excluded from the Class 1 boundary in accordance with 10 CFR Part 50, Section 50.55a, are as follows:

- *Those components where, in the event of postulated failure of the component during normal operation, the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only; or*

- *Components that are or can be isolated from the reactor coolant system by two valves (both closed, both open, or one closed and other open). Each open valve is capable of automatic actuation and, assuming the other valve is open, its closure time is such that, in the event of postulated failure of the component during normal reactor operation each valve remains operable and the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only.*

The NRC staff compared the proposed description of the system boundary subject to inspection and the exclusions with ASME Section XI and 10 CFR 50.55a. The staff found that the proposed system boundary and exclusions were in agreement with the ASME guidelines and regulations, and are therefore, acceptable. This portion of STD COL 5.2-2 is acceptable.

In Revision 0 of the BLN COL FSAR, the COL applicant states that NRC First Revised Order, EA-03-009, "Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors," will be used to establish the required inspections of RPV heads and associated penetration nozzles to detect primary stress corrosion cracking. In addition, the COL applicant states that ASME Code Case N-729-1 (N-729-1), "Alternative Examination Requirements for Pressurized-Water Reactor (PWR) Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds," will be used. N-729-1, as modified by the NRC staff may be used to perform the inspection of the AP1000 RPV head. Finally, a visual inspection to identify potential boric acid leaks from pressure-retaining components above the RPV head is performed by each refueling outage.

COL Information Item 5.2-2 includes a commitment that the COL applicant's PSI program will include specific preservice examinations of the RV closure head equivalent to those outlined in AP1000 DCD Tier 2, Section 5.3.4.7. The BLN COL FSAR added supplemental information to the end of Section 5.2.4.3.1, describing the design of the RV closure head as it pertains to meeting the PSI requirements. The staff could not determine from the information provided, the extent of PSI examinations. Based on the information provided by the applicant, the staff requested additional information in RAI 5.2.4-1.

In response to RAI 5.2.4-1, the COL applicant stated that the PSI related to the RV closure head and penetrations as discussed in DCD Section 5.3.4.7 includes the regions identified in the first revised order, EA-03-009. The design specification includes a requirement for PSIs consistent with the first revised order EA-03-009. As part of the RPV and integrated head package design *finalization, the RV closure head design and the design of components connected to, and in the region of, the RV closure head was reviewed.*

The COL applicant determined that the required PSI/ISI examinations can be performed as required by ASME Section III and Section XI. Based on the information provided by the COL applicant, the staff concludes that the PSI and ISI examinations will be accomplished in accordance with the first revised order, EA-03-009, ASME Sections III and XI, and are, thus, acceptable. As a result, RAI 5.2.4-1 is closed.

In Revision 1 to the BLN COL FSAR, the COL applicant states that its augmented inspection for the reactor vessel top head uses N-729-1 as modified by the NRC in the proposed rulemaking dated April 5, 2007 (72 FR 16740). The COL applicant further noted in response to RAI 5.2.4-5, that the wording in the final rule will be adopted when the final rule is issued. The final rule to amend 10 CFR 50.55a was issued on September 10, 2008 (73 FR 52730) and includes a requirement to inspect the RPV head in accordance with N-729-1 as amended by 10 CFR 50.55a(g)(6)(ii)(D). The COL applicant's methodology to inspect the RPV head in accordance with N-729-1, as amended by 10 CFR 50.55a(g)(6)(ii)(D) meets the regulations, and is therefore acceptable. The staff will verify that the next update of the BLN COL FSAR (Section 5.2.4.1) adequately incorporates reference to the final rule. This is ***Confirmatory Item 5.2-1.***

The COL applicant added the following after the second sentence of the first paragraph of DCD Section 5.2.4.4:

> *Because 10 CFR 50.55a(g)(4) requires 120-month inspection intervals, inspection Program B of IWB-2400 must be chosen. The inspection interval is divided into three periods. Each period can be extended up to one year to enable an inspection to coincide with a plant outage. The adjustment of period end dates shall not alter the rules and requirements for scheduling inspection intervals.*

RG 1.206 recommends that inspection intervals be described in comparison with the ASME Code. The information provided by the COL applicant indicated that Inspection Program B of IWB-2400 would be used over a 10-year interval. The three periods would be three, four, and three years to comprise the interval and extensions of a period may be performed up to a year to coincide with a plant outage. The staff finds that the supplemental information provided by the COL applicant meets the requirements of the ASME Code, Section XI and the guidelines of RG 1.206, and is, thus, acceptable.

The COL applicant proposed adding the following section after the last paragraph of DCD Section 5.2.4.7:

> *5.2.4.8 Relief Requests*
>
> *The specific areas where the applicable ASME Code requirements cannot be met are identified after the initial*

> *examinations are performed. Should relief requests be required, they will be developed through the regulatory process and submitted to the NRC for approval in accordance with 10 CFR 50.55a(a)(3) or 10 CFR 50.55a(g)(5). The relief requests include appropriate justifications and proposed alternative inspection methods.*

In addition to the above, the COL applicant stated at the end of Section 5.2.4.3:

> *The RPV nozzle-to-shell welds are 100 percent accessible for preservice inspection but might have limited areas that may not be accessible from the outer surface for inservice examination techniques. If accessibility is limited, an inservice inspection program relief request is prepared and submitted for review approval by the NRC.*

The information lead [sic] *the staff to believe that areas where preservice and inservice examination requirements cannot be met or where compliance with the ASME Code is impractical will result in a need for the licensee to submit a request for relief from impractical Code requirements pursuant to 10 CFR 50.55a(g)(5)(iii). This is not consistent with the regulations in 10 CFR 50.55a(g)(3)(i) which state that Class 1 components must be designed and provided with access to enable the performance of preservice and inservice examinations in accordance with the requirements of the ASME Code, Section XI. Furthermore, the information is not consistent with AP1000 DCD Section 5.2.4.2, which states that the components will be designed to eliminate any hindrances to performing preservice or inservice examinations. The only time a relief request for a newly designed system or component should occur is when the updated edition and addenda to the ASME Code, Section XI is selected 1 year before the initial fuel load date for the first 120-month ISI interval and during subsequent ISI intervals when later edition and addenda of the ASME Code, Section XI that are incorporated by reference in 10 CFR 50.55a(b) change the examination requirements or coverage.*

The staff considers accessibility to perform ISI on both sides of austenitic and dissimilar metal welds critical to making its safety determination in order to monitor structural integrity of these welds due to their history of cracking. Cracking of these welds due to primary water stress corrosion cracking (PWSCC) or intergranular stress corrosion cracking (IGSCC) is a well-known occurrence and a safety significant issue. Consequently, the NRC staff is not expecting to grant requests for relief from ISIs of these susceptible welds on the basis of design, geometry or materials of construction, since these factors can be rectified at the design stage before the plant is constructed. Based on the above discussion, the staff requested additional information from the COL applicant in RAIs 5.2.4-2 and 5.2.4-3 on accessibility for nondestructive examinations of the RV head and austenitic/dissimilar metal welds.

The COL applicant stated in its response to RAI 5.2.4-2 that as part of the design-for-inspectability process, the capability of examining the RV welds was assessed. The result was that with ISI tooling design and consideration of the

AP1000 RV design, examinations from the inside of the AP1000 pressure vessel, including examinations of the reactor nozzle-to-shell welds, can be completed without a need for the applicant to request relief from the ASME Code, Section XI examination requirements. Based on the response provided by the applicant, the staff concludes that the reactor nozzle-to-shell welds are adequately designed to enable the performance of inservice examinations in accordance with 10 CFR 50.55a(g)(3)(ii), and is, thus, acceptable. As a result, RAI 5.2.4-2 is closed.

The COL applicant stated in its response to RAI 5.2.4-3 that as part of the design-for-inspectability process, the ASME Class 1 portion of welds are designed for two-sided access for austenitic stainless steel piping welds wherever possible. Where two-sided access is not feasible, such as branch connection examination for circumferential degradation, the weld crowns are ground flush for one-sided examinations. The COL applicant stated that the examination procedures, equipment and personnel for one-sided examinations of austenitic/dissimilar metal welds would be qualified in accordance with Appendix VIII, as modified by 10 CFR 50.55a(b)(2)(xv)(A)(2) and 10 CFR 50.55a(b)(2)(xvi)(B). Based on the response provided by the applicant, in instances where one-sided examinations have to be performed for austenitic/dissimilar metal welds, the examinations will be conducted with ultrasonic systems that have demonstrated the capability to detect flaws, and is, thus, acceptable. As a result, RAI 5.2.4-3 is closed.

The COL applicant proposed adding the following section after the last paragraph of DCD Section 5.2.4.7:

> *<u>5.2.4.9 Preservice Inspection of Class 1 Components</u>*
>
> *Preservice examinations required by design specification and preservice documentation are in accordance with ASME Section III, NB-5281. Volumetric and surface examinations are performed as specified in ASME Section III, NB-5282. Components described in ASME Section III, NB-5283 are exempt from preservice examination.*

RG 1.206, Section C.III.1, Chapter 5, C.I.5.2.4 recommends that a preservice examination program that meets the standards of NB-5280 of ASME Code, Section III, Division 1, be described because it is an operational program and that the program implementation milestones should be fully described. The information indicated that preservice examinations and documentation are in accordance with ASME Code, Section III, NB-5281, and that volumetric and surface examinations are performed as specified in ASME Code, Section III, NB-5282. The information stated that components described in ASME Code, Section III, NB-5283 are exempt from preservice examination. The staff found that the information did not fully describe the preservice examination program, in scope and a level of detail, necessary for the staff to reach a reasonable assurance finding. Therefore, the staff requested additional information in RAI 5.2.4-4.

In its response to RAI 5.2.4-4, the applicant noted that AP1000 DCD Section 5.2.4.5, which is incorporated by reference in the COL FSAR, indicates PSI will meet the requirements in the ASME Code, Section XI, paragraph IWB-2200 consistent with NUREG-0800 acceptance criteria. FSAR Section 5.2.4.1 provides a discussion of the scope of the PSI and ISI programs by system. FSAR Section 5.2.4.3.1 describes the methods for examination for both PSI and ISI. FSAR Section 5.2.4.3.1 [sic] [5.2.4.3.2] *describes the qualification requirements of personnel performing ultrasonic examinations. In addition, DCD Section 5.2.4.5, incorporated by reference in the COL FSAR, indicates that PSIs of Class 1 components will meet the requirements of IWB-2200, and as indicated in the response to RAI 5.2.4-1, RV head preservice examinations are described in DCD Section 5.3.4.7, and are also incorporated by reference in the COL FSAR. These FSAR sections, combined with the DCD sections, provide a full description of the PSI program consistent with by* [sic] *SECY-05-0197. The response provided by the applicant addressed PSI program areas involving qualification requirements, scope, exemptions and methods of examination. The areas addressed meet the guidelines of Section 5.2.4 of NUREG-0800, and are therefore acceptable. Based on the information provided by the applicant, the staff concludes that the PSI program is fully described. As a result, RAI 5.2.4-4 is closed.*

The COL applicant proposed adding the following section after the last paragraph of DCD Section 5.2.4.7:

> *<u>5.2.4.10 Program Implementation</u>*
>
> *The milestones for preservice and inservice inspection program implementation are identified in Table 13.4-201.*

RG 1.206 states that the detailed procedures for performing the examinations may not be available at the time of the COL application, and the COL applicant should make a commitment to provide sufficient information to demonstrate that the procedures meet ASME Code standards. This information should be provided at a predetermined time agreed upon by both parties. In the BLN COL FSAR, Part 10, "License Conditions and ITAAC," proposed License Condition 6, "Operational Program Readiness," the COL applicant states:

> *The licensee shall submit to the appropriate Director of the NRC, a schedule, no later than 12 months after issuance of the COL, that supports planning for and conduct of the NRC inspection of the operational programs listed in the operation program FSAR Table 13.4-201. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until either the operation programs in the FSAR table have been fully implemented or the plant has been placed in commercial service.*

The staff reviewed the BLN COL FSAR Table 13.4-201, and notes that both the PSI and ISI programs are listed as operational programs required by NRC regulations. The staff concludes that the commitment under proposed License

Condition 6 meets the guidelines in RG 1.206, Section C.III.1, Chapter 5, C.I.5.2.4.1, and is, thus, acceptable.

The COL applicant proposed to add the following paragraphs at the end of Section 5.2.4.3 of the AP1000 DCD.

> ***Ultrasonic Examination of the Reactor Vessel***
>
> *Ultrasonic (UT) examination for the RPV is conducted in accordance with the ASME Code, Section XI. The RPV shell welds are designed for 100 percent accessibility for both preservice and inservice examinations. The RPV nozzle-to-shell welds are 100 percent accessible for preservice examinations but might have limited areas that may not be accessible from the outer surface for inservice examination techniques. If accessibility is limited, an inservice inspection program relief request is prepared and submitted for review approval by the NRC.*
>
> *Inner radius examinations are performed from the outside of the nozzle using several compound angle transducer wedges to obtain complete coverage of the required examination volume. Alternatively, nozzle inner radius examinations may be performed using enhanced visual techniques as allowed by 10 CFR 50.55a(b)(2)(xxi).*

The staff finds that the information provided by the COL applicant meets ASME Section XI and is in compliance with 10 CFR 50.55a. With respect to relief requests and accessibility, see the staff evaluation of BLN COL FSAR Section 5.4.2.8.

The COL applicant added the following after the first sentence of DCD Section 5.2.4.5:

> *Class 1 piping supports will be examined in accordance with ASME Section XI, IWF-2500.*
>
> *Preservice examinations required by design specifications and preservice documentation are in accordance with ASME Section III, NB-5280. Components exempt from preservice examination are described in ASME Section III, NB-5283.*

The staff finds that the information provided by the COL applicant meets ASME Section XI and is in compliance with 10 CFR 50.55a. With respect to preservice inspection, see the staff evaluation of BLN COL FSAR Section 5.4.2.9.

The COL applicant proposed adding the following after the last sentence of DCD Section 5.2.4.5:

> *The preservice examination is performed once in accordance with ASME Section XI, IWB-2200, on all of the items selected for inservice examination, with the exception of the examinations*

> *specifically excluded by ASME Section XI from preservice requirements, such ASME Section XI VT-3 examination of valve body and pump casing internal surfaces (B-L-2 and B-M-2 examination categories, respectively) and the visual VT-2 examinations for category B-P.*

The staff finds that the information provided by the COL applicant meets ASME Section XI and is in compliance with 10 CFR 50.55a. With respect to preservice inspection, see the staff evaluation of BLN COL FSAR Section 5.4.2.9.

The COL applicant proposed adding the following after the last sentence of DCD Section 5.2.4.3:

> ***Visual Examination***
>
> *Visual examination methods VT-1, VT-2, and VT-3 are conducted in accordance with ASME Section XI, IWA-2210. In addition, VT-2 examinations will meet the requirements of IWA-5240.*
>
> *Where direct VT-1 examinations are conducted without the use of mirrors or with other viewing aids, clearance is provided where feasible for the head and shoulders of a man within a working arm's length of the surface to be examined.*
>
> ***Surface Examination***
>
> *Magnetic particle (MT) and liquid penetrant (PT) examination techniques are performed in accordance with ASME Section XI, IWA-2221 and IWA-2222, respectively. Direct examination access for magnetic particle [MT] and liquid penetrant [PT] examination is the same as that required for direct visual (VT-1) examination (See Visual Examination), except that the additional access is provided as necessary to enable physical contact with the item in order to perform the examination. Remote MT and PT generally are not appropriate as a standard examination process; however, boroscopes and mirrors can be used at close range to improve the angle of vision.*
>
> ***Alternative Examination Techniques***
>
> *As provided by ASME Section XI, IWA-2240, alternative examination methods, a combination of methods, or newly developed techniques may be substituted for the methods specified for a given item in this section, provided that they are demonstrated to be equivalent or superior to the specified methods, techniques, etc., which may result in improvements in examination reliability and reductions in personnel exposure. In accordance with 10 CFR 50.55a(b)(2)(xix), IWA-2240 as written in the 1997 Addenda of ASME Section XI must be used when applying these provisions.*

> *5.2.4.3.2 Qualification of Personnel and Examination Systems for Ultrasonic Examination*
>
> *Personnel performing examinations shall be qualified in accordance with ASME Section XI, Appendix VII. Ultrasonic examination systems shall be qualified in accordance with industry accepted programs for implementation of ASME Section XI, Appendix VIII. Qualification to ASME Section XI, Appendix VIII, in compliance with the provisions of 10 CFR 50.55a is considered as a satisfactory alternative to Regulatory Guide 1.150.*

The COL applicant also proposed adding the following at the end of AP1000 DCD Section 5.2.4.6:

> *Components containing flaws or relevant conditions and accepted for continued service in accordance with the requirements of IWB-3132.4 or IWB-3142.4 are subjected to successive period examinations in accordance with the requirements of IWB-2420. Examinations that reveal flaws or relevant conditions exceeding Table IWB-3410-1 acceptance standards are extended to include additional examinations in accordance with the requirements of IWB-2430.*

10 CFR 50.55a requires that nondestructive testing procedures, methods, and techniques meet ASME Code standards, including ASME Section XI, Appendix VIII requirements for ultrasonic examinations and methodology for evaluation of flaws. The COL applicant indicated that the qualification of ultrasonic testing personnel and procedures would be in accordance with ASME Section XI, Appendices VII and VIII, respectively. Based on the information provided by the COL applicant, the staff concludes that the COL applicant referenced the appropriate sections of the ASME Code to describe visual, surface volumetric and alternative examinations.

The staff concludes that the PSI and ISI programs will conform to the guidelines and requirements provided under NUREG-0800, Order EA-03-009, and the ASME Code. Therefore, the staff finds that the COL applicant's proposed resolution to the COL information items and its supplementary information are acceptable on the basis that it meets GDC 32 of Appendix A to 10 CFR Part 50, as it relates to periodic inspection and testing of the RCPB and 10 CFR 50.55a.

Resolution of Standard Content Confirmatory Item 5.2-1

Confirmatory Item 5.2-1 required the applicant to update its FSAR to incorporate reference to the final rule. The NRC staff verified that the VEGP COL FSAR was appropriately updated to incorporate reference to 10 CFR 50.55a(g)(6)(ii)(D). As a result, Confirmatory Item 5.2-1 is now resolved.

<u>Correction of Error in the Standard Content Evaluation Text</u>

The NRC staff identified an error in the text reproduced above from the BLN SER, Section 5.2.4.4, that requires correction. The BLN SER quotes an applicant-proposed addition to its FSAR stating, in part:

> *Qualification to ASME Section XI, Appendix VIII, in compliance with the provisions of 10 CFR 50.55a is considered as a satisfactory alternative to Regulatory Guide 1.150.*

That quote is from Revision 0 of the BLN FSAR. The correct quote from Revision 1 of the BLN FSAR is:

> *Qualification to ASME Section XI, Appendix VIII, is in compliance with the provisions of 10 CFR 50.55a.*

This error does not impact the conclusions of the BLN or VEGP evaluations.

- *STD COL 5.3-7*

The NRC reviewed the applicant's proposal submitted in a letter dated August 27, 2010, to include additional information which addresses newly identified COL Information Item 5.3-7 in the AP1000 DCD. The applicant proposes to add the following item, STD COL 5.3-7, to the end of Section 5.2.4.1 of the VEGP COL FSAR:

> *The in-service inspection program is augmented to include the performance of a 100 percent volumetric examination of the weld build-up on the reactor vessel head for the instrumentation penetrations (Quickloc) conducted once during each 120-month inspection interval in accordance with the ASME Code, Section XI. The weld build-up acceptance standards are those provided in ASME Code, Section XI, IWB-3514. Personnel performing examinations and the ultrasonic examination systems are qualified in accordance with ASME Code, Section XI, Appendix VIII. Alternatively, an alternative inspection may be developed in conjunction with the voluntary consensus standards bodies (i.e., ASME) and submitted to the NRC for approval.*

The proposed information, which will augment the plant-specific ISI program to include a 100 percent volumetric examination of the weld build-up on the reactor vessel head for the instrumentation penetrations (Quickloc) conducted once during each 120-month inspection interval in accordance with the ASME Code, Section XI, is acceptable to the NRC staff because a volumetric examination ensures that potential degradation of the inside surface of the weld build-up during plant operation will be detected before it progresses through-wall. In addition, the NRC staff finds it acceptable that any alternative inspection will be submitted to the NRC for approval because it will ensure that (1) the NRC staff is informed of changes to inservice inspection requirements established in the reference design certification and (2) licensee submittals for NRC authorization to use alternatives to the regulations in 10 CFR 50.55a will be reviewed by the NRC *staff pursuant to 10 CFR 50.55a(a)(3). The NRC staff finds that this adequately addresses COL Information Item 5.3-7 and will ensure the integrity of the reactor coolant pressure boundary weld during service. The staff notes that since this information augments the ISI program, this augmentation is part of License Condition (5-1) described in SER Section 5.2.4.5. The incorporation of the changes associated with proposed STD COL 5.3-7 into a future revision of the VEGP COL FSAR is* ***Confirmatory Item 5.2-2****.*

Resolution of Standard Content Confirmatory Item 5.2-2

Confirmatory Item 5.2-2 is an applicant commitment to revise its FSAR Table 1.8-202 and Section 5.2.4.1 to address COL Information Item STD COL 5.3-7. The staff verified that the VEGP COL FSAR was appropriately revised. As a result, Confirmatory Item 5.2-2 is now closed.

The following portion of this technical evaluation section is reproduced from Section 5.2.4.4 of the BLN SER:

License Condition

- *License Condition 6, regarding PSI/ISI program details*

The BLN COL FSAR addresses implementation milestones for the PSI/ISI programs in Part 10, or [sic] *the application "Proposed License Conditions (Including ITAAC)." As discussed in Part 10, Section 6, the applicant proposes a license condition for BLN for all operational programs requiring that the licensee shall submit to the appropriate Director of the NRC, a schedule, no later than 12 months after issuance of the COL, that supports planning for and conduct of NRC inspections of operational programs. This proposed license condition is consistent with the policy established in SECY-05-0197, and is therefore acceptable.*

For PSI/ISI programs, the ASME Code, Section XI provides requirements for program implementation in Paragraph IWB-2200(a) for PSI programs and Paragraph IWA-2430(b) for ISI programs. As such, a license condition for program implementation requirements is not necessary in the BLN COL FSAR. However, submittal of the schedule for the program development is necessary to plan for and conduct NRC inspections during construction. The staff finds that the license condition complies with RG 1.206, and is therefore acceptable.

Operational programs are specific programs required by regulations. The COL application should fully describe operational programs as defined in SECY-05-0197. In addition, COL applicants should provide schedules for implementation milestones of these operational programs. The PSI and ISI programs are identified as operational programs in RG 1.206. This section of the SER addresses the PSI and ISI operational programs for ASME Code Class 1, 2, and 3 components.

As discussed in RG 1.206, a fully described PSI and ISI program should address: (1) system boundary subject to inspection; (2) accessibility; (3) examination categories and methods; (4) inspection intervals; (5) evaluation of examination *results; (6) system pressure tests; (7) Code exemptions; (8) relief requests; and (9) ASME Code cases. For BLN, the applicant incorporated by reference the PSI and ISI programs descriptions from AP1000 DCD Sections 5.2.4 and 6.6. The DCD descriptions as supplemented by the BLN COL FSAR address these nine items and therefore fully describe the PSI/ISI operational programs.*

Supplemental Information

- *STD SUP 5.2-2*

The COL applicant added the following text at the end of DCD Section 5.2.4.1:

> *The inservice inspection program, along with the boric acid corrosion control procedures, provides guidance for inspecting the integrity of bolting and threaded fasteners.*

NUREG-0800, Section 3.13, "Threaded Fasteners – ASME Code Class 1, 2, and 3," acceptance criteria states that the inspection provisions are acceptable if they conform to ASME Section XI. In addition, the staff position in Generic Letter 88-05, "Staff Position on Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," specifically recommends inspection in accordance with a boric acid corrosion control program. GL 88-05 also recommends that a boric acid control program contain four elements consisting of inspections, discovery of leak path, assessment, and follow-up inspections. In its proposed changes to Section 5.2.4.1, the COL applicant described the boric acid corrosion control procedures. The staff noted that the program description was in compliance with the four elements described under GL 88-05. Based on compliance with both ASME Section XI and staff guidance, the staff concludes that the proposed change under STD SUP 5.2-2 is acceptable.

Exception to RG 1.65

The Bellefonte FSAR Appendix 1AA provides conformance discussions for Regulatory Guides (RGs) applicable to the Bellefonte COLA. RG 1.65, "Materials and Inspections for Reactor Vessel Closure Studs," was not addressed in Revision 0 of the FSAR. In a response to the staff's RAI-1-5, the COL applicant added a conformance discussion for RG 1.65 which takes an exception to RG position C.4. The exception states:

> *ASME XI ISI criteria for reactor vessel closure stud examinations are applied in lieu of the ASME Section III, NB-2545 and NB-2546 surface examinations. The volumetric examination currently required by ASME Section XI provides improved (since 1973) detection of bolting degradation.*

The staff reviewed ASME Section XI, Table IWB-2500-1 examination requirements for the reactor vessel closure studs, Examination Category B-G-1, Item No. B 6.20. The subject table lists volumetric examination of the studs when in place. The staff finds that the COL applicant's proposed exception to

RG 1.65 is in compliance with the 1998 Edition of the ASME Code with the 2000 Addenda, and is therefore, acceptable. This portion of RAI 1-5 is closed.

In a February 5, 2009, letter, Duke Energy endorsed BLN RAIs 01-05 and 05.02.04-01 through 05.02.04-05. Therefore, the above discussion related to the BLN RAIs is equally applicable to WLS COL application. In addition, the staff also verified that the WLS COL FSAR, Section 5.2.4, has been adequately updated to include the applicable changes that were committed to in the RAI responses.

5.2.4.5 *Post Combined License Activities*

The license condition language in this section has been clarified from previously considered language. In a letter dated March 22, 2016 (ADAMS Accession No. ML16084A099), the applicant did not identify any concerns with the clarified license condition language. The changes do not affect the staff's above analysis of the conditions, and therefore, for the reasons discussed in the technical evaluation section above, the staff finds the following license condition acceptable:

- License Condition (5-1) – No later than 12 months after issuance of the COL, the licensee shall submit to the Director of NRO, or the Director's designee, a schedule for implementation of the operational programs listed in FSAR Table 13.4-201, including the associated estimated date for initial loading of fuel. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until all the operational programs listed in FSAR Table 13.4-201 have been fully implemented.

5.2.4.6 *Conclusion*

The staff reviewed the application and the referenced DCD. The staff's review confirmed that the applicant adequately addressed the required information relating to the RCPB ISI and testing, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff concludes that the information presented in the WLS COL FSAR meets the relevant acceptance criteria provided in NUREG-0800, Section 5.2.4, the policy established in SECY-05-0197, the guidelines addressed in RG 1.206, and the requirements of GDC 32, staff positions, and 10 CFR 50.55a. The staff based its conclusion on the following:

- STD COL 5.2-2, as it relates to the PSI and ISI programs, conforms to the guidelines provided under NUREG-0800, Order EA-03-009, and the ASME Code. Therefore, the staff finds that the COL applicant's resolution to the COL information items is acceptable on the basis that it meets GDC 32 in10 CFR Part 50, Appendix A, as it relates to periodic inspection and testing of the RCPB and 10 CFR 50.55a.

- STD SUP 5.2-2, as it relates to guidance for inspecting the integrity of bolting and threaded fasteners, is acceptable because it meets the relevant guidelines in the ASME Code, Section XI; NUREG-0800, Section 3.13; and GL 88-05.

- STD COL 5.3-7, as it relates to the ISI program augmentation to include 100 percent volumetric examination of the weld build-up on the reactor vessel head for the Quickloc penetrations ensures that the integrity of the RCPB weld will be maintained. Therefore, the staff finds that the applicant's information in STD COL 5.3-7 addressing COL Information Item 5.3-7 is acceptable on the basis that it meets GDC 32 in 10 CFR Part 50, Appendix A, as it relates to periodic inspection to ensure the integrity of the RCPB is maintained.

5.2.5 Detection of Leakage through Reactor Coolant Pressure Boundary (Related to RG 1.206, Section C.III.1, Chapter 5, C.I.5.2.5, "Reactor Coolant Pressure Boundary Leakage Detection")

5.2.5.1 *Introduction*

The RCPB leakage detection systems are designed to detect and, to the extent practical, identify the source of reactor coolant leakage.

5.2.5.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 5.2 incorporates by reference, with no departures or supplements, AP1000 DCD, Revision 19, Section 5.2.5.

In addition, the applicant provided the following additional information to address STD COL 5.2-3 in new WLS COL FSAR Section 5.2.5.3.5:

AP1000 COL Information Item

- STD COL 5.2-3

The applicant provided additional information in STD COL 5.2-3 to address COL Information Item 5.2-3. The information relates to prolonged low level unidentified reactor coolant leakage inside containment.

5.2.5.3 *Regulatory Basis*

The regulatory basis of the AP1000 DCD information incorporated by reference in the COL application is addressed in NUREG-1793 and its supplements.

The regulatory basis for raising the issue of prolonged low-level RCS leakage is in 10 CFR 52.79(a)(37), as it relates to "information necessary to demonstrate how operating experience insights have been incorporated into the plant design." The applicable regulatory requirements for acceptance of the resolution to COL Information Item 5.2-3 are established in GDC 30, "Quality of Reactor Coolant Pressure Boundary," as it relates to detecting RCPB leakage. The guidance for the staff's review is in RG 1.45, Revision 1, "Guidance on Monitoring and Responding to Reactor Coolant System Leakage."

5.2.5.4 *Technical Evaluation*

WLS COL FSAR, Revision 11, Section 5.2 incorporates by reference, with no departures or supplements, AP1000 DCD, Revision 19, Section 5.2.5. The staff reviewed the application and checked the applicable sections of the referenced DCD to ensure that no issue relating to this section remained for review.[1] The staff's review confirmed that there is no outstanding issue related to this section, with one exception. That exception is discussed in the standard content material below. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.
- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.
- The staff verified that the site-specific differences were not relevant.

The staff completed its review and determined that the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) contains evaluation material from the SER for the BLN Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 5.2.5.4:

> *The exception, which the NRC staff identified in its review, pertains to the operating experiences at Davis Besse concerning prolonged low-level RCS leakage. The operating experiences at Davis Besse (NRC Bulletin 2002-01) indicated that prolonged low-level unidentified reactor coolant leakage inside containment could cause corrosion and material degradation such that it could compromise the integrity of a system leading to the gross rupture of the RCPB. Therefore, pursuant to 10 CFR 52.79,(a) 37, "information necessary to demonstrate how operating experience insights have been incorporated into the plant design," the NRC staff requested additional information from both the DCD applicant (Westinghouse) and the COL applicant (Southern Nuclear Operating Company [SNC]) to address the issue of prolonged low-level RCS leakage. The NRC staff requested the COL applicant in VEGP RAI 5.2.5-1 and RAI 5.2.5-2 to address this issue as it relates to operating procedures. The NRC staff also asked Westinghouse in RAI-DCP-CN45-SBP-01 to address this issue as it related to Design Change Package (DCP) Change Number 45 for AP1000 DCD. The procedures should specify operator actions in response to prolonged low-level unidentified reactor coolant leakage conditions that exist above normal leakage rates and below the Technical Specification (TS) limits to provide*

operators sufficient time to take action before the TS limit is reached. The procedures would include identifying, monitoring, trending, and managing prolonged low-level leakage.

In a letter, dated July 29, 2010, Westinghouse responded to RAI-DCP-CN45-SBP-01 by stating that Revision 18 of the AP1000 DCD would add new COL Information Item 5.2-3, and described the COL item in Section 5.2.6.3 of the AP1000 DCD to address the prolonged low-level RCS leakage. The staff's review of DCP 45 is in Chapter 23 of a supplement to NUREG-1793.

AP1000 COL Information Item

- *STD COL 5.2-3*

In a letter, dated August 5, 2010, SNC responded to VEGP RAI 5.2.5-1 and RAI 5.2.5-2 and provided additional information in the markups of VEGP COL FSAR Table 1.8-202, Section 5.2.6.3 and Section 5.2.5.3.5 to add STD COL 5.2-3 to address the COL information item. VEGP COL FSAR Section 5.2.6.3 states that the COL item is addressed in Section 5.2.5.3.5. The proposed Section 5.2.5.3.5 reads as follows:

> *5.2.5.3.5 Response to Reactor Coolant System Leakage*
>
> *Operating procedures specify operator actions in response to prolonged low level unidentified reactor coolant leakage conditions that exist above normal leakage rates and below the Technical Specification (TS) limits to provide operators sufficient time to take action before the TS limit is reached. The procedures include identifying, monitoring, trending, and addressing prolonged low level leakage. The procedures for effective management of leakage, including low level leakage, are developed including the following operations related activities:*
>
> - *Trends in the unidentified leakage rates are periodically analyzed. When the leakage rate increases noticeably from the baseline leakage rate, the safety significance of the leak is evaluated. The rate of increase in the leakage is determined to verify that plant actions can be taken before the plant exceeds TS limits.*
>
> - *Procedures are established for responding to leakage. These procedures address the following considerations to prevent adverse safety consequence from the leakage:*
>
>   - *Plant procedures specify operator actions in response to leakage rates less than the limits set forth in the Technical Specifications. The procedures include actions for confirming the existence of a leak, identifying its source, increasing the frequency of monitoring, verifying the*

leakage rate (through a water inventory balance), responding to trends in the leakage rate, performing a walkdown outside containment, planning a containment entry, adjusting alarm setpoints, limiting the amount of time that operation is permitted when the sources of the leakage are unknown, and determining the safety significance of the leakage.

– *Plant procedures specify the amount of time the leakage detection and monitoring instruments (other than those required by Technical Specifications) may be out of service to effectively monitor the leakage rate during plant operation (i.e., hot shutdown, hot standby, startup, transients, and power operation).*

- *The output and alarms from leakage monitoring systems are provided in the main control room. Procedures are readily available to the operators for converting the instrument output to a common leakage rate. (Alternatively, these procedures may be part of a computer program so that the operators have a real-time indication of the leakage rate as determined from the output of these monitors.) Periodic calibration and testing of leakage monitoring systems are conducted. The alarm(s), and associated setpoint(s), provide operators an early warning signal so that they can take corrective actions, as discussed above, i.e., before the plant exceeds TS limits.*

- *During maintenance and refueling outages, actions are taken to identify the source of any unidentified leakage that was detected during plant operation. In addition, corrective action is taken to eliminate the condition resulting in the leakage.*

The procedures described above will be available prior to fuel load.

The staff found in the RAI response that the COL applicant committed to develop operating procedures prior to fuel load, and the procedures include identifying, monitoring, trending, and managing the prolonged low-level RCS leakage. Further, the procedures include converting the instrument output to a common leakage rate and the alarm setpoints for early warning for the operators. Therefore, the staff determined that the RAI response addressed all the questions being asked in VEGP RAI 5.2.5-1 and RAI 5.2.5-2 regarding the procedures for the prolonged low-level RCS leakage. Further, the staff reviewed the description of the procedures in the proposed VEGP COL FSAR Section 5.2.5.3.5 and determined that it is consistent with the guidance in RG 1.45, Revision 1, pertaining to managing the prolonged low-level RCS *leakage. Therefore, the staff finds that the RAI response is acceptable and concludes that GDC 30 is met based on the applicant's conformance to RG 1.45. The incorporation of the changes associated with proposed STD COL 5.2-3 into a future revision of the VEGP COL FSAR is* ***Confirmatory Item 5.2-3.***

<u>Resolution of Standard Content Confirmatory Item 5.2-3</u>

Confirmatory Item 5.2-3 is an applicant commitment to revise its FSAR Table 1.8-202 and Section 5.2.5.3.5 to address COL Information Item STD COL 5.2-3. The staff verified that the VEGP COL FSAR was appropriately revised. As a result, Confirmatory Item 5.2-3 is now closed.

In an April 25, 2011, letter, Duke Energy endorsed RAIs 05-02.05-01 and 05-02.05-02. Therefore, the above discussion related to the VEGP RAIs is equally applicable to WLS COL application. In addition, the staff also verified that the WLS COL FSAR has been adequately updated to include the applicable changes resulting from these RAIs.

5.2.5.5 *Post Combined License Activities*

For the reasons discussed in the technical evaluation above, the following FSAR commitment is identified as the responsibility of the licensee:

- Prior to initial fuel load, the operating procedures, that include identifying, monitoring, trending, and managing the prolonged low-level RCS leakage, will be developed.

5.2.5.6 *Conclusion*

The staff reviewed the application and checked the applicable sections of the referenced DCD. The staff's review confirmed that the applicant adequately addressed the required information relating to RCPB leakage detection, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the DCD information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the requirements of GDC 30. The staff based its conclusion on the following:

- STD COL 5.2-3 meets the relevant guidance in RG 1.45, Revision 1 with respect to operating procedures for the prolonged low-level RCS leakage detection. Conformance to these guidelines provides an acceptable basis to satisfy the requirements of GDC 30.

5.3 Reactor Vessel

5.3.1 Reactor Vessel Design

The RV, as an integral part of the RCPB, will be designed, fabricated, erected and tested to quality standards commensurate with the requirements set forth in 10 CFR Part 50, 10 CFR 50.55a, and GDC 1.

WLS COL FSAR, Revision 11, Section 5.3 incorporates by reference, with no departures or supplements, AP1000 DCD, Revision 19, Section 5.3.1. The staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The staff's review confirmed that there is no outstanding issue related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

5.3.2 Reactor Vessel Materials

5.3.2.1 *Introduction*

This section of this report addresses material specifications, special processes used for manufacture and fabrication of components, special methods for nondestructive examination, special controls and special processes used for ferritic steels and austenitic stainless steels, fracture toughness, material surveillance (which will be referred to as the reactor vessel surveillance capsule program (RVSP) to avoid confusion with material surveillance programs that exist in other parts of a nuclear power plant), and RV fasteners. RCS components are addressed separately in Section 5.2.3 of this SER.

5.3.2.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 5.3 incorporates by reference AP1000 DCD, Revision 19, Section 5.3 with departures and/or supplements. AP1000 DCD Section 5.3 includes Section 5.3.2.

To address the departures and/or supplements in WLS COL FSAR Section 5.3.2.6, the applicant provided the following additional information:

AP1000 COL Information Item

- STD COL 5.3-2

The applicant provided additional information in STD COL 5.3-2 to address COL Information Item 5.3-2 and COL Action Item 5.3.2.4-1 identified in NUREG-1793, Appendix F. The additional information discusses the RV material surveillance program.

License Conditions

- Part 10, License Condition 3.J.1, Reactor Vessel Material Surveillance

The COL Holder shall implement this operational program prior to initial criticality.

- Part 10, License Condition 6

The COL applicant shall provide an operational program schedule to support NRC inspections.

5.3.2.3 *Regulatory Basis*

The regulatory basis of the AP1000 DCD information incorporated by reference In the COL application is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of NRC regulations for the RV materials are given in NUREG-0800, Section 5.3.1.

The applicable regulatory requirements and guidance for acceptance of the COL information item are as follows:

- GDC 32 found in 10 CFR Part 50, Appendix A, as it relates to the RVSP
- 10 CFR 50.60, as it relates to compliance with the requirements of 10 CFR Part 50, Appendix G
- 10 CFR Part 50, Appendix G, as it relates to materials testing and acceptance criteria for fracture toughness
- 10 CFR 50.55a, as it relates to the requirements for testing and inspecting Code Class 1 components of the RCPB as specified in ASME Code, Section XI
- SECY-05-0197, as it relates to fully describing an operational program
- 10 CFR Part 50, Appendix H, as it relates to the RVSP

5.3.2.4 *Technical Evaluation*

The staff reviewed WLS COL FSAR Section 5.3.2 and the referenced DCD to ensure that the combination of the DCD information and the COL application information represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and that incorporated by reference addresses the relevant information related to RV materials. The results of the staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.
- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.
- The staff verified that the site-specific differences were not relevant.

The staff's review determined the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) contains evaluation material from the SER for the BLN Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 5.3.2.4:

> *The NRC staff reviewed conformance of Section 5.3 of the BLN COL FSAR to the guidance in RG 1.206, Section C.III.1, Chapter 5, C.I.5.3.1, "Reactor Vessel Materials." The RG 1.206 sections related to Material Specifications, Special Processes Used for Manufacturing and Fabrication, Special Methods for Nondestructive Examination, Special Controls for Ferritic and Austenitic Stainless Steels, Fracture Toughness and Reactor Vessel Fasteners all state that the COL applicants that reference a certified design do not need to include additional information. These topic areas were previously addressed in the AP1000 DCD and evaluated in NUREG-1793, Section 5.3.2. No COL action items were identified in these topic areas. The remaining topic area, RVSP, has a COL action item that must be addressed by a COL applicant.*
>
> *Appendix G to 10 CFR Part 50 specifies the fracture toughness requirements for ferritic materials of the pressure-retaining components of the RCPB. The RV beltline materials must have a Charpy Upper Shelf Energy (USE) in the transverse direction for base material and along the weld for weld material, of no less than 75 ft-lbs initially, and must maintain Charpy USE throughout the life of the vessel of no less than 50 ft-lbs. The fracture toughness tests required by ASME Code and by Appendix G to 10 CFR Part 50 provide reasonable assurance that adequate safety margins against the possibility of non-ductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the reactor coolant boundary. Appendix H to 10 CFR Part 50 presents the requirements for an RVSP to monitor the changes in the fracture toughness properties of the materials in the RV beltline region resulting from exposure to neutron irradiation and the thermal environment.*
>
> *Operational programs are specific programs required by regulations. The COL application should fully describe operational programs as defined in SECY-05-0197. In addition, COL applicants should provide schedules for implementation milestones for these operational programs. The RVSP is identified as an operational program in RG 1.206. This section of the SER addresses the adequacy of the RVSP description as it relates to meeting the requirements of Appendix H to 10 CFR Part 50.*
>
> *RG 1.206, Section C.III.1, Chapter 5, C.I.5.3.1.6, "Material Surveillance," provides guidelines for fully describing a material surveillance program. Specifically, this section states that the RVSP and its implementation must be described in sufficient detail to ensure that the program meets the requirements of Appendix H to 10 CFR Part 50.*
>
> *In addition, the application should describe the method for calculating neutron fluence for the RV beltline and the surveillance capsules. RG 1.206 lists some of the topics that should be addressed in the description of the RVSP:*
>
> - *Basis for the selection of material in the program.*

- *Number and type of specimens in each capsule.*
- *Number of capsules and proposed withdrawal schedule in compliance with the edition of American Society for Testing Materials (ASTM) E-185 Annual Book of ASTM Standards, Part 30, referenced in Appendix H to 10 CFR Part 50.*
- *Neutron flux and fluence calculations for vessel wall and surveillance specimens.*
- *Projected radiation embrittlement on vessel wall.*
- *Location of capsules, method of attachment, and provisions to ensure that capsules are retained in position throughout the vessel lifetime.*

Section 5.3.2.6 of the AP1000 DCD addresses the description of the RVSP. The DCD states that the base metal specimens are oriented both parallel and normal to the principal rolling direction of the limiting base material located in the core region of the RV. In accordance with the current DCD, there are no welds in the beltline region. Therefore, the applicant has addressed the entire beltline region in their RVSP. The DCD also addresses the number and type of specimens by meeting the ASTM E-185 requirements and describing 8 capsules, along with their proposed withdrawal schedule, that contain 72 tensile specimens, 480 Charpy V-notch specimens, and 48 compact tension specimens.

The DCD states that the neutron fluence assessments of the AP1000 RV are conducted in accordance with the guidelines that are specified in RG 1.190. The vessel fracture toughness data are given in Table 5.3-3 of the AP1000 DCD, Revision 17. The end-of-life nil-ductility reference transition temperature (RT_{NDT}) and upper shelf energy projections were estimated using RG 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” for the end-of-life neutron fluence at the ¼-thickness and inner-diameter RV locations.

Finally, BLN has addressed the location of the capsules, their method of attachment, and the provisions to ensure that capsules are retained in position throughout the vessel lifetime by referencing AP1000 DCD, Section 5.3.2.6, which states that the capsules are located in guide baskets welded to the outside of the core barrel and positioned directly opposite the center portion of the core. DCD Figure 5.3-4 shows the azimuthal locations of the capsules around the RV.

Information about the implementation of the BLN RVSP is provided in Part 10 of the BLN COL. Section 3 proposes the following license condition:

> *J. Initial Criticality – The licensee shall implement each operational program identified below prior to initial criticality.*
>
> *J.1 – Reactor Vessel Material Surveillance*

In addition, Section 6, “Operational Program Readiness,” states that the licensee will submit to the NRC a schedule, no later than 12 months after issuance of the

COL, that supports the planning for and conduct of NRC inspections of operational programs, including RVSP.

<u>AP1000 COL Information Item</u>

- *STD COL 5.3-2*

The NRC staff reviewed STD COL 5.3-2 related to the COL information item included under Section 5.3.6.2 of the BLN COL FSAR, which states:

> *The Combined License applicant will address a Reactor Vessel Reactor Material Surveillance program based on Section 5.3.2.6.*

The commitment was also captured as COL Action Item 5.3.2.4-1 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:

> *The COL applicant will provide its Reactor Vessel Material Surveillance program.*

RG 1.206 clarifies the intent of the COL information item. RG 1.206 Section C.III.1, Chapter 5, C.I.5.3.1.6, provides guidelines for addressing an RVSP. The applicant should fully describe the program and identify the implementation milestones. As previously discussed, the applicant references Section 5.3.2 of the AP1000 DCD, which addresses the topics listed in RG 1.206 that should be included in the description of the RVSP. The applicant provided License Condition 3.J.1 to implement the RVSP and License Condition 6 to support scheduling of NRC staff inspections, consistent with SECY-05-0197.

In addition, the applicant provided supplemental information in its FSAR to address COL Information Item 5.3-2 regarding the RVSP. The applicant added text between the first and second paragraphs of Section 5.3.2.6 to the AP1000 DCD, Revision 17 to reference the milestone of initial criticality for RVSP implementation. The applicant also added a new Section 5.3.2.6.3, "Report of Test Results," to the AP1000 DCD, Revision 17 to outline the reporting criteria associated with the RVSP. When each capsule is withdrawn, a summary technical report of the data required by ASTM E-185-82 and the results of the fracture toughness tests conducted on the beltline materials in the irradiated and unirradiated conditions will be submitted to the NRC within one year of the date of capsule withdrawal.

In its review of the FSAR, the staff noted that the information provided in Section 5.3.2 of the DCD, in addition to the RVSP program implementation information provided in Part 10 of the BLN COL application, meets the minimum guidelines in RG 1.206 for a description of the RVSP and its implementation. However, the staff determined that more information was needed to fully describe the RVSP in accordance with SECY-05-0197 to reach a resolution of the COL information item. A description of the process for preparing the capsule specimens must confirm that the materials selected for the capsules are samples of the same materials used in the fabrication of the RV. Therefore, the staff must receive this information before the vessel is fabricated. Other information, such as the capsule environment and the material types of the capsule specimens, *can be provided after the RV has been procured. Thus, the staff requested additional information in RAI 5.3.1-1 to complete its review.*

First, the staff requested additional information about the RVSP description. The purpose of the RVSP, as described in ASTM E-185, is to monitor radiation effects on RV materials under operating conditions. Section C.III.1, Chapter 5, C.I.5.3.1.6 of RG 1.206 states, "because the material surveillance program is an operational program, as discussed in SECY-05-0197, the applicant must describe the program and its implementation in sufficient scope and level of detail for the staff to make a reasonable assurance finding on its acceptability." The NRC staff recognizes that certain information about the program, such as actual material properties of the RV, is not currently known, but in order to complete its review of the adequacy of the RVSP, the staff requested that the applicant describe its process for preparing the capsule specimens. This description should confirm that the materials selected for the capsules are samples of those materials most likely to limit the operation of the RV.

Secondly, the staff requested additional information about the RVSP. The COL applicant must fully describe its RVSP to ensure that it meets ASTM E-185 and other requirements listed in 10 CFR Part 50, Appendix H. Specifically, the NRC staff requested detailed information on the RVSP associated with the AP1000 design, including, but not limited to, the capsule environment and the material types of the capsule specimens.

In RAI 5.3.1-1, the staff requested that the applicant describe the process for preparing the capsule specimens and to include detailed information on the capsule environment and material types of the capsule specimens. The applicant responded with a detailed description of the capsule specimen preparation process to be incorporated into the next revision of the BLN COL FSAR. The applicant also stated that the capsule environment and the material types of the capsule specimens are addressed in AP1000 DCD, Section 5.3.2.6 which is incorporated by reference.

The staff finds that the response to RAI 5.3.1-1 is acceptable, provided that the BLN COL FSAR is revised as stated by the applicant, and that the applicant confirms the staff's understanding that the surveillance capsules are backfilled with inert gas. Therefore, the staff identifies ***Confirmatory Item 5.3-1*** *to confirm that the BLN COL FSAR is revised as stated, and to confirm the staff's understanding that the surveillance capsules are backfilled with inert gas.*

Generic Letter 92-01

Generic Letter (GL) 92-01, "Reactor Vessel Structural Integrity," addressed NRC concerns regarding compliance with the requirements of Appendices G and H to 10 CFR Part 50, which address fracture toughness requirements and RVSP requirements, respectively. Specifically, NRC had concerns about Charpy USE predictions for end-of-life for the limiting beltline weld and the plate or forging, RVs constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition, and use of RG 1.99, Revision 2, to estimate the embrittlement of the materials in the RV beltline. These topics have been addressed in the

AP1000 DCD, Revision 17, which is incorporated by reference in the BLN COL FSAR.

The AP1000 DCD, Revision 17, also states that end-of-life RT_{NDT} and USE projections were estimated using RG 1.99. The construction of the RV to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition is not a concern for new reactors, including BLN. In the BLN COL FSAR Section 5.3.2.6.3, the applicant provides additional information, which states that when each capsule is withdrawn, a summary technical report of the data required by ASTM E-185-82 and the results of the fracture toughness tests conducted on the beltline materials in the irradiated and unirradiated conditions will be submitted to the NRC within one year of the date of capsule withdrawal.

On the basis of the information discussed above, the NRC staff concludes that the applicant has adequately addressed the issues in GL 92-01.

<u>*Resolution of Standard Content Confirmatory Item 5.3-1*</u>

The NRC staff verified that the VEGP FSAR was updated to include a detailed description of the capsule specimen preparation process and to document that the surveillance capsules are backfilled with inert gas. As a result, Confirmatory Item 5.3-1 is resolved.

In an April 25, 2011, letter, Duke Energy endorsed VEGP RAI 05-03.01-01. Therefore, the above discussion related to the VEGP RAI is equally applicable to WLS COL application. In addition, the staff also verified that the WLS COL FSAR has been adequately updated to include the applicable changes resulting from this RAI.

5.3.2.5 *Post Combined License Activities*

For the reasons discussed in the technical evaluation section above, the staff finds the following license conditions related to the RV Material Surveillance program acceptable:

- License Condition (5-2) – The licensee shall implement the RV Material Surveillance program prior to initial criticality.

- License Condition (5-3) – No later than 12 months after issuance of the COL, the licensee shall submit to the Director of NRO a schedule that supports planning for and conduct of NRC inspections of the RV Material Surveillance program. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until the RV Material Surveillance program has been fully implemented.

5.3.2.6 *Conclusion*

The staff reviewed the application and the applicable sections of the referenced DCD. The staff's review confirmed that the applicant adequately addressed the required information relating to RV materials, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the

information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff concludes that the information presented in the WLS COL FSAR is acceptable and meets the relevant regulatory guidance provided in NUREG-0800, Section 5.3.1 and RG 1.206, the policy established in SECY-05-0197, and the requirements of 10 CFR Part 50, Appendices G and H. The staff based its conclusion on the following:

- STD COL 5.3-2, relating to the RV material surveillance program, is acceptable because the program is consistent with the relevant guidelines addressed in NUREG-0800, Section 5.3.1 and in RG 1.206, Section C.III.1, Chapter 5, C.I.5.3.1. Conformance to these guidelines provides an acceptable basis to satisfy, in part, the requirements of to 10 CFR Part 50, Appendices G and H.

5.3.3 Pressure Temperature Limits (Related to RG 1.206, Section C.III.1, Chapter 5, C.I.5.3.2, "Pressure-Temperature Limits, Pressurized Thermal Shock, and Charpy Upper-Shelf Energy Data and Analyses")

5.3.3.1 *Introduction*

Pressure Temperature (P-T) limits are required as a means of protecting the RV during startup and shut down to minimize the possibility of fast fracture. The methods outlined in ASME Code, Section XI, Appendix G are employed in the analysis of protection against nonductile failure. Beltline material properties degrade with radiation exposure and this degradation is measured in terms of the adjusted reference temperature, which includes a reference nil-ductility temperature shift, initial RT_{NDT}, and margin.

5.3.3.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 5.3 incorporates by reference AP1000 DCD, Revision 19, Section 5.3 with departures and/or supplements. AP1000 DCD Section 5.3 includes Section 5.3.3.

To address departures and/or supplements in WLS COL FSAR Section 5.3.6.1, the applicant provided the following additional information:

AP1000 COL Information Item

- STD COL 5.3-1

The applicant provided additional information in STD COL 5.3-1 to address COL Information Item 5.3-1 of the AP1000 DCD and COL Action Item 5.2.2.2-1 in NUREG-1793. The information relates to plant-specific P-T curves.

Supplemental Information

- STD SUP 5.3-1

The applicant provided supplemental information related to development of operating procedures as required by Technical Specification (TS) 5.6.6.

<u>*License Condition*</u>

- Part 10, License Condition 2, Item 5.3-1

The license condition related to COL Information Item 5.3-1 sets the implementation milestone for development of plant-specific P-T curves.

5.3.3.3 *Regulatory Basis*

The regulatory basis of the AP1000 DCD information incorporated by reference in the COL application is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of NRC regulations for P-T limits are given in NUREG-0800, Section 5.3.2.

5.3.3.4 *Technical Evaluation*

The staff reviewed WLS COL FSAR Section 5.3.3 and the referenced DCD to ensure that the combination of the DCD information incorporated by reference and the information in the COL application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the COL application and that incorporated by reference addresses the required information relating to P-T limits. The results of the staff's evaluation of the DCD information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff's review determined the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) contains evaluation material from the SER for the BLN Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 5.3.3.4:

AP1000 COL Information Item

- *STD COL 5.3-1*

The NRC staff reviewed STD COL 5.3-1 related to COL Information Item 5.3-1 included under Section 5.3.6.1 of the COL FSAR. The applicant proposes to replace the text in AP1000 DCD Section 5.3.6.1 with the following:

> *The pressure-temperature curves shown in DCD Figures 5.3-2 and 5.3-3 are generic curves for AP1000 reactor vessel design, and they are limiting curves based on copper and nickel material composition. Plant-specific curves will be developed based on material composition of copper and nickel. Use of plant-specific curves will be addressed during procurement and fabrication of the reactor vessel. As noted in the bases to Technical Specification 3.4.14, use of plant-specific curves requires evaluation of the LTOP system. This includes an evaluation of the setpoint pressure for the RNS relief valve to determine if the setpoint pressure needs to be changed based on plant-specific pressure-temperature curves. The development of the plant-specific curves and evaluation of the setpoint pressure are required prior to fuel load.*

In addition, in Section 5.3.3.2 of NUREG-1793, the staff identified related COL Action Item 5.2.2.2-1 in which the COL applicant will address the use of plant-specific curves during procurement of the RV.

The COL applicant stated that the P-T limits shown in DCD Figures 5.3-2 and 5.3-3 are generic curves for AP1000 RV design, and they are limiting curves based on copper and nickel material composition. The applicant committed to provide P-T limits using the plant-specific material composition after the combined license is issued and when the RV is procured. The applicant also stated that the development of the plant-specific P-T limits is required prior to fuel load. The staff found that a more specific implementation milestone for completing the plant-specific P-T limits was needed. Thus, the following additional information was requested.

In RAI 5.3.2-1, the staff noted Westinghouse's plan to: a) submit a generic PTLR [pressure temperature limits report] for the AP1000 RV using the bounding properties for NRC staff review and approval; and b) update the AP1000 DCD to include the use of the generic AP1000 PTLR by all COL applicants. The NRC staff requested that Part 10 of the BLN COL, proposed license conditions, Section 2, COL holder items, and COL Information Item 5.3-1 be revised by adding the following statement:

> *The COL Holder shall update the P/T limits using the PTLR methodologies approved in the AP1000 DCD, and using the plant-specific material properties. The COL Holder will inform the NRC of the updated P/T limits.*

The approach described above is consistent with that used for all operating reactors where licensees using PTLRs (reference: GL 96-03) inform the NRC staff of any subsequent change in P-T limits with no NRC approval necessary when there are no changes to the approved PTLR methodology. Subsequently, in a letter dated May 30, 2008, Westinghouse submitted a generic PTLR for AP1000 plants. The NRC staff reviewed the PTLR and approved its use for AP1000 RVs in a safety evaluation (ML083470258) dated December 30, 2008.

In response to RAI 5.3.2-1, the applicant proposed to modify the COL application Part 10, Proposed Combined License Conditions, Section 2, COL Holder Item 5.3-1. Accordingly, the modified license condition states, "The COL Holder shall update the P/T limits using the PTLR methodologies approved in the AP1000 DCD using plant-specific material properties or confirm that the reactor vessel material properties meet the specifications and use the Westinghouse generic PTLR curves."

The staff finds that the applicant's modification to the proposed license condition is adequate and the staff verified that the revision to Part 10 of the application incorporates the above. As a result, RAI 5.3.2-1 is closed.

Supplemental Information

- *STD SUP 5.3-1*

Development of plant operating procedures as required by TS 5.6.6 ensures that P-T limits are adhered to during normal and abnormal operating conditions and system tests and is therefore, acceptable.

In a February 5, 2009, letter, Duke Energy endorsed BLN RAI 05.03.02-01. Therefore, the above discussion related to the BLN RAI is equally applicable to WLS COL application. In addition, the staff also verified that the WLS COL FSAR, Section 5.3 and WLS COL Part 10, have been adequately updated to include the applicable changes resulting from this RAI.

5.3.3.5 *Post Combined License Activities*

For the reasons discussed in the technical evaluation section above, the staff finds the following license condition related to P-T limits acceptable:

- License Condition (5-4) – Before initial fuel load, the licensee shall update the P-T limits using the pressure-temperature limits report (PTLR) methodologies approved in the AP1000 DCD using the plant-specific material properties or confirm that the RV material properties meet the specifications and use the Westinghouse generic PTLR curves.

5.3.3.6 *Conclusion*

The staff reviewed the application and the referenced DCD. The staff's review confirmed that the applicant adequately addressed the required information relating to P-T limits, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the regulatory basis addressed in NUREG-1793. Specifically, the relevant regulatory basis includes NUREG-0800, Section 5.3.2, GL 96-03, and 10 CFR Part 50, Appendix G. The staff based its conclusion on the following:

- STD COL 5.3-1, relating to plant-specific P-T curves, is acceptable because the program is consistent with the guidelines addressed in NUREG-0800, Section 5.3.2 . Conformance to these guidelines provides an acceptable basis to satisfy in part, the requirements of 10 CFR Part 50, Appendix G.

- STD SUP 5.3-1, relating to development of operating procedures, is acceptable because it ensures that P-T limits are adhered to during normal and abnormal operating conditions and system tests.

5.3.4 Reactor Vessel Integrity (Related to RG 1.206, Section C.III.1, Chapter 5, C.I.5.3.3 “Reactor Vessel Integrity”)

5.3.4.1 *Introduction*

AP1000 DCD Section 5.3.4 describes the RV integrity. The RV is the RCPB used to support and enclose the reactor core. The RV provides flow direction with the reactor internals through the core and maintains a volume of coolant around the core. The vessel is fabricated by welding together the lower head, the transition ring, the lower shell, and the upper shell. The upper shell contains the penetrations from the inlet and outlet nozzles and direct vessel injection nozzles.

As part of the RV integrity, AP1000 DCD Section 5.3.4 also addresses the pressurized thermal shock (PTS) for the PWR RV. PTS events are potential transients in a PWR RV that can cause severe overcooling of the vessel wall, followed by immediate repressurization. The thermal stresses, caused when the inside surface of the RV cools rapidly, combined with high-pressure stresses, will increase the potential for fracture if a flaw is present in a low-toughness material. The materials most susceptible to PTS are the materials in the RV beltline where neutron radiation gradually embrittles the material over time.

5.3.4.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 5.3 incorporates by reference AP1000 DCD, Revision 19, Section 5.3 with departures and/or supplements. AP1000 DCD Section 5.3 includes Section 5.3.4.

To address departures and/or supplements in WLS COL FSAR Section 5.3.6, the applicant provided the following additional information:

<u>*AP1000 COL Information Item*</u>

- STD COL 5.3-4

The applicant provided additional information in STD COL 5.3-4 to address COL Information Item 5.3-4 and related COL Action Item 5.3.4.3-1. The applicant proposed to verify the plant-specific beltline material properties consistent with the requirements in AP1000 DCD

Section 5.3.3.1 and AP1000 DCD Tables 5.3-1 and 5.3-3 prior to fuel load. The applicant also proposed in STD COL 5.3-4 to perform a PTS evaluation based on as procured RV material data and the projected neutron fluences for the plant design objective of 60 years.

License Condition

- Part 10, License Condition 2, Item 5.3-4

The milestone for the implementation of the proposed actions related to RV material properties will be prior to initial fuel load.

5.3.4.3 *Regulatory Basis*

The regulatory basis of the DCD information incorporated by reference in the COL application is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of NRC regulations for the RV integrity are given in NUREG-0800, Section 5.3.3.

In addressing the COL information item, PWRs are required, in part, to have the pressurized thermal shock reference temperature (RT_{PTS}), evaluated for the end-of-life fluence for each of the RV beltline materials in accordance with requirements of 10 CFR 50.61.

5.3.4.4 *Technical Evaluation*

The staff reviewed WLS COL FSAR Section 5.3.4 and checked the applicable sections of the referenced DCD to ensure that the combination of the DCD information and that in the COL application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and that incorporated by reference from the DCD addresses the required information relating to RV integrity. The results of the staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff's review determined the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) contains evaluation material from the SER for the BLN Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 5.3.4.3:

AP1000 COL Information Item

- *STD COL 5.3-4*

The NRC staff reviewed STD COL 5.3-4 related to COL Information Item 5.3-4 and related COL Action Item 5.3.4.3-1. The applicant proposed to verify the plant-specific beltline material properties consistent with the requirements in DCD Section 5.3.3.1 and DCD Tables 5.3-1 and 5.3-3 prior to fuel load. The applicant also proposed in STD COL 5.3-4 to perform a PTS evaluation based on as procured RV material data and the projected neutron fluences for the plant design objective of 60 years.

License Condition

- *Part 10, License Condition 2, Item 5.3-4*

In response to the COL information item, the applicant proposed a license condition (Part 10, Item 2, COL Information Item 5.3-4) that a plant-specific PTS evaluation would be performed by the COL holder using as-procured RV material data and submitted for NRC review prior to initial fuel loading.

The as-procured RV material properties will be available to the COL holder after the acceptance of the RV. In order to provide sufficient time for NRC review of the PTS evaluation using the as-procured RV material properties as required by 10 CFR 50.61, the staff requested a more specific and timely milestone for submitting the PTS evaluation to the NRC be established. Therefore, the staff requested that the proposed license condition for COL Information Item 5.3-4 be revised to state that, within a reasonable period of time following acceptance of the RV, the COL holder submit to the NRC staff the plant-specific PTS evaluation, for example, one year after the acceptance of the RV. This was identified in RAI 5.3.3-1. In response to RAI 5.3.3-1, the applicant proposed that the licensee shall submit to the appropriate Director of the NRC, a schedule, no later than 12 months after the issuance of the COL, that supports planning for and conduct of NRC inspections of operational programs listed in the operational program FSAR Table 13.4-201. This schedule shall include a submittal schedule for the RV pressurized thermal shock evaluation at least 18 months prior to initial fuel load. Accordingly, the applicant will revise the COL application, Part 10, proposed License Condition 6.

The staff finds that Revision 1 of the application incorporates the proposed change to the proposed License Condition 6, and therefore the applicant's response to COL Information Item 5.3-4 meets the implementation requirements of 10 CFR 50.61, and is therefore acceptable. As a result, RAI 5.3.3-1 is closed.

In a February 5, 2009, letter, Duke Energy endorsed BLN RAI 05.03.03-01. Therefore, the above discussion related to RAI 05.03.03-01 is equally applicable to the WLS COL application. In addition, the staff also verified that the WLS COL FSAR, Section 5.3 and WLS COL Part 10 have been adequately updated to include the applicable changes resulting from this RAI response.

5.3.4.5 *Post Combined License Activities*

The license condition language in this section has been clarified from previously considered language. In a letter dated March 22, 2016 (ADAMS Accession No. ML16084A099), the applicant did not identify any concerns with the clarified license condition language. The changes do not affect the staff's above analysis of the conditions, and therefore, for the reasons discussed in the technical evaluation section above, the staff finds the following license condition acceptable:

- License Condition (5-5) – Before initial fuel load, the licensee shall verify that plant-specific belt line material properties are consistent with the properties given in AP1000 DCD Rev. 19, Section 5.3.3.1 and Tables 5.3-1 and 5.3-3. The verification must include a pressurized thermal shock (PTS) evaluation based on as-procured reactor vessel material data and the projected neutron fluence for the plant design objective. Submit this PTS evaluation report to the Director of NRO, or the Director's designee, in writing, at least 18 months before the latest date set forth in the schedule for completing the inspections, tests, and analyses in the ITAAC submitted in accordance with 10 CFR 52.99(a).

5.3.4.6 *Conclusion*

The staff reviewed the COL application and checked the applicable sections of the referenced DCD. The staff's review confirmed that the applicant adequately addressed the required information relating to RV integrity, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the information presented in the WLS COL FSAR meets the relevant acceptance criteria provided in NUREG-0800, Section 5.3.3, and the requirements of 10 CFR Part 50, Appendix B, and 10 CFR 50.61. The staff based its conclusion on the following:

- STD COL 5.3-4, relating to plant-specific beltline material properties, is acceptable because the applicant's proposed resolution meets the relevant acceptance criteria addressed in NUREG-0800, Section 5.3.3 and thus provides an acceptable basis to satisfy, in part, the requirements of 10 CFR Part 50, Appendix B, and 10 CFR 50.61.

5.3.5 Reactor Vessel Insulation

RV insulation is provided to minimize heat losses from the primary system. Non-safety-related reflective insulation similar to that in use in current PWRs is utilized.

WLS COL FSAR, Revision 11, Section 5.3 incorporates by reference, with no departures or supplements, AP1000 DCD, Revision 19, Section 5.3.5. The staff reviewed the COL application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The staff's review confirmed that there is no outstanding issue related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

5.4 Component and Subsystem Design (Related to RG 1.206, Section C.III.1, Chapter 5, C.I.5.4, "Reactor Coolant System Component and Subsystem Design")

5.4.1 Introduction

This section pertains to the design of various components and subsystems within, or associated with, the RCS. Principal components or subsystems include the following:

- Reactor coolant pumps
- Steam generators (SG), including materials and ISI
- RCS piping and valves
- Main steam line flow restriction
- Pressurizer and pressurizer relief discharge
- Automatic depressurization system valves
- Normal residual heat removal system
- RCS pressure relief devices
- Component supports
- RCS high point vents
- Core makeup tank
- Passive residual heat removal heat exchanger

The majority of the design-related information in the AP1000 DCD is incorporated by reference in the WLS COL application. Regarding the SGs, a program is developed by the COL applicant to ensure tube structural and leakage integrity will be maintained at a level comparable to that of the original design requirements. An effective program depends on both the program and the design features of the SGs.

The RNS is a non-safety-related system. Since the RNS is not required to operate to mitigate design-basis events, it is not credited in the Chapter 15, "Transient and Accident Analysis," safety analysis. However, the RNS is considered an important system because the RNS provides residual heat removal capability to several reactor systems. These major RNS non-safety-related functions include the RCS shutdown heat removal, RCS LTOP, RCS and refueling cavity purification during refueling operations, in-containment refueling water storage tank (IRWST) cooling, low pressure makeup to the RCS, and post-accident heat removal recovery. In addition, the RNS provides safety-related functions that include containment isolation of the RNS lines penetrating the containment, preservation of the RCS pressure boundary, and long term post-accident makeup to the containment inventory.

5.4.2 Summary of Application

WLS COL FSAR, Revision 11, Section 5.4 incorporates by reference AP1000 DCD, Revision 19, Section 5.4 with departures and/or supplements.

To address these departures and/or supplements in WLS COL FSAR Section 5.4.2.5, the applicant provided the following additional information:

Departures

- WLS DEP 3.2-1 and WLS 6.3-1

The applicant provided additional information in WLS COL FSAR Section 5.4 about WLS DEP 3.2-1 and WLS DEP 6.3-1 related to design modifications to the condensate return portion of the Passive Core Cooling System and quantifying the duration that the passive residual heat removal heat exchanger (PRHR-HX) can maintain safe shutdown conditions, respectively. This information, as well as related WLS DEP 3.2-1 and WLS DEP 6.3-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.1 of this SER.

AP1000 COL Information Item

- STD COL 5.4-1

The applicant provided additional information in STD COL 5.4-1 to address COL Information Item 5.4-1 as described in AP1000 DCD Section 5.4.15. The information in STD COL 5.4-1 provides the SG program description, references the applicable ASME B&PV Code, Section XI requirements and industry guidelines, and refers to the TS for the program requirements.

The detailed inspection and reporting requirements are provided in WLS COL FSAR, Part 4, “Technical Specifications,” Sections 1.1 (“Definitions”), 3.4.7 (“RCS Operational Leakage”), 3.4.18 (“Steam Generator (SG) Tube Integrity”), 5.5.4 (“Steam Generator (SG) Program”), 5.6.8 (“Steam Generator Tube Inspection Report”), and in the associated bases sections of the TS.

5.4.3 Regulatory Basis

The regulatory basis of the AP1000 DCD information incorporated by reference in the COL application is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the applicable requirements of NRC regulations for the component and subsystem design are given in NUREG-0800, Section 5.4.2.

The applicable regulatory requirements for acceptance of the COL information item are 10 CFR 50.55a, “Codes and Standards,” as it relates to periodic inspection and testing of the

RCPB as detailed in ASME Code, Section XI, and 10 CFR Part 50, Appendix A, GDC 32, "Inspection of Reactor Coolant Pressure Boundary," as it relates to the accessibility of SG tubes for periodic testing. In addition, 10 CFR 50.55a(b)(2)(iii) states that if the TS include SG surveillance requirements that are different than those in ASME Code, Section XI, Article IWB-2000, then the SG tube inspection requirements are governed by the TS.

The regulatory basis for evaluating the RNS is documented in NUREG-1793, Section 5.4.7 and its supplements. While the RNS is a non-safety-related system, it is considered to be important to safety because it provides the first line of defense during an accident to prevent unnecessary actuation of the passive core cooling systems. Regulatory oversight of the active non-safety systems in passive plant designs is subject to a staff evaluation of the regulatory treatment of non-safety systems (RTNSS). NUREG-1793, Chapter 22, "Regulatory Treatment of Non-safety Systems," provides a detailed evaluation of the RTNSS issue in accordance with NRC policy for passive reactor plant designs. Non-safety-related systems that provide defense-in-depth capabilities for the AP1000 design includes the RNS. For this defense-in-depth system to operate, the associated systems and structures to support these functions must also be operable, including non-safety-related component cooling water system and the service water system. The staff's evaluation of the changes that are proposed focused primarily on confirming that the changes will not adversely affect safety-related SSCs or those that satisfy the criteria for RTNSS. Therefore, the proposed changes were evaluated using the guidance provided by NUREG-0800, Section 5.4.7, as it pertains to these considerations. Acceptability was determined based on conformance to the existing AP1000 licensing basis, the guidance specified by NUREG-0800, Section 5.4.7 (10 CFR Part 50, Appendix A, GDC 34, "Residual heat removal," as it relates to requirements for a RNS system), and NRC policy with respect to RTNSS as discussed in SECY-94-084 and SECY-95-132.

5.4.4 Technical Evaluation

The staff reviewed WLS COL FSAR Section 5.4 and the referenced DCD to ensure that the combination of the DCD information incorporated by reference and the COL application information represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and the DCD information incorporated by reference addressed the required information relating to RCS component and subsystem design. The results of the staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff's review determined the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) contains evaluation material from the SER for the BLN Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 5.4.4:

> *AP1000 COL Information Item*
>
> - *STD COL 5.4-1*
>
> *In AP1000 DCD Section 5.4.15, Westinghouse identified COL Information Item 5.4-1 for the COL applicant to address the SG tube integrity with an SG Tube Surveillance Program and address the need to develop a program for periodic monitoring of degradation of steam generator internals. Similarly, in NUREG-1793, Section 5.4.2.2.2, the staff identified COL Action Item 5.4.2.2.3-1 and noted that an SG tube surveillance program is necessary to address the concerns raised in GL 97-06, "Degradation of Steam Generator Internals."*
>
> *In Revision 17 of the AP1000 DCD, Westinghouse proposed changes to the AP1000 generic TS related to adopting TS Task Force Traveler (TSTF) 449, Revision 4, "Steam Generator Tube Integrity." TSTF 449 is incorporated in the current Westinghouse Owners Group Standard Technical Specifications (STS), NUREG-1431, Revision 3.1, December 1, 2005. The TS and bases sections listed above for SG tube integrity in the BLN SER are identical to those in Revision 17 of the AP1000 DCD.*
>
> *With respect to the information provided in STD COL 5.4-1, the staff reviewed the description in Chapter 5 of the FSAR using the guidelines in RG 1.206, Section C.III.1, Chapter 5, C.I.5.4.2.2; Section 5.4.2.2 of NUREG-0800; and the TS proposed in the AP1000 DCD (which are based on NUREG-1431, Revision 3.1 and are the STS for Westinghouse operating plants). The staff confirmed tube inspection will meet the requirements of Section XI of the ASME Code, and that the applicant referenced an acceptable method (RG 1.121) for determining the tube repair criteria for maintaining structural integrity. The staff determined the TS proposed for BLN Nuclear Plant, Units 3 and 4 are consistent with the approved STS and the leakage limits and SG tube integrity requirements are appropriate as they apply to BLN, and are therefore acceptable. In addition, the applicant took exception to the guidance contained in Regulatory Guide 1.83, Revision 1 and stated that the applicant's program will be implemented according to Nuclear Energy Institute (NEI) 97-06 ("Steam Generator Program Guidelines") and EPRI SG guidelines, which are referenced in the STS and, thus, provide acceptable methods for implementing ASME Code requirements. With respect to tube integrity considerations, the Model Delta-125 SG planned for the BLN*

> *units closely resembles the Model Delta-75 installed as replacement SGs at some operating plants*
>
> *According to Section 5.4.2.2 of NUREG-0800, because the SG program is part of the ISI requirements, it is an operational program that should be fully described, with implementation milestones listed in the appropriate table in Chapter 13 of the FSAR. In response to RAI 5.4.2.2-1 from the staff, in a letter dated June 5, 2008, the applicant proposed revising FSAR Chapter 13, Table 13.4-201 to add Section 5.4.2.5 ("Steam Generator Inservice Inspection") as one of the FSAR sections addressed by the operational program titled "Inservice Inspection Program." Similarly, in response to RAI 5.4.2.2-2, the applicant proposed revising Table 13.4-201 to add Section 5.4.2.5 as one of the FSAR sections addressed by the operational program titled "Preservice Inspection Program." These proposed revisions are acceptable because they make the SG tube ISI part of the operational programs and ensure PSIs will be performed, consistent with the acceptance criteria in Section 5.4.2.2 of NUREG-0800 and RG 1.206. The staff verified that Revision 1 of Table 13.4-201 adequately incorporates the above. As a result, RAI 5.4.2.2-1 and RAI 5.4.2.2-2 are closed.*

In a February 5, 2009, letter, Duke Energy endorsed BLN RAIs 05.04.02.02-01 and 05.04.02.02-02. Therefore, the above discussion related to these RAIs is equally applicable to the WLS COL application. In addition, the staff also verified that the WLS COL FSAR, Section 5.3 and WLS COL Part 10 have been adequately updated to include the applicable changes resulting from these RAI responses.

5.4.5 Post Combined License Activities

The license condition language in this section has been clarified from previously considered language. In a letter dated March 22, 2016 (ADAMS Accession No. ML16084A099), the applicant did not identify any concerns with the clarified license condition language. The changes do not affect the staff's above analysis of the conditions, and therefore, for the reasons discussed in the technical evaluation section above, the staff finds the following license condition acceptable:

- License Condition (5-6) – No later than 12 months after the issuance of the COL, the licensee shall submit to the Director of NRO a schedule that supports planning for and conduct of NRC inspections of the PSI/ISI program. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until either the PSI/ISI program has been fully implemented.

5.4.6 Conclusion

The staff reviewed the WLS COL application and the applicable sections of the referenced DCD. The staff's review confirmed that the applicant adequately addressed the required information relating to RCS component and subsystem design, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the relevant regulatory requirements provided in 10 CFR Part 50, Appendix A, GDC 32 and 10 CFR 50.55a, and the regulatory guidance addressed in RG 1.206 and RG 1.121. The staff based its conclusion on the following:

- WLS DEP 3.2-1 and WLS DEP 6.3-1, related to design modifications to the condensate return portion of the Passive Core Cooling System and quantifying the duration that the passive residual heat removal heat exchanger can maintain safe shutdown conditions, respectively, are reviewed and found acceptable by the staff in Section 21.1 of this SER.

- STD COL 5.4-1 relating to the SG Program, is acceptable because it meets the relevant guidelines of RG 1.206, Section C.III.1, Chapter 5, C.I.5.4.2.2 and RG 1.121. Conformance with these guidelines provides an acceptable basis for satisfying, in part, the requirements of 10 CFR Part 50, Appendix A, GDC 32, and 10 CFR 50.55a including the specific modification provided in 10 CFR 50.55a(b)(2)(iii).

6 ENGINEERED SAFETY FEATURES

6.0 Engineered Safety Features

Engineered safety features (ESF) are design features designed to preclude an accidental release of reactor fission products or to protect the public in the event of an accidental release from the reactor coolant system (RCS). The function of the ESF is to localize, control, mitigate, and terminate such accidents, and to maintain radiation exposure levels to the public below applicable limits and guidelines. William States Lee III Nuclear Station, Units 1 and 2, (WLS), combined license (COL) Final Safety Analysis Report (FSAR), Revision 11, Section 6.0, incorporates by reference with no departures or supplements, AP1000 Design Control Document (DCD), Revision 19, Section 6.0, “Engineered Safety Features.” . The U.S. Nuclear Regulatory Commission (NRC) staff (the staff) reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The staff confirmed that there are no outstanding issues related to this section. The results of the staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design,” and its supplements.

6.1 Engineered Safety Features Materials

This section of the report provides the evaluation of the materials used in the fabrication of ESF components and of the provisions to avoid material interactions that could impair the operation of the ESF. The design information in WLS COL FSAR Section 6.1, is divided into two sections, Section 6.1.1, “Metallic Materials”; and Section 6.1.2, “Organic Materials.” The staff’s evaluation of these sections is provided below.

6.1.1 Metallic Materials

6.1.1.1 *Introduction*

In this section of the report, the staff reviewed metallic materials used in ESF components to ensure that they are compatible with one another and with ESF fluids. The compatibility of ESF component materials with fluids in ESF systems should ensure that there is a low probability of causing abnormal leakage, of rapidly propagating failure, and of gross rupture of reactor coolant pressure boundary components. Metallic materials and fluids should also be compatible with the auxiliary systems that directly support ESF systems.

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[1] See Section 1.2.2 of this report for a discussion of the staff’s review related to verification of the scope of information to be included in a COL application that references a design certification (DC).

6.1.1.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 6.1, incorporates by reference AP1000 DCD, Revision 19, Section 6.1. In addition, in WLS COL FSAR Section 6.1.1, the applicant provided the following:

<u>*AP1000 COL Information Item*</u>

- STD COL 6.1-1

The applicant provided information in Standard (STD) COL 6.1-1 to resolve AP1000 COL Information Item 6.1-1. STD COL 6.1-1 describes quality assurance measures for special processes in fabricating austenitic stainless steels. STD COL 6.1-1 is discussed in WLS COL FSAR Sections 6.1.1.2, Fabrication Requirements," and 6.1.3.1, "Procedure Review."

6.1.1.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. The acceptance criteria associated with the relevant requirements of NRC regulations for the metallic materials are given in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 6.1.1. This reference will be referred to in this report as the SRP.

The regulatory basis for the AP1000 COL information item is Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," as it relates to the quality assurance requirements for the design, fabrication, and construction of safety-related structures, systems, and components (SSCs). Guidance used in the review of this COL information item is described in Regulatory Guide (RG) 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," Revision 3, and RG 1.44, "Control of the Use of Sensitized Stainless Steel."

6.1.1.4 *Technical Evaluation*

The staff reviewed WLS COL FSAR Section 6.1.1, and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to metallic materials. The results of the staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the DC and use this review to evaluate subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the safety evaluation report (SER) for the Vogtle (VEGP) reference COL (RCOL) application were equally applicable to the WLS COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from responses to requests for additional information (RAIs).

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed and that those changes were actualized in the WLS COL FSAR.

- The staff verified that the site-specific differences, if any, did not adversely affect any previous relevant evaluation or conclusion.

The staff completed its review and concluded that the evaluation performed for the standard content is directly applicable to the WLS COL application. This standard content material is identified in this report by use of italicized, double-indented formatting. Section 1.2.3 of this report provides an explanation of why the standard content material from the SER for the VEGP reference COL application includes evaluation material from the SER for the Bellefonte Nuclear Plant, Units 3 and 4 (BLN), COL application.

The following portion of this technical evaluation section is reproduced from the VEGP SER Section 6.1.1.4:

> <u>*COL Information Item*</u>
>
> - *STD COL 6.1-1*
>
> *The NRC staff reviewed STD COL 6.1-1 related to COL Information Item 6.1-1 included under Section 6.1.1.2 of the BLN COL FSAR, which addresses the COL information item identified in AP1000 DCD Section 6.1.3.1 related to the fabrication requirements for austenitic stainless steel.*
>
> *The COL information item identified in AP1000 DCD Section 6.1.3.1 states:*
>
>> *The Combined License applicants referencing the AP1000 will address review of vendor fabrication and welding procedures or other quality assurance methods to judge conformance of austenitic stainless steels with Regulatory Guides 1.31 and 1.44.*
>
> *This commitment was also documented as COL Action Item 6.1.1-1 in the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:*
>
>> *The COL applicant will review vendor fabrication and welding procedures or other quality assurance methods to ensure that austenitic stainless steels meet the guidelines of RGs 1.31 and 1.44.*
>
> *The COL information in the FSAR that is to be added to AP1000 DCD Section 6.1.1.2 states:*

> *In accordance with Appendix B to 10 CFR Part 50, the quality assurance program establishes measures to provide control of special processes. One element of control is the review and acceptance of vendor procedures that pertain to the fabrication, welding, and other quality assurance methods for safety related component [sic] to determine both code and regulatory conformance. Included in this review and acceptance process are those vendor procedures necessary to provide conformance with the requirements of Regulatory Guides 1.31 and 1.44 for engineered safety features components as discussed in DCD Section 6.1 and reactor coolant system components as discussed in DCD Section 5.2.3.*

The staff finds the COL information provided by the applicant meets the quality assurance guidelines for austenitic stainless steels specified in RG 1.31 (weld metal ferrite content) and RG 1.44 (the use of sensitized stainless steel). The staff's conclusion is based on the applicant's statement affirming that its Appendix B quality assurance program will address the concerns of these RGs. It is also based on Appendix 1A of the AP1000 DCD, as modified by a letter dated April 7, 2010, from the AP1000 applicant. The modified DCD appendix will be incorporated by reference in a future version of the BLN COL FSAR and will indicate full conformance with these RGs. In addition, the discussions in AP1000 DCD Sections 6.1.1.2 and 5.2.3.4 provide details about how conformance will be accomplished.

6.1.1.5 *Post Combined License Activities*

There are no post COL activities related to this section.

6.1.1.6 *Conclusion*

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to metallic materials used in ESF and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff concludes that the relevant information presented in the WLS COL FSAR is acceptable because it meets the requirements of 10 CFR Part 50, Appendix B, and the guidance provided in RG 1.31 and RG 1.44. STD COL 6.1-1 is acceptable because the Appendix B quality assurance program proposed by the applicant provides adequate controls over vendor fabrication and welding procedures to ensure that austenitic stainless steels meet the guidelines of RG 1.31 and RG 1.44.

6.1.2 Organic Materials

6.1.2.1 *Introduction*

Protective coatings are applied for corrosion prevention to the interior and exterior surfaces of the containment vessel, radiologically controlled areas outside containment, and the remainder of the plant. The coatings selection process accounts for these differing coating needs for these four areas. The AP1000 design considers the function of the coatings, their potential failure modes, and their requirements for maintenance. Other organic materials that may be present in the containment are associated with the specific type of equipment and the supplier selected to provide it. Materials are evaluated for potential interaction with the ESF to provide confidence that the performance of the ESF is not unacceptably affected.

6.1.2.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 6.1, incorporates by reference the AP1000 DCD, Revision 19, Section 6.1. AP1000 DCD Section 6.1 includes Section 6.1.2. In addition, in WLS COL FSAR, Section 6.1.2, the applicant provided the following:

AP1000 COL Information Item

- STD COL 6.1-2

The applicant provided information in STD COL 6.1-2 to resolve AP1000 COL Information Item 6.1-2. STD COL 6.1-2 discusses a program to control procurement, application, inspection, and monitoring of Service Level I and Service Level III coatings. This COL information item is discussed in WLS COL FSAR Sections 6.1.2.1.6, “Quality Assurance Features,” and 6.1.3.2, “Coating Program.” In a March 31, 2010, letter, the AP1000 DCD applicant (Westinghouse) proposed revisions to AP1000 COL Information Item 6.1-2 in AP1000 DCD Section 6.1.3.2, to address Service Level II coatings. In a March 24, 2011, letter the WLS COL applicant endorsed the VEGP August 13, 2010 letter, respectively, that proposed revising the FSAR to address the updated AP1000 COL Information Item 6.1-2.

6.1.2.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. The acceptance criteria associated with the relevant requirements of NRC regulations for protective coatings are given in NUREG-0800, Section 6.1.2. The applicable regulatory basis for the resolution of the COL information item is 10 CFR Part 50, Appendix B, as it relates to the quality assurance requirements for the design, fabrication, and construction of safety-related SSCs. Guidance for the resolution of the COL information item is described in RG 1.54, “Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants,” Revision 1.

6.1.2.4 *Technical Evaluation*

The staff reviewed WLS COL FSAR Section 6.1.2, and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The staff’s review confirmed that the information in the application and incorporated by reference addresses the required information relating to protective coatings and other organic materials inside containment. The results of the staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff completed its review and concluded that the evaluation performed for the standard content is directly applicable to the WLS COL application. This standard content material is identified in this report by use of italicized, double-indented formatting. Section 1.2.3 of this report provides an explanation of why the standard content material from the SER for the VEGP reference COL application includes evaluation material from the SER for the BLN COL application. Although the staff concluded that the evaluation performed for the standard content is directly applicable to the WLS COL application, there is a difference in how the VEGP applicant addressed STD COL 6.1-2 and how the BLN applicant addressed this review item. This difference, which is based on a change proposed in the AP1000 DCD, is evaluated by the staff below, following the standard content material for STD COL 6.1-2. The two confirmatory items in the standard content material retain the number assigned in the VEGP SER, and are also addressed in the standard content material.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 6.1.2.4:

> *<u>AP1000 COL Information Item</u>*
>
> - *STD COL 6.1-2*
>
> *The NRC staff reviewed STD COL 6.1-2 included under Section 6.1.2.1.6 of the BLN COL FSAR related to COL Information Item 6.1-2. COL Information Item 6.1-2 states:*
>
> > *The Combined License applicants referencing the AP1000 will provide a program to control procurement, application, and monitoring of Service Level I and Service Level III coatings. The program for the control of the use of these coatings will be consistent with [DCD] subsection 6.1.2.1.6.*
>
> *This commitment was also captured as COL Action Item 6.1.2-1 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:*
>
> > *The COL applicant will prepare a program to control procurement, application, and monitoring of Service Level I and Service Level III coatings.*
>
> *The added information in the BLN COL FSAR replaces the third paragraph under the section titled, "Service Level I and Service Level III Coatings," in AP1000 DCD Section 6.1.2.1.6 with the following:*
>
> > *During the design and construction phase the coatings program associated with selection, procurement and application of safety related coatings is performed to applicable quality standards. Regulatory Guide 1.54 and [American Society for Testing and*

> *Materials] ASTM D5144 form the basis for the coating program. During the operations phase, the coatings program is administratively controlled in accordance with the quality assurance program implemented to satisfy 10 CFR Part 50, Appendix B, and 10 CFR Part 52 requirements. The coatings program provides direction for the procurement, application, and monitoring of safety related coating systems. Coating system monitoring requirements for the containment coating systems are based on ASTM D5163, "Establishing Procedures to Monitor the Performance of Coating Service Level I Coating Systems in an Operating Nuclear Power Plant," and ASTM D7167, "Establishing Procedures to Monitor the Performance of Safety-Related Coating Service Level III Lining Systems in an Operating Nuclear Power Plant." Any anomalies identified during coating monitoring are resolved in accordance with applicable quality assurance requirements.*

The AP1000 DCD, which the applicant incorporates by reference, includes the following description of the quality assurance program:

> *The quality assurance program for Service Level I and Service Level III coatings conforms to the requirements of [American Society of Mechanical Engineers] ASME NQA-1-1983 as endorsed in Regulatory Guide 1.28 ["Quality Assurance Program Criteria (Design and Construction)"]. Safety related coatings meet the pertinent provisions of 10 CFR Part 50 Appendix B to 10 CFR Part 50. The service level classification of coatings is consistent with the positions given in Revision 1 of Regulatory Guide 1.54, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants." Service Level I and Service Level III coatings used in the AP1000 are tested for radiation tolerance and for performance under design basis accident conditions. Where decontaminability is desired, the coatings are evaluated for decontaminability. The coating applicator submits and follows acceptable procedures to control surface preparation, application of coatings and inspection of coatings. The painters are qualified and certified, and the inspectors are qualified and certified.*
>
> *The inorganic zinc coating used on the inside surface (Service Level I coatings) and outside surface (Service Level III coatings) of the containment shell is inspected using a non-destructive dry film thickness test and a MEK rub test. These inspections are performed after the initial application and after recoating. Long term surveillance of the coating is provided by visual inspections performed during refueling outages. Other inspections are not required.*

Section 6.1.2 of NUREG-0800 references RG 1.54 as providing an acceptable method of complying with the quality assurance requirements in regard to protective coatings applied to ferritic steels, aluminum, stainless steel, *zinc-coated (galvanized) steel, concrete, or masonry surfaces of nuclear facilities. RG 1.54 lists a number of ASTM standards that provide guidance on practices and programs that are acceptable to the NRC staff for the selection, application, qualification, inspection, and maintenance of protective coatings applied in nuclear power plants. Section 6.1.2 of NUREG-0800 also states that a coating system to be applied inside the containment vessel is acceptable if it meets the regulatory positions of RG 1.54 and the standards of ASTM D5144-00 and ASTM D3911-03. By contrast, the AP1000 DCD references RG 1.54, but only with respect to classification of coating service level as I, II, or III.*

The AP1000 DCD text to be replaced with the COL information item stated that the procurement, application, and monitoring of Service Level I and Service Level III coatings are controlled by a program prepared by the COL applicant The information provided clarified that the applicant's coatings program, with respect to procurement, application, inspection, and monitoring, will be consistent with the recommendations of RG 1.54, which is endorsed in Section 6.1.2 of NUREG-0800 as an acceptable method of meeting the quality assurance requirements of 10 CFR Part 50, Appendix B for safety-related and nonsafety-related coatings. However, the information provided by the applicant to resolve the COL information item merely states that the protective coatings program complies with RG 1.54, when, in fact, the program was not yet developed. Therefore, the COL applicant had not provided a coatings program as committed in COL Information Item 6.1-2.

To resolve this issue, in request for additional information (RAI) 6.1.2-1, the staff requested the following information:

1. *The applicant should describe the standards to be applied to maintenance of the protective coatings in the program description. The description of the proposed coatings program should also describe the standards to be applied to selection and qualification of coatings, if the applicant intends to use coatings systems different than those described in the AP1000 DCD, either during construction or after plant operation commences.*

2. *The program description should describe the administrative controls that will be applied to the coatings program.*

3. *Provide the schedule for full implementation of the coatings program with respect to major milestones in the construction of the plant; for example, prior to application of coatings, prior to preparation of surfaces to be coated, or prior to procurement of coatings materials.*

In a letter dated May 23, 2008, the applicant provided the following response:

Item 1) The coating program will be based on Revision 1 of RG 1.54 and the referenced ASTM standards in ASTM D5144. Also, the guidance provided in ASTM D5163, "Establishing Procedures to Monitor the Performance of Coating Service Level I Coating Systems in an Operating

Nuclear Power Plant," and in ASTM D7167, "Establishing Procedures to Monitor the Performance of Coating Service Level III Coating Systems in an Operating Nuclear Power Plant," will be used to specify monitoring (maintenance) requirements for the safety-related coating systems pertaining to containment. While a change in coating systems (from those described in the AP1000 DCD) is not anticipated, if a different safety-related coating system is needed, it will be evaluated in accordance with the appropriate change process, i.e., 10 CFR 50.59 or 10 CFR Part 52, Appendix D, Section VIII.

Item 2) FSAR Section 6.1.3.2, Coating Program, will be revised to indicate compliance with 10 CFR Part 50, Appendix B, and 10 CFR Part 52 requirements implemented by the quality assurance program for the plant (see FSAR Chapter 17 and Part 11 of the COL application) for design, construction, and operation of the units.

Item 3) During the design and construction phase, the requirements for the coating program will be contained in certified drawings and/or standards and specifications controlling the coating processes of the designer (Westinghouse); these design documents will be available prior to the procurement and application of the coating material by the constructor of the plant. Prior to initial fuel loading, a consolidated plant coating program will be in place to address procurement, application, and monitoring (maintenance) of those coating system(s) for the life of the plant.

The staff finds the applicant's response to Item 1 acceptable because, pursuant to RG 1.54, ASTM D5163 provides guidelines that are acceptable to the NRC staff for establishing an in-service coatings monitoring program for Service Level I coating systems in operating nuclear power plants and for Service Level II and other areas outside containment (as applicable). The applicant also specified ASTM D7167 for monitoring (maintenance) requirements for the safety-related coating systems pertaining to containment. Although ASTM D7167 is not listed in RG 1.54 or ASTM D5144, the staff finds it an appropriate standard because it addresses maintenance of Service Level III coatings. Additionally, ASTM D7167 references ASTM D4541 and ASTM D3359, which are listed in RG 1.54 as acceptable standards for maintenance of protective coatings in nuclear power plants. Further, if a change in any of the originally specified coatings systems is necessary, the applicant will use an appropriate process, either the 10 CFR 50.59 or 10 CFR Part 52, Appendix D, Section VIII process, to evaluate the change. The staff finds the application of these regulations an appropriate alternative to control of the selection of coatings by the consolidated coatings program.

The BLN application references later versions of ASTM D5144 and ASTM D5163 than those referenced in RG 1.54, Revision 1. The use of the 2008 revision of ASTM D5144 is acceptable because it provides detailed requirements through reference to other coatings standards applicable to BLN. In this regard, it is not changed with respect to the 2000 revision referenced in the RG 1.54, Revision 1.

Similarly, the 2005 revision of ASTM D5163 is referenced in the BLN COL application rather than the 1996 revision referenced in RG 1.54, Revision 1. The staff finds this acceptable because the NRC staff has accepted the 2005 revision of ASTM D5163 as the basis for the Aging Management Program XI.S8 in NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," Volume 2, Revision 2 (license renewal). With respect to simulated design-basis accident qualification testing for coatings, the staff notes that the applicable version of ASTM D3911 is the 1995 revision, as indicated in Appendix 1A of the AP1000 DCD.

In response to Item 2, the applicant stated that the administrative controls spelled out in its Quality Assurance Program Document (QAPD) will be applied to the coatings program. The staff finds that this will ensure compliance with the requirements of 10 CFR Part 50, Appendix B, which is a regulatory acceptance criterion of Section 6.1.2 of NUREG-0800. However, the staff notes that the QAPD references ASME NQA-1-1994 as an acceptable means to implement the requirements of 10 CFR Part 50, Appendix B, rather than ASME NQA-1-1983 as referenced by AP1000 DCD Section 6.1.2.1.6. ASME NQA-1-1994 is used as the basis for NUREG-0800 Section 17.5, "Quality Assurance Program Description - Design Certification, Early Site Permit and New License Applicants," which is applicable to the quality assurance program for a COL. Therefore, the staff finds the use of ASME NQA-1-1994 acceptable with respect to quality assurance requirements for coatings.

The staff finds the response to Item 3 acceptable because the applicant indicated the consolidated plant coating program will be in place to address procurement, application, and monitoring (maintenance) of those coating system(s) for the life of the plant, prior to initial fuel loading. During the construction phase, the requirements for the coating program will be contained in certified drawings and/or standards and specifications controlling the coating processes, which meets the requirements of 10 CFR Part 50, Appendix B, Criterion III with respect to design control and instructions, Criterion IV with respect to procurement document control, and Criterion V with respect to procedures and drawings.

The applicant also provided proposed changes to BLN COL FSAR Section 6.1.2.1.6 to incorporate the information included in the response to RAI 6.1.2-1. The staff confirmed that FSAR Section 6.1.2.1.6 has been revised to include information on the quality assurance program. However, since the information proposed to be added does not include the detailed information on control of coatings during the design and construction phase, the staff identified Open Item 6.1.2-1 to ensure that BLN COL FSAR Section 6.1.2.1.6 is revised to include the information from the response to RAI 6.1.2-1, Item 3, related to control of the coating program during the design and construction phase and the schedule for full implementation of the consolidated coatings program.

<u>*Resolution of Standard Content Open Item 6.1.2-1*</u>

Standard Content Open Item 6.1.2-1 was identified by the staff because the information the BLN applicant provided about the control of coatings during the *design and construction phase, although acceptable, was not included in the BLN COL FSAR. In the July 2, 2010, letter, the VEGP applicant proposed inserting the three paragraphs below in Section 6.1.2.1.6 of the VEGP FSAR. These paragraphs would replace the third paragraph under "Service Level I and Service Level III Coatings" in DCD Section 6.1.2.1.6.*

> *During the design and construction phase, the coatings program associated with selection, procurement and application of safety related coatings is performed to applicable quality standards. The requirements for the coatings program are contained in certified drawings and/or standards and specifications controlling the coating processes of the designer (Westinghouse) (these design documents will be available prior to the procurement and application of the coating material by the constructor of the plant). Regulatory Guide 1.54 and ASTM D5144 ([FSAR] Reference 201) form the basis for the coatings program.*
>
> *During the operations phase, the coatings program is administratively controlled in accordance with the quality assurance program implemented to satisfy 10 CFR Part 50, Appendix B, and 10 CFR Part 52 requirements. The coatings program provides direction for the procurement, application, inspection, and monitoring of safety related coating systems. Prior to initial fuel loading, a consolidated plant coatings program will be in place to address procurement, application, and monitoring (maintenance) of those coating system(s) for the life of the plant.*
>
> *Coating system monitoring requirements for the containment coating systems are based on ASTM D5163 ([FSAR] Reference 202), "Standard Guide for Establishing Procedures to Monitor the Performance of Coating Service Level I Coating Systems in an Operating Nuclear Power Plant," and ASTM D7167 ([FSAR] Reference 203), "Standard Guide for Establishing Procedures to Monitor the Performance of Safety-Related Coating Service Level III Lining Systems in an Operating Nuclear Power Plant." Any anomalies identified during coating inspection or monitoring are resolved in accordance with applicable quality assurance requirements.*

As discussed above in the portion of the staff's evaluation reproduced from Section 6.1.2.4 of the BLN SER, the staff found the COL information related to control of coatings during the design and construction phase acceptable. The staff finds that the FSAR revisions proposed above are consistent with the information reviewed for the BLN SER and applicable to VEGP. Therefore, the staff finds the FSAR revisions proposed in the July 2, 2010, letter, acceptable for closing Open Item 6.1.2-1. The incorporation of these proposed revisions is being tracked as Confirmatory Item 6.1-1.

<u>Resolution of Standard Content Confirmatory Item 6.1-1</u>

Confirmatory Item 6.1-1 is an applicant commitment to revise its FSAR Section 6.1.2.1.6 to provide information regarding Service Level I and Service Level III coatings. The Staff verified that the VEGP COL FSAR was appropriately revised. As a result, Confirmatory Item 6.1-1 is now closed.

<u>Evaluation of Additional Design Information</u>

As discussed above, AP1000 DCD Section 6.1.3.2 requires the COL applicants to provide a program for procurement, application, and monitoring of Service Level I and Service Level III coatings consistent with DCD Section 6.1.2.1.6. However, DCD Section 6.1.2.1.6 also states that COL applicants will also address the program for Service Level II coatings, and that coatings programs for Service Level I, II, and III will include inspection. Therefore, in a letter dated March 31, 2010, the AP1000 DCD applicant proposed the following revision to DCD Section 6.1.3.2:

> *The Combined License applicants referencing the AP1000 will provide programs to control procurement, application, inspection, and monitoring of Service Level I, Service Level II, and Service Level III coatings. The programs for the control of the use of these coatings will be consistent with subsection 6.1.2.1.6.*

In letters dated July 2 and August 13, 2010, the VEGP applicant addressed the addition of Service Level II to the COL information item by proposing the following additions to Section 6.1.2.1.6 of the VEGP COL FSAR. The first is a new second paragraph under "Service Level II Coatings" in DCD Section 6.1.2.1.6.

> *Such safety-related Service Level II coatings used inside containment are procured to the same standards as Service Level I coatings with regard to radiation tolerance and performance under design basis accident conditions as discussed below.*

The second addition replaces the second sentence of the third paragraph under "Service Level II Coatings" in DCD Section 6.1.2.1.6.

> *Coating system application, inspection, and monitoring requirements for the Service Level II coatings used inside containment will be performed in accordance with a program based on ASTM D5144 ([FSAR] Reference 201), "Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants," and the guidance of ASTM D5163 ([FSAR] Reference 202), "Standard Guide for Establishing Procedures to Monitor the Performance of Coating Service Level I Coating Systems in an Operating Nuclear Power Plant." Any anomalies identified during coating inspection or monitoring are resolved in accordance with applicable quality requirements.*

The NRC staff finds it acceptable to procure Service Level II coatings in containment to the same standards as Service Level I coatings because the staff, through RG 1.54, has endorsed the use of these standards to procure safety-related coatings inside containment. The staff also finds it acceptable to use ASTM D5144 and D5163 as a basis for application, inspection, and monitoring requirements for Service Level II coatings. As discussed in RG 1.54, ASTM D5144 is a top-level standard that provides general guidance on coating programs and detailed guidance by reference to other ASTM standards. Since it contains a single set of application requirements for all coatings, the staff finds it an acceptable basis for Service Level II coatings application and inspection. The staff finds ASTM D5163 acceptable for monitoring Service Level II coatings in containment because the use of ASTM D5163 conforms to the guidance in RG 1.54 for monitoring the performance of safety-related (Service Level I) coatings in containment, and there is no separate standard for Service Level II coatings. The incorporation of the proposed revisions to address Service Level II coatings into a future revision of the VEGP COL FSAR is being tracked as Confirmatory Item 6.1-2

Resolution of Standard Content Confirmatory Item 6.1-2

Confirmatory Item 6.1-2 is an applicant commitment to revise its FSAR Section 6.1.2.1.6 to provide information regarding the procurement of Service Level II coatings. The staff verified that the VEGP COL FSAR was appropriately revised. As a result, Confirmatory Item 6.1-2 is now closed.

6.1.2.5 *Post Combined License Activities*

There are no post COL activities related to this section.

6.1.2.6 *Conclusion*

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to protective coatings and other organic materials inside containment, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements. The staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the requirements of 10 CFR Part 50, Appendix B, and the guidance provided in RG 1.54, Revision 1. STD COL 6.1-2 is acceptable because the coating program meets a 10 CFR Part 50, Appendix B, quality assurance program, with the additional guidance provided in RG 1.54, Revision 1, and provides adequate controls over the programs to control procurement, application, inspection, and monitoring of Service Level I, Service Level II, and Service Level III coatings.

6.2 Containment Systems

6.2.1 Introduction

The containment systems (CSs), which include the primary containment, passive cooling system (heat removal system), isolation system, hydrogen control system, and leak rate test system, are discussed in this section. The containment encloses the reactor system and is the final barrier against the release of significant amounts of radioactive fission products in the event of an accident. The containment structure must be capable of withstanding, without loss of function, the pressure and temperature conditions resulting from postulated loss-of-coolant, steam line break, or feed water line break accidents. The containment structure must also maintain functional integrity in the long term following a postulated accident (i.e., it must remain a low leakage barrier against the release of fission products for as long as postulated accident conditions require).

6.2.2 Summary of Application

WLS COL FSAR, Revision 11, Section 6.1.2.4, incorporates by reference AP1000 DCD, Revision 19, Section 6.2. AP1000 DCD Section 6.2 includes Sections 6.2.1, “Containment Functional Design”; 6.2.2, “Passive Containment Cooling System”; 6.2.3, “Containment Isolation System”; 6.2.4, “Containment Hydrogen Control System”; and 6.2.5, “Containment Leak Rate Test System.” AP1000 DCD Section 6.2.5, is evaluated by the staff in NUREG-1793, Section 6.2.6. NUREG-1793 also includes the staff’s evaluation of fracture prevention of the containment pressure boundary in accordance with NUREG-0800, Section 6.2.7, and in-containment refueling water storage tank (IRWST) hydrodynamic loads. The staff’s evaluation of fracture prevention of the containment pressure boundary is found in Section 3.8 of this report. With respect to the hydrodynamic loads, the staff’s evaluation is discussed in NUREG-1793, Section 6.2.8. The staff’s evaluation of the containment cleanliness program associated with Generic Safety Issue (GSI)-191, “Assessment of Debris Accumulation on PWR [Pressurized-Water Reactor] Sump Performance,” is evaluated in Section 6.3 of this report.

WLS COL FSAR Table 1.9-203, “Listing of Unresolved Safety Issues and Generic Safety Issues,” includes a line item for Task Action Plan Item A-23, “Containment Leak Testing.” This item is addressed in WLS COL FSAR Section 6.2.5.1. In addition, in WLS COL FSAR Section 6.2, and in WLS COLA Part 10, the applicant provided information to address the following:

Departures

- WLS DEP 6.2-1

The applicant provided additional information in Section 6.2.4.5.1 of the WLS COL FSAR about WLS DEP 6.2-1 related to changes to the acceptance criteria applied to a specific ITAAC design commitment and associated inspection, test, or analysis in Tier 1 Table 2.3.9-3, Item 3 (for control of containment hydrogen concentration for beyond-design-basis accidents) to establish consistency with the current detailed design of the plant. This information, as well as related

WLS DEP 6.2-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.4 of this SER.

AP1000 COL Information Item

- STD COL 6.2-1

The applicant provided information in STD COL 6.2-1 to address AP1000 COL Information Item 6.2-1, which addresses the containment leak rate test program.

In addition to addressing the AP1000 COL information item, STD COL 6.2-1 also addresses, WLS COL FSAR Table 1.9-203, “Listing of Unresolved Safety Issues and Generic Safety Issues,” Task Action Plan Item A-23, “Containment Leak Testing.” This item is discussed in WLS COL FSAR Sections 6.2.5.1, ”Design Basis”; 6.2.5.2.2, “System Operation”; and 6.2.6, “Combined License Information for Containment Leak Rate Testing.”

License Conditions

- WLS COLA Part 10, “Proposed License Conditions (Including ITAAC),” License Condition 3, “Operational Program Implementation,” Section G “Fuel Loading,” Item G.8 –“Containment Leakage Rate Testing”

This proposed license condition states that the COL holder shall implement the containment leakage rate testing program prior to initial fuel load, as stated in WLS COL FSAR Table 13.4-201, “Operational Programs Required by NRC Regulations.”

- WLS COLA Part 10, “Proposed License Conditions (Including ITAAC),” License Condition 6, “Operational Program Readiness.”

This proposed license condition states that the COL holder shall provide an operational program implementation schedule to support NRC inspections. Specifically, the operational program relevant to this evaluation in Section 6.2 is WLS COL FSAR, Table 13.4-201, Item #7, “Containment Leakage Rate Testing Program.”

6.2.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. The acceptance criteria associated with the relevant requirements of NRC regulations for containment functional design are given in NUREG-0800, Section 6.2.1.1A. The regulatory requirements related to this section are 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants”; General Design Criterion (GDC) 16, “Containment Design”; GDC 38, “Containment Heat Removal”; and GDC 50, “Containment Design Basis.” The acceptance criteria associated with the relevant requirements of NRC regulations for containment leak rate testing are given in NUREG-0800, Section 6.2.6. The regulatory requirements related to this section are GDC 52, “Capability for Containment Leakage Rate Testing”; GDC 53, “Provisions for Containment Testing and Inspection”; GDC 54, “Piping System Penetrating Containment”; and 10 CFR Part 50, Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.” Compliance with the requirements of 10 CFR Part 50, Appendix J, Option A, or the requirements of 10 CFR Part 50,

Appendix J, Option B, and the provisions of RG 1.163, "Performance-Based Containment Leak-Test Program," constitutes an acceptable basis for satisfying the requirements of the GDC applicable to containment leakage rate testing. In addition, the staff used guidance in Nuclear Energy Institute (NEI) 94-01, as endorsed and modified by RG 1.163, "Performance-Based Containment Leak-Test Program." The staff used the guidelines of NuStart Technical Report, AP-TR-NS01-A, Revision 2, "Containment Leak Rate Test Program," to review the containment leakage rate testing operational program.

6.2.4 Technical Evaluation

The staff reviewed WLS COL FSAR Section 6.2, and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the containment systems. The results of the staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the DC and use this review to evaluate subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the VEGP reference COL application were equally applicable to the WLS COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from responses to RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed and that those changes were actualized in the WLS COL FSAR.

- The staff verified that the site-specific differences, if any, did not adversely affect any previous relevant evaluation or conclusion.

The staff completed its review and concluded that the evaluation performed for the standard content is directly applicable to the WLS COL application. This standard content material is identified in this report by use of italicized, double-indented formatting. Section 1.2.3 of this report provides an explanation of why the standard content material from the SER for the VEGP reference COL application includes evaluation material from the SER for the BLN COL application.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 6.2.4:

<u>AP1000 COL Information Item</u>

- *STD COL 6.2-1*

The NRC staff reviewed STD COL 6.2-1 related to COL Information Item 6.2-1 included under Section 6.2.5 of the BLN COL FSAR regarding the text added to Section 6.2.6 of the COL application. The added text references the program, which was reviewed and approved by the NRC in a letter from Stephanie Coffin, NRC, to Marilyn Kray, NuStart, "Final Safety Evaluation for AP1000 Technical Report No. AP-TR-NS01, Containment Leak Rate Test Program (TAC No. MD5136)," dated October 25, 2007.

<u>License Conditions</u>

- *Part 10, License Condition 3, Item G.<u>8</u>*

- *Part 10, License Condition 6*

The portion of License Conditions 3 and 6 relevant to this SER section is the containment leakage rate testing program listed in BLN COL FSAR Table 13.4-201. As noted in Section 13.4 of this SER, the containment leakage rate testing program meets the criteria for an operational program as specified in SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria." Therefore, the NRC staff finds License Conditions 3 and 6 acceptable, with respect to the inclusion of the containment leakage rate testing program in Table 13.4-201.

Due to discrepancies in the implementation milestones provided in various locations in the BLN COL application, RAI 6.2.6-1 was forwarded to the applicant. The applicant's response was that the milestones were meant to reflect the implementation of an approved testing program and when the tests were actually to be performed. However, the applicant agreed that this was not consistently reflected. The discrepancies have been addressed in BLN COL FSAR, Table 13.4-201, Sheet 2 of 7, and Part 10, License Conditions and ITAAC. The changes indicate that the containment leak rate testing program will be implemented prior to initial fuel load. This RAI is closed.

6.2.5 Post Combined License Activities

For the reasons discussed in the technical evaluation section above, the staff finds the following license conditions related to the containment leakage rate testing program acceptable:

- License Condition (6-1) – The licensee shall implement the containment leakage rate testing program before initial fuel load.

- License Condition (6-2) – No later than 12 months after issuance of the COL, the licensee shall submit to the appropriate Director of the Office of New Reactors (NRO) a schedule that supports planning for and conduct of NRC inspections of the containment leakage rate testing program. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until the containment leakage rate testing program has been fully implemented.

6.2.6 Conclusion

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to the containment systems, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements. In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and conforms to the guidance in NUREG-0800, Sections 6.2.1 and 6.2.6. The staff based its conclusion on the following:

- WLS DEP 6.2-1, related to changes to the acceptance criteria applied to a specific ITAAC design commitment and associated inspection, test, or analysis in Tier 1 Table 2.3.9-3, Item 3 (for control of containment hydrogen concentration for beyond-design-basis accidents) to establish consistency with the current detailed design of the plant, is reviewed and found acceptable by the staff in Section 21.4 of this SER.

- STD COL 6.2-1, as related to the containment leak rate testing program, is acceptable because the staff has determined that the requirements of 10 CFR Part 50, Appendix J, have been met.

6.3 Passive Core Cooling System

6.3.1 Introduction

The passive core cooling system is designed to provide emergency core cooling to mitigate design-basis events that involve a decrease in the reactor coolant system (RCS) inventory, such as a loss-of-coolant accident (LOCA), a decrease in heat removal by the secondary system, such as a feedwater system piping failure, or an increase in heat removal by the secondary system, such as a steam system piping failure. It also provides core cooling for shutdown events, such as a loss of the normal residual heat removal system during a shutdown operation. The passive core cooling system is designed to perform the following safety-related functions:

- emergency core decay heat removal
- RCS emergency makeup and boration
- safety injection
- containment sump pH control

During long-term operation, the AP1000 passive core cooling system must withstand the effects of debris loading on the containment recirculation screens IRWST screens and the fuel assemblies. The concern that debris may lead to unacceptable head loss for the recirculating flow was raised in GSI-191 and it is the topic of BL 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors," and Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during

Design Basis Accidents at Pressurized-Water Reactors." Section 6.3 of the AP1000 DCD includes an evaluation of this issue and Section 6.2.1.8 of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," includes the staff's review, which was performed in accordance with the NRC-approved evaluation methodology.

In order to support long term operation in a closed loop configuration, the AP1000 passive core cooling system must also achieve a sufficient condensate return rate such that inventory in the IRWST is maintained in order to retain the heat transfer capability of the passive residual heat removal (PRHR) heat exchanger (HX) (and return condensate to the sump during recirculation). Water is steamed from the IRWST during transients that require the PRHR-HX to remove decay heat from the RCS. The steam that reaches the containment shell condenses and returns to the IRWST through a gutter system. WLS DEP 3.2-1, a departure from the AP1000 DCD requested by the applicant reviewed in Section 21.1 of this report, proposes design changes to improve condensate return to the IRWST and quantifies the condensate losses associated with the pressurizing of the containment atmosphere, condensation on heat sinks within the containment, and from dripping or splashing from structures and components attached to the containment.

6.3.2 Summary of Application

WLS COL FSAR, Revision 11, Section 6.3, incorporates by reference AP1000 DCD, Revision 19, Section 6.3. In addition, in WLS COL FSAR Section 6.3.8.1, the applicant provided the following:

Departures

- WLS DEP 3.2-1 and WLS DEP 6.3-1

The applicant provided additional information in Section 6.3 of the WLS COL FSAR about WLS DEP 3.2-1 and WLS DEP 6.3-1 related to design modifications to the condensate return portion of the Passive Core Cooling System and quantifying the duration that the PRHR-HX can maintain safe shutdown conditions, respectively. This information, as well as related WLS DEP 3.2-1 and WLS DEP 6.3-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.1 of this SER.

AP1000 COL Information Items

- STD COL 6.3-1

The applicant provided information in STD COL 6.3-1 to address AP1000 COL Information Item 6.3-1 identified in AP1000 DCD Table 1.8-2, "Summary of AP1000 Standard Plant Combined License Information Items." STD COL 6.3-1 describes the features of the administrative procedures implementing a containment cleanliness program to limit the amount of debris that might be left in the containment following refueling and maintenance outages. STD COL 6.3-1 is discussed in WLS COL FSAR Section 6.3.8.1, "Containment Cleanliness Program." WLS COL FSAR Section 1.9, incorporates by reference AP1000 DCD Section 1.9, "Compliance with Regulatory Criteria." In addition, in WLS COL FSAR Section 1.9, the applicant provided the following information related to the effect of debris accumulation on long-term cooling:

- STD COL 1.9-2

The applicant provided information in STD COL 1.9-2 to address BL 03-01 and GL 04-02, both relating to the potential impact of containment debris on emergency recirculation during design basis accidents, identified in the AP1000 DCD Section 1.9.4.2.3, Table 1.9-2 and actualized in WLS COL FSAR Section 1.9.5.5, "Operational Experience," and Table 1.9-204, "Generic Communications Assessment."

- STD COL 1.9-3

The applicant provided information in STD COL 1.9-3 to address GSI-191, relating to the performance of a PWR sump during ECCS operation, identified in the AP1000 DCD Section 1.9.4.2.3, Table 1.9-2 and actualized in WLS COL FSAR Section 1.9.4.1, "Review of NRC List of Unresolved Safety Issues and Generic Safety Issues," and related Table 1.9-203.

6.3.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. In conducting its review of STD COL 6.3-1, the staff used the guidance and staff positions of RG 1.82, Revision 3, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors," and the "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02," for NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology."

The changes proposed in WLS DEP 3.2-1 and WLS DEP 6.3-1 are also required to meet NRC general design criteria, which also apply to the AP1000 DCD, 10 CFR Part 50, Appendix A, GDC 34, "Residual heat removal," as it applies to the capability of the PRHR-HX to perform safety related safe shutdown cooling of the RCS. Additionally, WLS DEP 3.2-1 and WLS DEP 6.3-1 are required to meet GDC 44, "Cooling Water," as it applies to the ability of the containment systems to transfer heat from the PRHR-HX to the ultimate heat sink via the passive containment cooling system.

6.3.4 Technical Evaluation

The staff reviewed WLS COL FSAR Section 6.3, and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the passive core cooling system. The results of the staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the DC and use this review to evaluate subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the VEGP reference COL application were equally applicable to the WLS COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from responses to RAIs.
- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed and that those changes were actualized in the WLS COL FSAR.
- The staff verified that the site-specific differences, if any, did not adversely affect any previous relevant evaluation or conclusion.

The staff completed its review and concluded that the evaluation performed for the standard content is directly applicable to the WLS COL application. This standard content material is identified in this report by use of italicized, double-indented formatting. Section 1.2.3 of this report provides an explanation of why the standard content material from the SER for the VEGP reference COL application includes evaluation material from the SER for the BLN COL application.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 6.3.4:

> *<u>AP1000 COL Information Items</u>*
>
> - *STD COL 6.3-1*
>
> *The applicant provided additional information in STD COL 6.3-1 to address COL Action Item 6.2.1.8.1-1 identified in NUREG-1793 and COL Information Item 6.3-1 identified in Table 1.8-2 of the AP1000 DCD. The applicant added information to BLN COL FSAR Section 6.3.8.1, "Containment Cleanliness Program," providing details of the program and procedures to minimize the amount of debris that might be left in containment following refueling and maintenance outages, including requirements for cleanliness inspections and limits on materials introduced into containment. TVA states that the cleanliness program will be consistent with the evaluation discussed in the AP1000 DCD.*
>
> *In its June 9, 2009, response to [RAI 30, Question 06.02.02-1], the applicant addressed the changes made to Revision 17 of the AP1000 DCD in APP-GW-GLE-002 and staff questions on cleanliness measurements with a modification to STD COL 6.3-1. This included adding that the cleanliness program will meet the DCD limits on latent debris, that housekeeping procedures will be implemented to return work areas to original conditions upon completion of work, and that a sampling program will be used to quantify the amount of latent debris. The sampling program is stated to be consistent with NEI 04-07 Volumes 1 (guidance report) and 2 (NRC safety evaluation). The sampling will be done after containment exit cleanliness inspections, prior to start up, and the results will be evaluated post-start up. Any non-conforming results will be addressed in the Corrective Action Program.*

The resulting cleanliness program is consistent with the RG 1.82 recommendation that procedures be in place to regularly clean the containment and to control and remove foreign materials from containment. The sampling program included in STD COL 6.3-1 is required to demonstrate that the latent debris found in containment is within the AP1000 DCD specified limits of 130 pounds, of which, up to 6.6 pounds may be fibrous material. The DCD specified limits were demonstrated to be acceptable through scale testing and analysis. Thus, STD COL 6.3-1 is consistent with the RG 1.82 recommendation that the cleanliness program be correlated to the amount of debris used in the long term cooling analysis. It is appropriate that the sampling program be in accordance with NEI 04-07, Volumes 1 and 2, because these documents contain the most recent NRC-approved evaluation methodology for cleanliness programs. The response to [RAI 30, Question 06.02.02-1] is acceptable and incorporation of the changes to STD COL 6.3-1 in the BLN FSAR will be tracked as Confirmatory Item 6.3-1.

The staff reviewed the following information in the BLN COL FSAR as it relates to the effect of debris accumulation on long term cooling:

- *STD COL 1.9-3*

The applicant added information to Section 1.9.4.2.3, "New Generic Issues," regarding Issue 191. The applicant states that the design aspects are addressed by the AP1000 DCD and the COL applicant portions are the protective coatings program discussed in BLN COL FSAR Section 6.1.2.1.6 and the containment cleanliness program discussed in BLN COL FSAR Section 6.3.8.1. The staff agrees that these are the only two COL items identified in the staff's review of GSI-191 from Section 6.2.1.8 of NUREG-1793.

- *STD COL 1.9-2*

The applicant added line items for Bulletin 03-01 and GL 04-02 in Table 1.9-204, "Generic Communications Assessment." The new information states that the design aspects are addressed in the AP1000 DCD and that the COL applicant aspects are addressed in BLN COL FSAR Section 6.3 for Bulletin 03-01 and BLN COL FSAR Section 6.3.8.1 for GL 04-02. The staff agrees that the design aspects of these generic communications are addressed in the staff's review of GSI-191 from Section 6.2.1.8 of NUREG-1793. The COL applicant aspects are addressed in the staff's review of BLN COL FSAR Section 6.1.2.1.6 and BLN COL FSAR Section 6.3.8.1.

<u>Resolution of Standard Content Confirmatory Item 6.3-1</u>

Confirmatory Item 6.3-1 required the applicant to update its FSAR to include the information related to the cleanliness program provided in the BLN applicant's above-mentioned June 9, 2009, response to [RAI 30, Question 60.02.02-1] (which was endorsed by the VEGP applicant). The NRC staff verified that the VEGP COL FSAR was appropriately updated with this information. As a result, Confirmatory Item 6.3-1 is resolved.

6.3.5 Post Combined License Activities

There are no post COL activities related to this section.

6.3.6 Conclusion

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to the passive containment cleanliness program, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements. In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the regulatory requirements and guidance discussed in Section 6.3.3 of this report. The staff based its conclusion on the following:

- WLS DEP 3.2-1 and WLS DEP 6.3-1, related to design modifications to the condensate return portion of the Passive Core Cooling System and quantifying the duration that the PRHR-HX can maintain safe shutdown conditions, respectively, are reviewed and found acceptable by the staff in Section 21.1 of this SER.

- STD COL 6.3-1 is acceptable because the containment cleanliness program complies with the guidance in RG 1.82.

- STD COL 1.9-2, related to BL 03-01 and GL 04-02, is acceptable because the only two items that need to be addressed by the COL applicant have been resolved. The protective coatings program is evaluated in Section 6.1.2 of this report, and the containment cleanliness program is evaluated under STD COL 6.3-1.

- STD COL 1.9-3, related to GSI-191, is acceptable because the only two items that need to be addressed by the COL applicant have been resolved. The protective coatings program is evaluated in Section 6.1.2 of this report, and the containment cleanliness program is evaluated under STD COL 6.3-1.

6.4 Habitability Systems

6.4.1 Introduction

There are a set of systems which provide habitability functions in the design and for the operation of WLS. These systems include the nuclear island non-radioactive ventilation system (VBS), the main control room (MCR) emergency habitability system (VES), the radiation monitoring system (RMS), the plant lighting system (ELS), and the fire protection system (FPS).

6.4.2 Summary of Application

WLS COL FSAR, Revision 11, Section 6.4, incorporates by reference AP1000 DCD, Revision 19, Section 6.4. In addition, in WLS COL FSAR Section 6.4, the applicant provided information to address the following:

Departures

- WLS DEP 6.4-1

The applicant provided information about WLS DEP 6.4-1 in Section 6.4 of the FSAR related to design changes affecting habitability of the MCR and changes to the calculated doses to control room operators. This information, as well as related WLS DEP 6.4-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.2 of this report.

- WLS DEP 6.4-2

The applicant provided information about WLS DEP 6.4-2 in Section 6.4 of the FSAR related to design changes affecting habitability of the MCR and changes to the maximum temperatures and heat generated in the MCR. This information, as well as related WLS DEP 6.4-2 information appearing in other chapters of the FSAR, is reviewed in Section 21.3 of this report.

AP1000 COL Information Items

- STD COL 6.4-1

The applicant provided a list of onsite chemicals to supplement the list of chemicals identified in AP1000 DCD Table 6.4-1. STD COL 6.4-1 is discussed in WLS COL FSAR Chapter 6, Sections 6.4.4, "System Safety Evaluation"; 6.4.4.2, "Toxic Chemical Habitability Analysis"; and 6.4.7, "Combined License Information," as well as in WLS COL FSAR Sections 2.2.3.1.1.4, 2.2.3.1.3, and 2.2.3.1.4. The chemicals in Table 6.4-202 associated with STD COL 6.4-1 (as annotated in the left margin) include: hydrogen (both in a gas and liquid form); nitrogen; carbon dioxide; hydrazine, morpholine; sulfuric acid; sodium hydroxide; fuel oil; sodium molybdate; sodium hexametaphosphate; sodium hypochlorite and ammonium comp. polyethoxylate. In a November 4, 2010, letter, the applicant endorsed the June 17, 2010, letter from VEGP regarding the storage of standard chemicals described under STD COL 6.4-1. In an April 25, 2011, letter, the WLS COL applicant endorsed the July 30, 2010, letter from the VEGP applicant that proposed modifications to the WLS COL FSAR related to the maximum size and stated location of the liquid hydrogen storage tank.

- STD COL 6.4-2

The applicant provided information in STD COL 6.4-2 to address AP1000 COL Information Item 6.4-2 regarding the procedures and training for control room (CR) habitability pursuant to the resolution of GSI-83, "Control Room Habitability." STD COL 6.4-2 is discussed in WLS COL FSAR Sections 6.4.3, "System Operation"; and 6.4.7, "Combined License Information."

- WLS COL 6.4-1 and WLS COL 9.4-1b

The applicant provided WLS COL 6.4-1 to address AP1000 COL Information Item 6.4-1. The applicant provided information in the WLS COL FSAR regarding the storage of plant-specific hazardous chemicals. WLS COL 6.4-1 is discussed in WLS COL FSAR Chapter 6, Sections 6.4.4, "System Safety Evaluation"; and 6.4.4.2, "Toxic Chemical Habitability Analysis"; and 6.4.7, "Combined License Information," as well as in WLS COL FSAR Sections 2.2.3.1.1.4, 2.2.3.1.3, and 2.2.3.1.4.

Supplemental Information

- STD SUP 6.4-1

The applicant provided information in STD SUP 6.4-1 to address Control Room (CR) doses for accident analyses in the downwind unit of a dual-unit site. STD SUP 6.4-1 is discussed in WLS COL FSAR Section 6.4.4.1, "Dual Unit Analysis."

6.4.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. The acceptance criteria associated with the relevant requirements of NRC regulations for habitability systems are given in NUREG-0800, Section 6.4. MCR habitability is addressed in the following:

- GDC 4, "Environmental and Dynamic Effects Design Bases," as it relates to SSCs important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions associated with postulated accidents

- GDC 5, "Sharing of Structures, Systems and Components," as it relates to ensuring that sharing among nuclear power units of SSCs important to safety will not significantly impair the ability to perform safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining unit(s)

- GDC 19, "Control Room," as it relates to maintaining the nuclear power unit in a safe condition under accident conditions and providing adequate radiation protection

- 10 CFR 50.34(f)(2)(xxviii), as it relates to evaluations and design provisions to preclude certain MCR habitability problems

- 10 CFR 52.80(a), which requires that a COL application address the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the COL, the provisions of the Atomic Energy Act of 1954, as amended, and NRC regulations

- NUREG-0737, "Clarification of TMI [Three Mile Island] Action Plan Requirements"

- TMI Action Plan, Item III.D.3.4, "Control Room Habitability"

- RG 1.78, Revision 1, "Evaluating the Habitability of a Nuclear Power Plant Control Room during a Postulated Hazardous Chemical Release"

- RG 1.52, Revision 3, "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"

- RG 1.196, "Control Room Habitability at Light Water Nuclear Power Reactors," May 2003

6.4.4 Technical Evaluation

The staff reviewed WLS COL FSAR Section 6.4, and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the WLS COL application and incorporated by reference addresses the required information relating to habitability systems. The results of the staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the DC and use this review to evaluate subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the VEGP reference COL application were equally applicable to the WLS COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR to the WLS COL FSAR, except for the evaluation of STD COL 6.4-1 and STD SUP 6.4-1. For these two items, the staff compared the BLN COL FSAR, Revision 2, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from responses to RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed and that those changes were actualized in the WLS COL FSAR.

- The staff verified that the site-specific differences, if any, did not adversely affect any previous relevant evaluation or conclusion.

The staff completed its review and concluded that the evaluation performed for the standard content is directly applicable to the VEGP COL application. This standard content material is identified in this report by use of italicized, double-indented formatting. Section 1.2.3 of this report provides an explanation of why the standard content material from the SER for the VEGP reference COL application includes evaluation material from the SER for the BLN COL application. Any confirmatory items in the standard content material retain the numbers assigned in the VEGP SER.

<u>Control Room Radiological Habitability</u>

The staff evaluated the WLS COL FSAR incorporation by reference of the AP1000 DCD evaluation of control room habitability, taking into consideration the WLS site characteristics.

Compliance with the control room habitability dose requirements of GDC-19 requires that the applicant show that, for a plant located at the site, the control room provides adequate radiation protection to ensure that radiation exposures shall not exceed 0.05 Sv (5 radiation equivalent man (rem)) total effective dose equivalent (TEDE) to permit access and occupancy of the control room under accident conditions for the duration of the accident. The applicant did not provide site-specific doses in the control room for the design basis accidents referenced in the AP1000 DCD, Revision 19, but instead incorporated by reference the analysis of the radiological control room habitability from the AP1000 DCD, Revision 19, Section 6.4.4, “System Safety Evaluation.”

AP1000 DCD, Revision 19, Chapter 6.4, provided the results of the analysis of the control room radiological consequences for the design basis accidents analyzed for offsite radiological consequences in AP1000 DCD Chapter 15. The details and assumptions used in modeling the radiological consequences to control room operators were described in detail in the AP1000 DCD Section 15.6.5.3. The staff’s technical evaluation of the information incorporated by reference related to the modeling of the control room in design basis accident radiological consequences analyses is documented in a supplement to NUREG-1793. The design basis accident control room radiological consequences analyses in the AP1000 DCD used design reference (site parameter) values for the atmospheric dispersion factors (χ/Qs), in place of site-specific values. The atmospheric dispersion factors are the only input to the design base accident (DBA) radiological consequences analyses that are impacted by the site characteristics. The applicant provided and discussed the WLS site-specific site characteristic control room χ/Qs in resolution of WLS COL 2.3-4. The WLS site characteristic control room χ/Qs are given in WLS COL FSAR Tables 2.3-222 and 2.3-223. In Section 2.3 of this report, the staff discusses its review of the resolution to WLS COL 2.3-4, related to the WLS site characteristic χ/Qs. All other inputs and assumptions in the radiological consequences analyses remain the same as in the AP1000 DCD. Smaller χ/Q values are associated with greater dilution capability, resulting in lower radiological doses. When comparing a DCD site parameter χ/Q value and a site characteristic χ/Q value, the site is acceptable for the design if the site characteristic χ/Q value is smaller than the site parameter χ/Q value. Such a comparison shows that the site has better dispersion characteristics than that required by the reactor design.

For each time averaging period, the WLS site-specific control room χ/Q values are less than the design reference control room χ/Q values used in AP1000 DCD, Revision 19, for the radiological consequence analyses for each of the design basis accidents. Since (1) the result of the radiological consequence analysis for a design basis accident during any time period of radioactive material release from the plant is directly proportional to the atmospheric dispersion factor for that time period, and (2) the WLS site characteristic χ/Q values are less than the comparable design site parameter χ/Q values in the DCD for all time periods for each accident, the WLS site-specific total dose for each design basis accident is therefore less than the AP1000 DCD, Revision 19, reference total dose for each design basis accident.

Since the AP1000 DCD, Revision 19, analyses show that the radiological consequences in the control room meet the regulatory dose requirements of GDC-19 by resulting in a TEDE less than 0.05 Sv (5 rem), and since the WLS site-specific design basis accident control room radiological consequences are less than those for the AP1000 DCD, Revision 19, then the staff concludes that the applicant has adequately shown that the design basis accident control room radiological consequences meet the dose requirements of GDC-19, and is therefore acceptable.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 6.4.4:

<u>AP1000 COL Information Items</u>

- *STD COL 6.4-1*

The following portion of this technical evaluation section is reproduced from Section 6.4.4 of the BLN SER. The staff notes that Table 6.4-202 in the BLN FSAR, Revision 2, is equivalent to Table 6.4-201 in the VEGP COL FSAR. Information in the BLN COL FSAR having a left margin annotation STD SUP 6.4-2 was assigned a left margin annotation of STD SUP 6.4-3 in the VEGP COL FSAR, and revisions proposed by the applicant, described below, combined the information from STD SUP 6.4-3 and STD COL 6.4-1 under a single left margin annotation of STD COL 6.4-1. Therefore, the evaluation of STD COL 6.4-1 in this SER includes references to material identified as STD SUP 6.4-2 in the BLN COL FSAR.

- *STD SUP 6.4-2*

STD SUP 6.4-2 provides the chemical names, state of the chemical, quantity and location of the chemicals. The chemicals include: hydrogen (both in a gas and liquid form), hydrazine, morpholine, sulfuric acid, sodium hydroxide, fuel oil, sodium molybdate (molybdic acid, disodium salt), sodium hexametaphosphate, and sodium hypochlorite.

Subsequent to the issuance of Section 2.2.3 of this report, the staff reviewed the applicant's inventory of chemicals contained in STD SUP 6.4-2 for threats to CR habitability. The staff has determined, with the exception of hydrazine, that the STD SUP 6.4-2 chemicals do not warrant additional analysis for CR habitability because they do not exceed the immediate danger to life and health (IDLH) limit at ground level at the location of the CR.

Regarding hydrazine, a further analysis with the HABIT computer code (RG 1.78) confirms that the hydrazine may exceed the IDLH limit at ground level. However, additional analysis shows that the hydrazine concentrations at the CR intake and inside the CR will not exceed the IDLH limit when crediting the design of the CR ventilation intake located at the auxiliary building (57 ft. above ground), calculations show concentrations much less than the IDLH limit. These results are based on a temperature of 25 °C and a wind speed of 1 m/sec, with meteorology F class, which are the conditions used by the applicant and RG 1.78. Hence, it is determined that the hydrazine listed in STD SUP 6.4-2 will not pose a threat to CR habitability.

AP1000 COL Information Items

- *STD COL 6.4-1*

STD COL 6.4-1 information also provides the chemical names, state of the chemical, quantity and location of the chemicals. The chemicals include: nitrogen, carbon dioxide, and ammonium comp polyethoxylate.

Subsequent to the issuance of Section 2.2.3 of this report, the staff reviewed the applicant's inventory of chemicals listed in STD COL 6.4-1, and screened out the toxic chemicals that do not pose a threat to CR habitability. The staff has determined that with the exception of carbon dioxide the STD COL 6.4-1 chemicals do not warrant additional analysis because they do not exceed the IDLH limit at ground level at the location of the CR.

Regarding carbon dioxide, analysis with the HABIT computer code (RG 1.78) finds that carbon dioxide will not exceed the IDLH limit at ground level. This analysis is based on a temperature of 25 °C and a wind speed of 1 m/sec, with meteorology F class, which are the conditions used by the applicant and RG 1.78. Hence, it is determined that the carbon dioxide contained in STD COL 6.4-1 will not pose a threat to CR habitability.

The staff notes that the chemical analysis relied on by the COL applicant includes assumptions associated with design features, such as the intake location for the CR ventilation system. In RAI 6.4-8, the staff asked if any of the analyses of the chemicals in Table 6.4-202 credit design features, such as an elevated CR intake, to keep the chemical concentration in the CR below the IDLH levels, in which case a description of the design features credited in the safety analyses should be provided in the FSAR. This is Open Item 6.4-1.

Resolution of Standard Content Open Item 6.4-1

In a letter dated June 17, 2010, the applicant proposed modifications to Table 6.4-201 in the VEGP COL FSAR to address Open Item 6.4-1. The proposed modifications included addition of a column entitled "MCR Habitability Impact Evaluation" to the table that indicated when design features were considered in the impact evaluation, including either the MCR intake height or other design details beyond the intake height. The staff determined that the modifications sufficiently described the design assumptions considered by the applicant, and Open Item 6.4-1 is resolved. The incorporation of this modification to Table 6.4-201 into a future revision of the VEGP COL FSAR is being tracked as Confirmatory Item 6.4-1.

Resolution of Standard Content Confirmatory Item 6.4-1

Confirmatory Item 6.4-1 is an applicant commitment to revise its FSAR Table 6.4-201 to add a column entitled "MCR Habitability Impact Evaluation" that will indicate when design features are considered in the impact evaluation, including either the MCR intake height or other design details beyond the intake *height. The staff verified that VEGP COL FSAR Table 6.4-201 was appropriately revised. As a result, Confirmatory Item 6.4-1 is now closed.*

Evaluation of Additional Revisions to STD COL 6.4-1

In the letter dated June 17, 2010, the applicant proposed additional voluntary revisions to Table 6.4-201 in the VEGP COL FSAR regarding the storage of standard chemicals described under STD COL 6.4-1. The proposed revisions included changes to the chemical quantities, evaluated distances, and storage locations, as well as changes to the table organization, column headings, and table notes. The proposed revisions also included combining the chemicals listed under separately STD COL 6.4-1 and STD SUP 6.4-3 under a single left margin annotation of STD COL 6.4-1, thereby eliminating STD SUP 6.4-3.

In a letter dated July 30, 2010, the applicant proposed additional revisions to STD COL 6.4-1 related to the evaluated maximum quantity and location of the liquid hydrogen storage tank.

On April 14 and June 7, 2010, the NRC staff audited the applicant's proprietary calculation notes, APP-VES-M3C-006, entitled "Main Control Room Emergency Habitability from Toxic Chemical Effluents," Revision 0 and Revision 1 to verify the information supporting STD COL 6.4-1 and VEGP COL FSAR Table 6.4-201. As a result of these audits, the staff issued RAI 6.4-5. The applicant subsequently prepared calculation notes APP-PGS-M3C-011, entitled "AP1000 Gas Spill or Release Effects on Control Room Habitability," Revision 0 and Revision 1 that were audited by the staff on July 26 and August 23, 2010. In a letter dated September 3, 2010, the applicant proposed the following changes to the FSAR and provided the following additional information about calculated concentrations of chemicals that would occur at the MCR intake to address RAI 6.4-5:

- *Proposed to change the evaluated minimum distance between the MCR and the storage locations for liquid hydrogen, nitrogen, and carbon dioxide.*

- *For hydrogen, nitrogen, and carbon dioxide, proposed to indicate that MCR design details were considered in evaluating the potential impact to the MCR.*

- *Proposed to clarify that the MCR design details considered included MCR volume, envelope boundaries, ventilation systems, and occupancy factor.*

- *Provided information about how the analysis considered the effect of wind speeds less than 1 meter/second.*

- *Provided information about concentrations occurring at the MCR intake more than two minutes after a potential release occurs.*

- *For hydrogen, nitrogen, and carbon dioxide, provided information about concentrations occurring at the MCR intake when no building wake effects are considered.*

- *For carbon dioxide, provided information about concentrations occurring in the MCR based on a corrected conservative value for the MCR outside air exchange rate.*

In the evaluation presented in Section 2.2.3 of this SER, the staff reviewed the applicant's revised chemical inventory information listed in STD COL 6.4-1, and screened out the toxic chemicals that do not pose a threat to MCR habitability. The staff determined that, with the exception of hydrazine and carbon dioxide, the STD COL 6.4-1 chemicals do not warrant additional analysis for MCR habitability because they would not exceed the IDLH limit at ground level below the MCR ventilation intake. Hydrazine and carbon dioxide are evaluated below.

Regarding hydrazine, the NRC staff used the HABIT computer code (as referenced in RG 1.78) to confirm that hydrazine concentration may exceed the IDLH limit at ground level below the MCR intake. The staff then conducted an additional analysis showing that the hydrazine concentration at the MCR intake and inside the MCR would not exceed the IDLH limit when crediting the design of the MCR ventilation intake located at the auxiliary building (which is located 17.37 m (57 ft) above ground). The applicant annotated "IH" in VEGP COL FSAR Table 6.4-201 to indicate that the credit of MCR ventilation intake height had been taken in the safety analysis.

Regarding carbon dioxide, the NRC staff used the HABIT computer code to confirm that the carbon dioxide concentration may exceed the IDLH limit at the MCR intake. The staff then conducted an additional analysis showing that the carbon dioxide concentration inside the MCR would remain below the IDLH limit.

Based on the FSAR revisions proposed and additional information provided by the applicant and the confirmatory analyses performed by the staff, the staff determined that the hydrazine and carbon dioxide would not pose a threat to MCR habitability, and RAI 6.4-5 is closed.

The incorporation of the revisions to STD COL 6.4-1 Table 6.4-201 into a future revision of the VEGP COL FSAR, as proposed in letters from the applicant dated June 17, July 30, and September 3, 2010, is being tracked as Confirmatory Item 6.4-2.

Resolution of Standard Content Confirmatory Item 6.4-2

Confirmatory Item 6.4-2 is an applicant commitment to revise its FSAR Table 6.4-201 to revise information related to standard chemicals. The staff verified that VEGP COL FSAR Table 6.4-201 was appropriately revised. As a result, Confirmatory Item 6.4-2 is now closed.

The following portion of this technical evaluation section is reproduced from Section 6.4.4 of the BLN SER:

- *STD COL 6.4-2*

The NRC staff reviewed STD COL 6.4-2, related to COL Information Item 6.4-2 and COL Action Item 6.4-1, included under Section 6.4.3 of the BLN COL FSAR. The applicant stated that procedures and training for CR habitability are written in accordance with Section 13.5 for CR operating procedures, and Section 13.2 for operator training. In Section 6.4.3 of the FSAR, the applicant states that the procedures and training will be verified to be consistent with the intent of GSI-83.

However, the level of detail provided in the standard portion of BLN COL FSAR Section 6.4.3 is not adequate to determine if the regulatory requirements are met. As a result, the staff issued RAI 6.4-7, which asked the applicant to provide in the FSAR the essential elements of the training and procedures necessary to demonstrate that the regulatory requirements are met. The staff questioned what the operators would be directed and trained to do to meet the recommendations in RG 1.196. Specifically, in RAI 6.4-7, the staff requested information addressing the following:

- *RG 1.78, Regulatory Position C.5, "Emergency Planning"*
- *RG 1.196, Regulatory Position 2.5, "Hazardous Chemicals"*
- *RG 1.196, Regulatory Position 2.2.1, "Comparison of System Design, Configuration, and Operation with the Licensing Basis"*
- *RG 1.196, Regulatory Position 2.7.1, "Periodic Evaluations and Maintenance"*

The resolution of RAI 6.4-7 is identified as Open Item 6.4-2.

<u>Resolution of Standard Content Open Item 6.4-2</u>

The BLN response to RAI 6.4-7 dated January 5, 2010, stated that the operational aspects of the identified guidance had been met as documented in BLN COL FSAR Appendix 1AA. The BLN applicant's response also stated that the additional information would be provided in a future revision to BLN COL FSAR Section 6.4.3, addressing how procedures, testing and training related to CR habitability would be consistent with the above stated regulatory positions in RG 1.78 and RG 1.196. The VEGP applicant endorsed the BLN response to RAI 6.4-7 in a letter dated June 17, 2010, and committed to appropriately update Section 6.4.3 of the VEGP COL FSAR. Therefore, Standard Content Open Item 6.4-2 is resolved for the VEGP application, and incorporation of the proposed revision to Section 6.4.3 of the VEGP COL FSAR is being tracked as Confirmatory Item 6.4-3.

<u>Resolution of Standard Content Confirmatory Item 6.4-3</u>

Confirmatory Item 6.4-3 is an applicant commitment to revise its FSAR Section 6.4.3 to include information regarding procedures, testing and training *related to CR habitability. The staff verified that VEGP COL FSAR Section 6.4.3 was appropriately revised. As a result, Confirmatory Item 6.4-3 is now closed.*

- WLS COL 6.4-1 and WLS COL 9.4-1b

The applicant added Section 6.4.4.2, “Toxic Chemical Habitability Analysis,” at the end of AP1000 DCD Section 6.4.4, “System Safety Evaluation.” In WLS COL FSAR Section 6.4.4.2, the applicant added new text and a table for WLS COL 6.4-1 to address the issue of handling local toxic gas services and monitoring. The applicant determined that no toxic gas monitoring is required, while AP1000 DCD Section 6.4.7 states, “Combined License applicants referencing the AP1000 certified design are responsible for the amount and location of possible sources of hazardous chemicals in or near the plant and for seismic Category I Class 1E hazardous chemical monitoring, as required.”

The staff reviewed the text added to AP1000 DCD Section 6.4, by WLS COL 6.4-1. WLS performed an analysis of a chlorine release event and concluded that chlorine monitors are not needed, as permitted by RG 1.78, because the results of the HABIT EXTRAN analysis indicate that under ideal conditions a pressurized liquid chlorine tractor-trailer burst-type accident would not elevate control room HVAC intake concentrations beyond IDLH values. The applicant incorporated Table 6.4-202 from the VEGP COL FSAR, which consists of an updated list of toxic chemicals and parameters that have been reviewed and determined acceptable with respect to impact on the control room. In RAI 19, Questions 06.04-1 through 06.04-7, the staff requested that the applicant provide the results of its analysis of an offsite accidental release of chlorine, the source and assumptions of the input values used in the HABIT analysis, the number of operators to be protected from toxic gas release, operator response during chlorine release, and the definition of “steam vent” where atmospheric dispersion factors were considered.

Regarding the chlorine analysis methodology, in a May 31, 2012, response, the applicant stated that alternative modeling software was used. The applicant used a combination of ALOHA and HABIT to properly analyze the accidental release of chlorine. ALOHA code is limited to a 1 hour run time and terminates prior to the chemical release reaching the control room intake 5100 meter (m) (3.2 miles (mi)) from the release point. Therefore, a two phase analysis approach was developed using ALOHA for the initial phase of the event when the release behaves as a heavier-than-air gas until the chlorine cloud concentration decreases to 10,000 parts per million (ppm), and using HABIT for the remaining distance because according to the user’s manual for ALOHA, when the concentration of heavy gas drops below about 10,000 ppm, the chlorine behaves like a neutrally buoyant gas. HABIT treats the input chemical as neutrally buoyant and can be modified to allow for longer run times (while ALOHA is capped at 1 hour), therefore, HABIT can model long enough to capture the plume peak, the trailing plume edge, and the control room peak concentration. The staff finds the applicant’s proposed modeling approach for accidental off-site releases of chlorine, described in the previous paragraph and in WLS COL FSAR Section 2.2.3.1.3.3, is acceptable and, therefore, considers RAI 19, Questions 06.04-1 through 06.04-7 resolved. With regard to the other chemical analysis provided, the staff finds the applicant’s results also acceptable because the assumptions were conservative and the modeling approach reasonable, resulting in concentrations that remained below the IDLH.

The staff also issued RAI 103, Question 02.02.03-8, regarding analysis details on chemicals Methoxypropylamine (MPA) and Dimethylamine (DMA). In a March 28, 2012, response, the applicant provided information pertaining to the analysis of MPA and DMA. The applicant selected U.S. Environmental Protection Agency (EPA) code SLAB to conduct the analysis. The key points considered in the selection of SLAB are that the model is developed specifically to deal with dispersion of dense gases, and that the model has the capability of predicting concentrations at an elevated control room intake. The staff notes that the SLAB code enabled the applicant to correctly credit the heavy gas aspects of the release and also correctly account for the elevated intake in the AP1000 design. Also the applicant revised the WLS COL FSAR to indicate the credit for the elevated intake by modifying the table to add the “IH” for the chemicals in question. The staff finds the applicant’s modeling approach acceptable and, therefore, considers RAI 103, Question 02.02.03-8, resolved. The applicant’s WLS COL FSAR Sections 2.2.3.1.3, 6.4.4.2, and Table 6.4-202 reflect this discussion.

Supplemental Information

The following portion of this technical evaluation section is reproduced from VEGP SER Section 6.4.4:

> *The following portion of this technical evaluation section is reproduced from Section 6.4.4 of the BLN SER:*
>
> - *STD SUP 6.4-1*
>
> *The NRC staff reviewed STD SUP 6.4-1 related to the evaluation of CR doses in the other unit of a dual unit plant included under Section 6.4.4 of the BLN COL FSAR. The staff concludes that STD SUP 6.4-1 is acceptable because the dose to the CR operators at an adjacent AP1000 due to a radiological release from another unit is bounded by the dose to CR operators on the affected unit. Further, simultaneous accidents at multiple units at a common site are not considered to be a credible event, unless there is a reliance on shared systems between the two units. This is not the case for the AP1000 design.*

A portion of the standard technical evaluation from the VEGP COL SER is not included above. The staff concluded that the omitted portion was not relevant to WLS.

6.4.5 Post Combined License Activities

There are no post COL activities related to this section

6.4.6 Conclusion

The staff reviewed the WLS COL application and checked the referenced DCD. The staff’s review confirmed that the applicant addressed the required information relating to MCR habitability, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements. Based on its review of the application, the staff concludes that the information presented in the WLS COL FSAR is acceptable because it meets the acceptance criteria associated with the relevant requirements of NRC regulations for habitability systems given in NUREG-0800, Section 6.4. The staff based its conclusions on the following:

- WLS DEP 6.4-1, relating to design changes affecting habitability of the MCR and changes to the calculated doses to control room operators, is reviewed and found acceptable by the staff in Section 21.2 of this SER.

- WLS DEP 6.4-2, related to design changes affecting habitability of the MCR and changes to the maximum temperatures and heat generated in the MCR, is reviewed and found acceptable by the staff in Section 21.3 of this SER.

- STD COL 6.4-1 is acceptable because the chemicals do not exceed the IDLH limit at ground level at the intake of the MCR, using the regulatory guidance in RG 1.78.

- STD COL 6.4-2 is acceptable because the procedures, testing and training related to MCR habitability will be consistent with the stated regulatory positions in RG 1.78 and RG 1.196.

- WLS COL 6.4-1 and WLS COL 9.4-1b are acceptable, because the plant-specific chemicals do not exceed the IDLH limit at the intake of the MCR, using the regulatory guidance in RG 1.78.

- STD SUP 6.4-1 is acceptable because the dose to the MCR operators at an adjacent AP1000 due to a radiological release from another unit is bounded by the dose to MCR operators on the affected unit, using the regulatory guidance in NUREG-0800, Section 6.4.

6.5 Fission Product Removal and Control Systems

In the event of a design basis LOCA there is an assumed core degradation that results in a significant release of radioactivity to the containment atmosphere. This activity would consist of noble gases, particulates, and a small amount of elemental and organic iodine. Fission product removal and control systems are considered to be those systems for which credit is taken in reducing accidental release of fission products. The AP1000 design has no active system to control fission products in the containment following a postulated accident. The fission product control system is the primary containment. AP1000 DCD, Appendix 15B, “Removal of Airborne Activity from the Containment Atmosphere Following a LOCA,” discusses satisfactory removal of airborne activity (elemental iodine and particulates) from the containment atmosphere by natural removal processes (e.g., deposition and sedimentation) without the use of containment spray.

WLS COL FSAR, Revision 11, Section 6.5, incorporates by reference, with no departures or supplements, AP1000 DCD, Revision 19, Section 6.5. The staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The staff’s review confirmed that there is no outstanding issue related to this section. The results of the staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

6.6 Inservice Inspection of Class 2, 3, and MC Components

6.6.1 Introduction

Inservice inspection (ISI) programs must meet requirements of 10 CFR 50.55a, “Codes and Standards,” in which American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV)) is incorporated by reference. This section addresses the ISI of ASME Code Class 2 and 3 components. ASME Code Class 2 and 3 components must meet the applicable inspection requirements set forth in ASME B&PV Code, Section XI, “Rules for Inservice Inspection (ISI) of Nuclear Power Plant Components,” Subsections IWC and IWD.” Subsection IWC and IWD also include requirements for preservice examinations prior to initial plant startup as provided in Subarticles IWC-2200 and IWD-2200.

6.6.2 Summary of Application

WLS COL FSAR, Revision 11, Section 6.6, incorporates by reference AP1000 DCD, Revision 19, Section 6.6. In addition, in WLS COL FSAR Section 6.6, the applicant provided information to address the following:

AP1000 COL Information Items

- STD COL 6.6-1

The applicant provided information in STD COL 6.6-1 to address AP1000 COL Information Item 6.6-1. The information relates to plant-specific preservice inspection (PSI) and ISI programs. STD COL 6.6-1 is discussed in WLS COL FSAR Sections 6.6, “Inservice Inspection of Class 2, 3, and MC Components”; 6.6.1, “Components Subject to Examination”; 6.6.3.1, “Examination Methods”; 6.6.3.2, “Qualification of Personnel and Examination Systems for Ultrasonic Examination”; 6.6.3.3, “Relief Requests”; 6.6.4, “Inspection Intervals”; 6.6.6, “Evaluation of Examination Results”; and 6.6.9.1, Inspection Programs.”

- STD COL 6.6-2

The applicant provided information in STD COL 6.6-2 to address AP1000 COL Information Item 6.6-2. The information relates to preservation of component accessibility design considerations during the construction phase. This COL information item is discussed in WLS COL Sections 6.6.2, “Accessibility,” and 6.6.9.2, “Construction Activities.”

Supplemental Information

- STD SUP 6.6-1

The applicant provided information related to the design stage consideration of component accessibility to enable the performance of ISI examinations.

License Condition

- WLS COLA Part 10, “Proposed License Conditions (Including ITAAC),” License Condition 6, “Operational Program Readiness”

This proposed license condition states that the COL holder shall provide an operational program implementation schedule to support NRC inspections. For the purpose of the review of WLS COL FSAR Section 6.6, the relevant operational programs are the Inservice Inspection Program and the Preservice Inspection Program.

6.6.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. The acceptance criteria associated with the relevant requirements of NRC regulations for ISI of ASME B&PV Code Class 2 and 3 components are given in NUREG-0800, Section 6.6. The applicable regulatory requirements for acceptance of the resolution of COL information items and supplementary information on ISI and testing of ASME B&PV Code Class 2 and 3 components are established in GDC 45, “Inspection of Cooling Water System” located in 10 CFR Part 50, Appendix A, as it relates to periodic inspection of important components, such as heat exchangers and piping to assure the integrity and capability of the system. The applicable policy for acceptance of COL information items, as it relates to fully describing an operational program, is located in SECY-05-0197.

6.6.4 Technical Evaluation

The staff reviewed Section 6.6 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The staff’s review confirmed that the information in the application and incorporated by reference addresses the required information relating to the ISI of Class 2 and 3 components. The results of the staff’s evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the DC and use this review to evaluate subsequent COL applications. To ensure that the staff’s findings on standard content that were documented in the SER for the VEGP reference COL application were equally applicable to the WLS COL application, the staff undertook the following reviews:

- The staff compared the VEGP to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed and that those changes were actualized in the WLS COL FSAR.

- The staff verified that the site-specific differences, if any, did not adversely affect any previous relevant evaluation or conclusion.

The staff has completed its review and found the evaluation performed for the standard content is directly applicable to the WLS COL application. This standard content material is identified in this report by use of italicized, double-indented formatting. Section 1.2.3 of this report provides an explanation of why the standard content material from the SER for the VEGP reference COL application includes evaluation material from the SER for the BLN COL application.

The following portion of this technical evaluation section is reproduced from Section 6.6.4 of the VEGP SER:

> *<u>AP1000 COL Information Items</u>*
>
> - *STD COL 6.6-1*
>
> *In Section 6.6 of the NRC staff FSER (NUREG-1793, dated September 2004), the staff concluded that the AP1000 ISI program for ASME Code Class 2 and 3 components is acceptable and meets the requirements of 10 CFR 50.55a with regard to the preservice and inservice inspectability of these components. The specific version of the ASME Code, Section XI used as the baseline Code in the AP1000 certified design, is the 1998 Edition up to and including the 2000 Addenda. It should be noted that the staff did not identify any portions of the AP1000 ISI program for Class 1, 2, and 3 components that were excluded from the scope of the staff's review of the AP1000 DC (as the staff did for inservice testing of valves in AP1000 FSER Section 3.9.6.4). Therefore, the staff's conclusions regarding the acceptability of the AP1000 ISI program based on the 1998 Edition up to and including the 2000 Addenda of the ASME Code, Section XI with regard to preservice and inservice inspectability of Class 2 and 3 components remains unchanged. The staff's evaluation of the operational program aspects of the ASME Code Class 2 and 3 ISI program is addressed with Class 1 ISI in Section 5.2.4 of this SER. The review of the COL applicant's supplemental information also includes the adequacy of the ISI program for reactor containment (Class MC). In Revision 17 of the AP1000 DCD, Class MC components were added to the DCD, Section 6.6, as being within the scope of the ISI Program. The COL applicant incorporated DCD Section 6.6 in its entirety under Revision 1 of its FSAR. Accordingly, the staff's evaluation of this section focused on the acceptability of the COL applicant's supplemental information and responses to AP1000 COL information items and action items as they relate to ISI of ASME Code Class 2, 3, and MC components.*
>
> *As part of STD COL 6.6-1, the COL applicant added to the end of DCD Section 6.6.2 words to state that the initial ISI program will incorporate the latest Edition and Addenda of the ASME Code (Section XI) approved in 10 CFR 50.55a(b) on the date 12 months before initial fuel load. The COL applicant stated that successive 120-month inspection intervals must comply with the requirements of the latest Edition and Addenda of the Code incorporated by reference in 10 CFR 50.55a(b) 12 months before the start of the 120-month interval, subject to the limitations and modifications listed in 10 CFR 50.55a(b). The requirements in 10 CFR 50.55a(g) state that inservice examinations of components and system pressure tests conducted during the initial 120-month inspection interval must comply with the requirements in the latest Edition and*

Addenda of the Code incorporated by reference in paragraph (b) of 10 CFR 50.55a on the date 12 months before the date scheduled for initial loading of fuel under a COL under 10 CFR Part 52. The staff concludes that the supplemental information provided by the COL applicant meets the NRC's regulations and is, therefore, acceptable.

As part of STD COL 6.6-1, the COL applicant added to the end of DCD Section 6.6.1 words to state that Class 2 and 3 components are included in the equipment designation list contained in the ISI program. The requirements in 10 CFR 50.55a(g)(3)(ii) state, in part, that Class 2 and 3 components be designed and provided with access to enable the performance of ISI examinations. In addition, the inclusion of Class 2 and 3 components is consistent with the requirements of an ISI program as defined under ASME Section XI, and is, therefore, acceptable. The staff concludes that the supplemental information provided by the COL applicant meets the NRC's regulations and is, therefore, acceptable.

In Section 6.6 of the FSER (NUREG-1793), the staff identified COL Action Item 6.6-1 in which the COL applicant will prepare a PSI program and an ISI program for ASME Code, Class 2 and 3 systems, components and supports. The PSI and ISI programs will address the equipment and techniques used. As part of STD COL 6.6-1, the COL applicant describes the use of visual, surface, ultrasonic, alternative examination techniques, and the use of automated equipment to perform the examinations. The COL applicant referenced the relevant portions of the ASME Code, Section XI to describe the nondestructive examination techniques and alternative examinations. The COL applicant also added information to describe the 120-month inspection interval as defined by IWB-2400 for Inspection Program B and the evaluation of examination results as defined by the ASME Code, Section XI, paragraphs IWC-, IWD-, IWE-, or IWF-3400 acceptance criteria. In addition, the COL applicant appropriately referenced 10 CFR 50.55a(b)(2)(xix) and IWA-2240 as described in the 1997 Addenda of the ASME Code, Section XI when applying alternative examination provisions. The supplemental information provided by the COL applicant meets the requirements in 10 CFR 50.55a, the ASME Code, Section XI, and the guidelines in RG 1.206, Section C.III.1, Chapter 6, C.I.6.6.3, and is, therefore, acceptable. Based on the discussion above, the staff concludes that the supplemental information under STD COL 6.6-1 is acceptable.

- *STD COL 6.6-2*

As part of STD COL 6.6-2, the COL applicant states that during the construction phase of the project, anomalies and construction issues are addressed using change control procedures. Modifications reviewed following DC will adhere to the same level of review as the certified design, thus, control of accessibility is maintained during post-DC activities. Control of accessibility for inspectability and testing during post-DC activities is provided via procedures for design control and plant modifications. In the NRC staff's FSER (NUREG-1793), the staff identified COL Action Item 6.6-2, which recommends COL applicants referencing the AP1000 certified design address the controls to preserve accessibility and *inspectability for ASME Code, Section III, Class 2 and 3 components and piping during construction or other post-DC activities. The NRC staff reviewed the applicant's proposed resolution of COL Action Item 6.6-2 using NUREG-0800, Section 6.6. The staff finds that the accessibility needed to perform PSI/ISI examinations is maintained during the design, construction and operational phases, which satisfies NUREG-0800, Section 6.6 recommendations for accessibility. In addition, the supplemental information meets the regulations under 10 CFR 50.55a(g)(3)(ii), which requires that Class 1, 2, and 3 components be designed and provided with access that enables the performance of ISI examinations, and the requirements under ASME Code, Section XI, IWA-1500. Based on the discussion above, the staff concludes that STD COL 6.6-2 is acceptable.*

<u>Supplemental Information</u>

- *STD SUP 6.6-1*

As part of STD SUP 6.6-1, the COL applicant added supplemental information to the AP1000 DCD, Section 6.6.2, to address accessibility of Class 2, 3, and Class MC pressure retaining components to permit preservice and inservice examinations. Factors considered, such as examination requirements, techniques, accessibility, geometry, and material selections, are used in establishing the designs with the goals being to eliminate uninspectable components, reduce occupational radiation exposure, reduce inspection times, allow state-of-the-art inspection systems, and enhance detection and the reliability of flaw characterization.

The requirements in 10 CFR 50.55a(g)(3)(ii) state, in part, that Class 2 and 3 components be designed and provided with access to enable the performance of ISI examinations. ASME Code, Section XI, IWA-1500 requires that access be provided to enable the performance of ISI examinations, along with design considerations to render ISI practical. The staff finds that the supplemental information under STD SUP 6.6-1 meets the requirements of 10 CFR 50.55a and ASME Code, Section XI, and is, therefore, acceptable.

<u>License Condition</u>

- *Part 10, License Condition 6*

The COL applicant proposed a license condition for BLN for all operational programs requiring that the licensee shall submit to the appropriate Director of the NRC a schedule, no later than 12 months after issuance of the COL, that supports planning for and conduct of NRC inspections of operational programs. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until either the operational program has been implemented or the plant has been placed into commercial service. A separate license condition for PSI and ISI program implementation requirements is not necessary in the BLN COL FSAR since it is a requirement under 10 CFR 50.55a. However, submittal of the schedule for the PSI and ISI program *development is necessary to plan for and conduct NRC inspections during construction. The staff finds that this schedule will enable the staff to adequately plan and schedule inspections of the PSI and ISI programs during the construction phase. This proposed license condition is consistent with the policy established in SECY-05-0197, and is acceptable.*

6.6.5 Post Combined License Activities

For the reasons discussed in the technical evaluation section above, the staff finds the following license condition associated with the PSI and ISI programs acceptable:

- License Condition (6-3) – No later than 12 months after issuance of the COL, the licensee shall submit to the appropriate Director of NRO a schedule that supports planning for and conduct of NRC inspections of the PSI and ISI programs. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until the PSI and ISI programs have been fully implemented.

6.6.6 Conclusion

The staff reviewed the application and checked the referenced AP1000 DCD. The staff's review confirmed that the applicant addressed the required information relating to ISI of ASME Code Class 2 and 3 components, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the requirements of GDC 45 and 10 CFR 50.55a. The staff based its conclusion on the following:

- STD COL 6.6-1 is acceptable because the staff concluded that the applicant's AP1000 ISI program for ASME Code Class 2, 3, and Metal Containment (MC) components is acceptable and meets the requirements of 10 CFR 50.55a with regard to the preservice and inservice inspectability of these components.

- STD COL 6.6-2 is acceptable because the staff concluded that the accessibility needed to perform PSI/ISI examinations is maintained during the design, construction and operational phases, and satisfies NUREG-0800, Section 6.6 acceptance criteria for accessibility.

- STD SUP 6.6-1 is acceptable because the staff concluded that accessibility to perform ISI examinations would be incorporated into the design, and satisfies the regulations under 10 CFR 50.55a(g)(3)(ii).

7 INSTRUMENTATION AND CONTROLS

Nuclear power plant instrumentation senses various plant parameters and transmits appropriate signals to the control systems during normal operation and to the reactor trip and engineered safety feature systems during abnormal and accident conditions. The information provided in this chapter emphasizes those instruments and associated equipment that constitute the protection and safety systems.

7.1 Introduction

7.1.1 Introduction

The Westinghouse AP1000 Design Control Document (DCD), Revision 19, contains Combined License (COL) Information Item 7.1-1 that requires the COL applicant to address setpoint calculations for protective functions

7.1.2 Summary of Application

Section 7.1 of the William States Lee III Nuclear Station (WLS) COL Final Safety Analysis Report (FSAR), Revision 11, incorporates by reference AP1000 DCD, Revision 19, Section 7.1, with the departures and/or supplements with respect to AP1000 DCD, Section 7.1.6.1, pertaining to "Setpoint Calculations for Protective Functions." This change to AP1000 DCD, Section 7.1.6.1 addressed the new COL Information Item (COL 7.1-1), which was incorporated in AP1000 DCD, Revision 18.

To address the above COL Information Item 7.1-1, the applicant provided the following additional information in the COL application:

AP1000 COL Information Item

- STD COL 7.1-1

Standard (STD) COL 7.1-1 addresses setpoint calculations for protective functions.

7.1.3 Regulatory Basis

The regulatory basis of the information incorporated by reference in the WLS COL application is addressed in NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of NRC regulations for Instrumentation and Controls are included in NUREG-0800, Section 7.1, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants."

The applicable regulatory requirements for the information being reviewed in this section are as follows:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36, "Technical specifications"

- 10 CFR 52.79(a)(30), "Contents of applications; technical information"

7.1.4 Technical Evaluation

The staff reviewed WLS COL FSAR Section 7.1 and checked the referenced DCD to ensure that the combination of the DCD information incorporated by reference and the information in the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the WLS COL application and the information incorporated by reference from the AP1000 DCD application addressed the required information relating to setpoint calculations for protective functions. The results of the staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this safety evaluation report (SER) provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the design certification (DC) and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (Vogtle Electric Generating Plant (VEGP), Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews.

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from requests for additional information (RAIs).

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) may include evaluation material from the SER for the Bellefonte Nuclear Plant (BLN), Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 7.1.4:

> *The applicant, in its letter dated May 21, 2010, proposed to incorporate the Setpoint Program (SP) that will be added to the AP1000 DCD into the VEGP Technical Specifications (TS). This proposal was made to address Open Item 6.1-1. In Chapter 16 of this safety evaluation report (SER), the staff concludes that the response to Open Item 16.1-1 is acceptable. The incorporation of this program into the VEGP TS in a later revision was being*

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[1] See Section 1.2.2 for a discussion of the staff's review related to verification of the scope of information to be included in a COL application that references a design certification (DC).

*tracked as **Confirmatory Item 16.1-1**. The closure of this Confirmatory Item is addressed in SER Section 16.1.*

In addition, in a letter dated June 4, 2010, the applicant proposed adding STD COL 7.1-1 as a new COL information item addressed in the VEGP COL FSAR.

<u>AP1000 COL Information Item</u>

- STD COL 7.1-1

The applicant proposed adding a new line item to VEGP COL FSAR Table 1.8-202 to address COL Information Item 7.1-1. The applicant also proposed the following addition to VEGP COL FSAR Section 7.1:

> *7.1.6.1 Setpoint Calculations for Protective Functions*
>
> *The Setpoint Program described in Technical Specifications Section 5.5 provides the appropriate controls for update of the instrumentation setpoints following completion of the calculation of setpoints for protective functions and the reconciliation of the setpoints against the final design.*

The applicant states that the TS program identified in the proposed Section 7.1.6.1 was that addressed in the VEGP revised response to Bellefonte Nuclear Plant (BLN) Open Item 16.1-1, dated May 21, 2010, and that the calculation and reconciliation of the setpoints discussed is required by the AP1000 Inspections, Tests, Analyses and Acceptance Criteria (ITAAC) included in AP1000 DCD Tier 1, Table 2.5.2-8, Item 10. In Chapter 16 of this SER, the staff concludes that the May 21, 2010, response to BLN Open Item 16.1-1 is acceptable.

Based on the ITAAC in Table 2.5.2-8, Item 10 and the TS controls in Section 5.5, the staff finds there are adequate controls for updating the instrumentation and controls (I&C) setpoints. Therefore, the staff finds STD COL 7.1-1 acceptable.

*The incorporation of the changes associated with proposed STD COL 7.1-1 into a future revision of the VEGP COL FSAR is **Confirmatory Item 7.1-1**.*

<u>Resolution of Standard Content Confirmatory Item 7.1-1</u>

Confirmatory Item 7.1-1 is an applicant commitment to revise its WLS COL FSAR Table 1.8-202 and Section 7.1 to address COL Information Item STD COL 7.1-1. The staff verified that the WLS COL FSAR was appropriately revised to address STD COL 7.1-1. As a result, Confirmatory Item 7.1-1 is now closed.

7.1.5 Post Combined License Activities

There are no post COL activities related to this section.

7.1.6 Conclusion

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to setpoint calculations for protective functions, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff compared the application to relevant NRC regulations and other NRC regulatory guides and concludes that, pending closure of the identified confirmatory items, the applicant is in compliance with NRC regulations. The staff based its conclusion on the following:

- To address AP1000 COL Information Item STD COL 7.1-1, the applicant provided a program for setpoint calculations for protective functions in accordance with the requirements of 10 CFR 50.36 and 10 CFR 52.79(a)(30).

7.2 Reactor Trip

WLS COL FSAR, Revision 11, Section 7.2, incorporates by reference, AP1000 DCD, Revision 19, Section 7.2, "Reactor Trip." In addition, in WLS COL FSAR, the applicant provided the following:

Departures

- WLS DEP 6.4-2

The applicant provided additional information in Figure 7.2-202 of the WLS COL FSAR about WLS DEP 6.4-2 related to design changes affecting how the temperature and humidity in the main control room are maintained within the limits for reliable human performance. This information, as well as related WLS DEP 6.4-2 information appearing in other chapters of the FSAR, is reviewed in Section 21.3 of this SER.

- WLS DEP 7.3-1

The applicant provided additional information in Figure 7.2-201 of the WLS COL FSAR about WLS DEP 7.3-1 related to required design changes for the protection and safety monitoring system (PMS) source range neutron flux doubling logic to comply with the requirements of IEEE Std. 603-1991, Clause 6.6. This information, as well as related WLS DEP 7.3-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.5 of this SER.

The NRC staff reviewed Section 7.2 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of the information relating to this section.[1] The NRC staff's review confirmed that the applicant addressed the required information to satisfy the evaluation criteria. There is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

7.3 Engineered Safety Features

WLS COL FSAR, Revision 11, Section 7.3, incorporates by reference, AP1000 DCD, Revision 19, Section 7.3, “Engineered Safety Features.” In addition, in WLS COL FSAR, the applicant provided the following:

Departures

- WLS DEP 6.4-1

The applicant provided additional information in Section 7.3.1.2.17 of the WLS COL FSAR about WLS DEP 6.4-1 related to design changes affecting habitability of the main control room and changes to the calculated doses to control room operators. This information, as well as related WLS DEP 6.4-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.2 of this SER.

- WLS DEP 6.4-2

The applicant provided additional information in Section 7.3.1.2.17 of the WLS COL FSAR about WLS DEP 6.4-2 related to design changes affecting how the temperature and humidity in the main control room are maintained within the limits for reliable human performance. This information, as well as related WLS DEP 6.4-2 information appearing in other chapters of the FSAR, is reviewed in Section 21.3 of this SER.

- WLS DEP 7.3-1

The applicant provided additional information in Section 7.3.1.2.14 of the WLS COL FSAR about WLS DEP 7.3-1 related to required design changes for the PMS source range neutron flux doubling logic to comply with the requirements of IEEE Std. 603-1991, Clause 6.6. This information, as well as related WLS DEP 7.3-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.5 of this SER.

The NRC staff reviewed Section 7.3.1.2 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this section.[1] The NRC staff’s review confirmed that the applicant addressed the required information to satisfy the evaluation criteria. There is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In RAI 01-04, issued to the applicant for the BLN Units 3 and 4, the staff questioned how the applicant would verify that the as-built Instrument & Control (I&C) system configuration conformed to schematics. In its response to RAI 01-04, the BLN applicant indicated that it or a designee would verify I&C cabinets as-built against the design drawings during manufacturing and would functionally test each system. In addition, the BLN applicant’s response indicated that the I&C cabinets would be tested during preoperational testing and in accordance with several ITAAC related to the I&C system. The BLN response to RAI 01-04 was endorsed as standard for WLS by Duke Energy Carolinas, Inc., in its February 5, 2009, letter.

The staff notes that vendor qualification testing, which may be done offsite, and preoperational testing fall under the applicant's quality assurance program. Any anomalies found during the testing or any problems identified from the time the testing is complete until the components are installed at the site would be corrected in accordance with the applicant's quality assurance program.

The staff finds the verification of the as-built I&C system configuration against schematics using a combination of vendor and onsite testing that falls under the applicant's quality assurance program acceptable. In addition, the staff finds that adequate program controls exist to ensure that once the testing was complete, the I&C system configuration would be maintained as valid throughout the life of the plant. Based on the above, the staff finds the response to BLN RAI 01-04 and the WLS endorsement of that response acceptable.

7.4 Systems Required for Safe Shutdown

WLS COL FSAR, Revision 11, Section 7.4 incorporates by reference AP1000 DCD, Revision 19, Section 7.4, "Systems Required for Safe Shutdown," with the following departures:

Departures

- WLS DEP 6.3-1 and WLS DEP 3.2-1

The applicant provided additional information for WLS DEP 6.3-1 and WLS DEP 3.2-1 in WLS COL FSAR Section 7.4.1.1 related to extended operation of the PRHR-HX, the ability to maintain safe shutdown conditions, changing the indefinite duration to at least 72 hours, and operator directed actions to preserve battery capability. This information, as well as related WLS DEP 6.3-1 information appearing in other chapters of the WLS COL FSAR, is reviewed in Section 21.1 of this SER.

The staff reviewed WLS COL FSAR Section 7.4.1.1 and checked the referenced DCD to ensure that the combination of the DCD and the WLS COL application represents the complete scope of information relating to this section. The staff's review confirmed that the applicant adequately addressed the required information to satisfy the evaluation criteria. There is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements. Section 21.1 of this report evaluates the departures from the AP1000 DCD provided in WLS DEP 6.3-1 and WLS DEP 3.2-1.

7.5 Safety-Related Display Information (Related to RG 1.206, Section C.III.1, Chapter 7, C.I.7.5, "Information Systems Important to Safety")

7.5.1 Introduction

Safety-related display information includes equipment that processes safety-related information and displays it for use by the operator to monitor and maintain the safety of the AP1000 throughout operating conditions that include anticipated operational occurrences and accident and post-accident conditions.

The AP1000 DCD contains COL Information Item 7.5-1 that requires the COL applicant to address post accident monitoring variables listed as site-specific in DCD Tables 7.5-1 and 7.5-8.

7.5.2 Summary of Application

WLS COL FSAR, Revision 11, Section 7.5, incorporates by reference AP1000 DCD, Revision 19, Section 7.5 with departures and/or supplements.

To address the departures and/or supplements, the applicant provided the following additional information:

Departure

- WLS DEP 6.4-2

The applicant provided additional information in Section 7.5 of the WLS COL FSAR about WLS DEP 6.4-2 related to design changes affecting how the temperature and humidity in the main control room are maintained within the limits for reliable human performance. This information, as well as related WLS DEP 6.4-2 information appearing in other chapters of the FSAR, is reviewed in Section 21.3 of this SER.

AP1000 Information Items

- WLS COL 7.5-1 and STD COL 7.5-1

The applicant provided additional information in WLS COL FSAR Section 7.5, "Safety-Related Display Information," describing the WLS COL FSAR Table 7.5-201 supplement (SUP) to AP1000 DCD Table 7.5-1 and AP1000 DCD Table 7.5-8 providing variable data shown in the DCD tables as "site specific."

7.5.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the information systems important to safety are given in NUREG-0800, Section 7.5.

The applicable regulatory requirements, guidelines, and related acceptance criteria for the supplemental information item are as follows:

- General Design Criterion (GDC) 13, "Instrumentation and Control"
- GDC 64, "Monitoring Radioactivity Releases"

The regulatory bases require, in part, 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. Monitoring should include checking the plant environs for radioactivity that may be released from postulated accidents.

7.5.4 Technical Evaluation

The staff reviewed WLS COL FSAR Section 7.5 and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the COL application and the DCD information incorporated by reference addressed the required information relating to safety-related display information. The results of the staff's evaluation of the DCD information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the WLS COL FSAR:

AP1000 Information Items

- WLS COL 7.5-1 and STD COL 7.5-1

The AP1000 DCD references and commits to Regulatory Guide (RG) 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Revision 3, as the method of complying with GDC 13 and GDC 64.

WLS COL FSAR, Appendix 1AA, Revision 3, takes exception to RG 1.97, Revision 4. Instead, the applicant stated conformance to RG 1.97, Revision 3. The applicant stated, "Portable equipment outside the DCD scope conforms to Revision 3 of this Regulatory Guide for consistency with DCD scope since Revision 4 indicates that partial implementation is not advised." The acceptability of RG 1.97, Revision 3, is discussed by the staff in Section 12.1 of this report.

RG 1.97, Revision 3, states that the variable and range information should be provided for environs radiation and radioactivity, and meteorological instrumentation.

The staff issued RAI 4, Question 07.05-1, to the applicant requesting information on boundary environs radiation and meteorological instrumentation. The staff finds that the range of the boundary environs radiation instruments is necessary to ensure that the instruments are adequate for monitoring radioactivity that may be released from a postulated accident. The meteorological range and accuracy information conforms to the guidance of RG 1.97, Revision 3 with the exception of the differential temperature range. In its October 1, 2009, response to RAI 4, Question 07.05-1, the applicant proposed a range of -4°C to 8°C instead of - 5°C to 10°C and provided the following justification:

> *Stability class is estimated from the vertical temperature gradient between the 60m and 10m levels on the meteorological tower. As noted in footnote (f) of revised FSAR Table 2.3-281 transmitted as an enclosure to Reference 1, Delta-T is calculated by the datalogger. The datalogger Delta-T reading is verified to be within +0.05 degrees when the temperature sensors are placed in a drywell isothermal environment during calibration. A stated range of -4°C to +8°C is referenced based on procedural tolerances and ranges used at Duke Energy's operating nuclear stations. This range is adequate for Delta-T in determining the stability class per Table 1 of Regulatory Guide 1.23, Revision 1, with Delta-T criteria covering an overall range of (dT **<** -1.9°C) to (dT **>** +4.0°C) per a 100m separation. For the Lee 60m meteorological tower, with a 50m separation between the upper and lower temperature measurement levels, this equates to*

Delta-T criteria for estimating stability class within the overall range of (dT **<** *-0.95°C) to (Dt* **>** *+2.0°C).*

The staff finds the applicant's reduced differential temperature range acceptable because the data logger reading is verified to be within 0.05 degrees, which meets the 0.1 degrees resolution requirements of RG 1.23 (Revision 1), and the Delta-T calibration range (-4°C to +8°C) bounds RG 1.23 (Revision 1) Ambient Temperature Change with Height criteria as identified in RG 1.23, Revision 1, Table 1(AS1), "Classification of Atmospheric Stability." RG 1.23 provides guidance on meteorological instrumentation, which is referenced by RG 1.97.

The staff notes that the supplemental information conforms to the guidance of RG 1.97, Revision 3. The staff confirmed the incorporation of the instrumentation supplemental information in the WLS COL FSAR. The staff finds the supplemental response acceptable and considers RAI 4, Question 07.05-1 resolved.

In a May 26, 2010, letter, Westinghouse proposed a change to the AP1000 DCD to add COL Information Item 7.5-1 requiring that COL applicants provide information for variables listed as "site specific" in DCD Tables 7.5-1 and 7.5-8. Although this information was provided for WLS as part of WLS SUP 7.5-1 and incorporated in the WLS COL FSAR, the identification of COL Information Item 7.5-1 in theAP1000 DCD required that the applicant address this information with a COL identifier rather than as supplemental information. Accordingly, the applicant's November 4, 2010, letter proposes to replace WLS SUP 7.5-1 with STD COL 7.5-1 (for standard information) and WLS COL 7.5-1 (for WLS specific information). This change of identifiers does not impact the staff's conclusion regarding the instrumentation information added to the WLS COL FSAR. The incorporation of the changed identifiers into the WLS COL FSAR was treated as **Confirmatory Item 7.5-1**.

<u>Resolution of Standard Content Confirmatory Item 7.5-1</u>

Confirmatory Item 7.5-1 is an applicant commitment to revise its WLS COL FSAR Table 1.8-202, and Sections 7.5.2, 7.5.3.5, and 7.5.5 to address COL Information Item STD COL 7.5-1. The staff verified that the WLS COL FSAR was appropriately revised. Accordingly, the staff considers Confirmatory Item 7.5-1 closed.

7.5.5 Post Combined License Activities

There are no post COL activities related to this section.

7.5.6 Conclusion

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant adequately addressed the required information relating to safety-related display information, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the DCD information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff compared the application to the relevant NRC regulations and other NRC regulatory guides and concludes that, the applicant is in compliance with NRC regulations. The applicant has satisfactorily addressed the guidance of Revision 3 of RG 1.97 through the response to RAI 4, Question 07.05-1. The staff based its conclusion on the following:

- WLS DEP 6.4-2, related to design changes affecting how the temperature and humidity in the main control room are maintained within the limits for reliable human performance, is reviewed and found acceptable by the staff in Section 21.3 of this SER.
- To address WLS COL 7.5-1 and STD COL 7.5-1, the applicant provided sufficient information regarding the safety-related display information, which is acceptable in accordance with the requirements of 10 CFR Part 50, Appendix A, GDC 13 and GDC 64.

7.6 Interlock Systems Important to Safety

WLS COL FSAR Section 7.6, Revision 11, incorporates by reference, with no departures or supplements, AP1000 DCD, Section 7.6, “Interlock Systems Important to Safety,” Revision 19. The staff reviewed the WLS COL application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The staff’s review confirmed that there is no outstanding issue related to this section. The results of the staff’s technical evaluation of the DCD information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

7.7 Control and Instrumentation Systems (Related to RG 1.206, Section C.III.1, Chapter 7, C.I.7.7, “Control Systems Not Required for Safety”)

WLS COL FSAR, Revision 11, Section 7.7, incorporates by reference, with no departures or supplements, AP1000 DCD, Revision 19, Section 7.7, “Control and Instrumentation Systems,” The staff reviewed the WLS COL application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The staff’s review confirmed that there is no outstanding issue related to this section. The results of the staff’s technical evaluation of the DCD information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

8 ELECTRIC POWER

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 during abnormal and accident conditions. This chapter provides information on the functional adequacy of the offsite power systems and safety-related onsite electric power systems, as applicable to the AP1000 passive design, and ensures that these systems have adequate capacity, capability, redundancy, independence, and testability in conformance with the current criteria established by the U.S. Nuclear Regulatory Commission (NRC).

Chapter 8, “Electric Power,” of this safety evaluation report (SER) describes the results of the review by the staff (the staff) of the William States Lee III Nuclear Station (WLS) combined license (COL) final safety analysis report (FSAR), Part 2 of the COL application, submitted by Duke Energy (DE), the COL applicant (the applicant).

8.1 Introduction

8.1.1 Introduction

This section provides the applicant’s description of the electric power system with regard to the interrelationships between the nuclear unit, the utility grid, and the interconnecting grids.

In addition, this section includes a regulatory requirements applicability matrix that lists all design bases, criteria, regulatory guides (RGs), standards, and other documents to be implemented in the design of the electrical systems that are beyond the scope of the design certification (DC).

8.1.2 Summary of Application

WLS COL FSAR, Revision 11, Section 8.1, incorporates by reference AP1000 design control document (DCD), Revision 19, Section 8.1 with departures and/or supplements.

To address these departures and supplements, in WLS COL FSAR, Section 8.1, the applicant provided the following additional information:

Supplemental Information

- WLS SUP 8.1-1

The applicant provided supplemental (SUP) information in WLS COL FSAR Section 8.1, “Introduction,” describing DE’s 525 and 230-kilovolt (kV) transmission systems, and the connection interface with the WLS Unit 1 to the 230-kV switchyard and WLS Unit 2 to the 525-kV switchyard at the plant site.

- WLS SUP 8.1-2

The applicant provided supplemental information in WLS COL FSAR Section 8.1 describing additional information pertaining to regulatory guides and Institute of Electrical and Electronics Engineers (IEEE) standards identified in AP1000 DCD, Table 8.1-1 and to other applicable regulatory guides as indicated in WLS COL FSAR Table 8.1-201, "Site-Specific Guidelines for Electric Power Systems."

8.1.3 Regulatory Basis

The regulatory basis for the information incorporated by reference is addressed in NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of NRC regulations for the introduction to the electric power systems are given in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," Section 8.1.

The applicable regulatory requirements, guidelines, and related acceptance criteria for the supplemental information items are as follows:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.63, "Loss of all alternating current power"
- RG 1.155, "Station Blackout"
- RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)"

8.1.4 Technical Evaluation

The staff reviewed WLS COL FSAR Section 8.1 and checked the referenced DCD to ensure that the combination of the DCD and the application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the introduction to the electric power systems. The results of the staff's evaluation of the information incorporated by reference in the WLS application are documented in NUREG-1793 and its supplements.

The staff reviewed the following information in the WLS COL FSAR:

Supplemental Information

- WLS SUP 8.1-1

The staff reviewed the resolution to the supplemental information WLS SUP 8.1-1 related to the DE transmission system and its connection to the WLS Units 1 and 2 included in WLS COL FSAR Section 8.1. The applicant provided the following supplements to WLS COL FSAR Section 8.1.1.

The DE transmission system consists of interconnected hydro plants, fossil-fueled plants, combustion turbine units and nuclear plants supplying energy to the service area at various

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[1] See Section 1.2.2 of this report for a discussion of the staff's review related to verification of the scope of information to be included in a COL application that references a DC.

voltages up to 525 kV. The 525-kV switchyard is tied to DE's 525-kV transmission network by two single-circuit overhead lines. The 230-kV switchyard is tied to the DE's 230-kV transmission network by two double-circuit overhead lines. Both switchyards utilize breaker-and-a-half bus configurations. The two switchyards are connected by two 230-kV to 525-kV autotransformers.

The staff finds that the applicant has adequately described the WLS Units 1 and 2, connection to the utility grid and the information provided is in accordance with the recommendation of RG 1.206 and the guidance in NUREG-0800, Section 8.1.

- WLS SUP 8.1-2

The staff reviewed supplemental information in SUP 8.1-2, related to regulatory guidelines and industry standards and found it to be consistent with NUREG-0800, Section 8.1 with the exception of the information discussed below.

WLS COL FSAR Table 8.1-201, Item (1b) indicates that RG 1.155 is not applicable to WLS. This item was deemed as standard among COL applications being discussed in Bellefonte Nuclear Station's (BLN) response to request for additional information (RAI) 08.01-2. In a February 5, 2009, letter, the WLS applicant stated that the standard response to RAI 08.01-2 also applies to the WLS application.

The standard response submitted by BLN in a June 24, 2008, letter is summarized as follows. BLN stated that the AP1000 design meets the requirements of 10 CFR 50.63 for 72 hours and, therefore, no specific procedures or training specific to station blackout (SBO) are necessary. The staff determined that the above response was inconsistent with the recommendations of RG 1.155 and the requirements of 10 CFR 50.63. The staff recognizes that the passive systems can maintain safe-shutdown conditions after design-basis events for 72 hours, without operator action, following a loss of both onsite and offsite alternating current (ac) power sources. However, the applicant needs to establish SBO procedures and training for operators to include actions necessary to restore offsite power after 72 hours by addressing ac power restoration (e.g., coordination with transmission system load dispatcher), and severe weather guidance (e.g., identification of site-specific actions to prepare for the onset of severe weather such as an impending tornado) in accordance with RG 1.155, Regulatory Positions C.2 and C.3.4.

Several discussions were held between the staff and the BLN applicant regarding this issue. Subsequently, in an April 15, 2009, letter, the BLN applicant stated that the training and procedures to support mitigation of an SBO event would be implemented in accordance with BLN FSAR Sections 13.2 and 13.5, respectively. As recommended by NUMARC 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," which is endorsed by RG 1.155, the loss of all ac power event mitigation procedures will address response (e.g., restoration of onsite power sources), ac power restoration (e.g., coordination with transmission system load dispatcher), and severe weather guidance (e.g., identification of actions to prepare for the onset of severe weather such as an impending tornado), as applicable. In addition, the BLN applicant stated that there are no nearby large power sources, such as a gas turbine or black start fossil fuel plant that can directly connect to the station to mitigate the event.

The staff verified that the WLS applicant has updated WLS COL FSAR Sections 1.9.5.1.5 and 1.9.6 to include the above-mentioned items including the implementation of training and procedures to support mitigation of an SBO event. The staff finds this update satisfies RG 1.155, Regulatory Positions C.2 and C.3.4. Based on the above, the staff finds this item resolved.

8.1.5 Post Combined License Activities

There are no post COL activities related to this section.

8.1.6 Conclusion

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to the introduction to the electric power systems, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS application are documented in NUREG-1793 and its supplements.

In addition, the staff compared the additional COL-specific supplemental information in the application to the relevant NRC regulations; guidance in NUREG-0800, Section 8.1, and other NRC regulatory guides and concluded that the applicant is in compliance with NRC regulations. The staff based its conclusion on the following:

- WLS SUP 8.1-1 is acceptable because the applicant provided sufficient information regarding the WLS connections to various transmission systems in accordance with the recommendations of RG 1.206.

- WLS SUP 8.1-2 is acceptable because the COL-specific regulatory guidelines and industry standards, and additional new regulatory guidelines are adequately addressed by the applicant. The applicant has also provided sufficient information for satisfying the requirements of 10 CFR 50.63 and the guidance in RG 1.155

8.2 Offsite Power System

8.2.1 Introduction

The offsite power system is referred to in regulatory guides and industry standards as the "preferred power system." It includes two or more physically independent circuits capable of operating independently of the onsite standby power sources and encompasses the grid, transmission lines (overhead or underground), transmission line towers, transformers and other switchyard components.

The AP1000 passive reactor plant standard design supports an exemption in 10 CFR Part 52, "Licenses, certifications, and approvals for nuclear power plants," Appendix D, "Design Certification Rule for the AP1000 Design," paragraph V.B.3, to the requirement of 10 CFR Part 50, "Domestic licensing of production and utilization facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 17, "Electric Power Systems," to have only one (not two) physically independent offsite circuit to provide for safety-related passive systems for core cooling and containment integrity.

Therefore, for WLS Units 1 and 2, the single offsite power source provided from the transmission network is reviewed below to assure that it satisfies the requirements of GDC 17 with respect to its capacity and capability.

8.2.2 Summary of Application

WLS COL FSAR, Revision 11, Section 8.2, incorporates by reference AP1000 DCD, Revision 19, Section 8.2, with departures and supplements.

To address these departures and/or supplements, in WLS COL FSAR Section 8.2 the applicant provided the following additional information:

AP1000 COL Information Items

- WLS COL 8.2-1

The applicant provided additional information in WLS COL 8.2-1 to address COL Information Item 8.2-1 (COL Action Items 8.2.3-1 and 8.2.3.3-1) to address the design of the ac power transmission system and its testing and inspection plan. The information describes: (1) the designs of the plant site 525-kV and 230-kV switchyards, the two 525-kV transmission lines connecting the plant switchyard to DE’s 525-kV transmission system, and the four 230-kV lines connecting the 230-kV switchyard, the connection of two switchyards through autotransformers, and the interface of the switchyards with the transmission grid; (2) the connections of the generator step-up (GSU) transformers and the reserve auxiliary transformers (RATs) to the switchyard; (3) the designs of the switchyard circuit breakers and disconnect switches; (4) the transformer area arrangement for each unit; (5) the ratings of the GSU transformers, unit auxiliary transformers (UATs), RATs, and autotransformers; 6) the design of the control building in the plant site 525-kV and 230-kV switchyards; (7) the administrative control of 525-kV and 230-kV switchyards and transmission lines circuit breakers; and (8) the switchyard and transmission lines testing and inspection plan; and 9) voltage operating range, frequency decay rate, and preservation of grid connection. WLS COL 8.2-1 is addressed in WLS COL FSAR Sections 8.2.1, 8.2.1.1, 8.2.1.2, 8.2.1.3, and 8.2.1.4.

In addition, the WLS applicant provided supplemental information describing details of a failure mode and effects analysis (FMEA) performed for the offsite power distribution system and the plant site switchyard.

- WLS COL 8.2-2

The applicant provided additional information in COL 8.2-2 to address COL Information Item 8.2-2 (COL Action Items 8.2.3.1-1, 8.2.3.1-2, and 8.2.3.1-3) describing: (1) the 525-kV and 230-kV switchyards arrangement and design of the protective relaying scheme; and (2) a transmission system study performed regularly to verify grid stability, switchyard voltage, and frequency to confirm the transmission system capability to maintain reactor coolant pump (RCP) operation for 3 seconds following a turbine trip as specified in AP1000 DCD Section 8.2.2. WLS COL 8.2.2 is addressed in WLS COL FSAR Sections 8.2.1.2.2, and 8.2.2.

Site-Specific Information Replacing Conceptual Design Information

- WLS CDI

The applicant provided COL-specific conceptual design information (CDI) describing the transformer area located next to each unit's turbine building and containing the GSU transformer, the UATs, and the RATs. This replaced the CDI located in the AP1000 DCD.

Supplemental Information

- WLS SUP 8.2-1

The applicant provided supplemental information on the transmission system provider/transmission system operator (TSP/TSO), and the detailed voltage and other requirements to be maintained by the TSP/TSO.

- WLS SUP 8.2-2

The applicant provided supplemental information describing the formal agreement between DE's Nuclear Generation Department (NGD) and DE's Power Delivery (PD) department, which is the TSO, setting the requirements for transmission system studies and analyses.

- WLS SUP 8.2-3

The applicant provided supplemental information describing the establishment of the PD department's responsibility for maintaining area bulk transmission system reliability and demonstrating, by power system simulation studies, projections, and analyses, the current and future reliability of the system.

- WLS SUP 8.2-4

The applicant provided supplemental information describing the agreement between the NGD and the PD departments demonstrating that protocols are in place for WLS to remain cognizant of grid vulnerabilities in order to make informed decisions regarding maintenance activities critical to the electric system.

- WLS SUP 8.2-5

The applicant provided supplemental information describing the reliability of the DE transmission system's 525-kV and 230-kV transmission lines based on 12 years of outage data.

Interface Requirements

The plant/offsite electrical power interfaces for the AP1000 standard design are discussed in AP1000 DCD Tier 2, Section 8.2.5. AP1000 DCD Tier 2, Table 1.8-1, Items 8.1, 8.2, and 8.3 identify these interfaces as non-nuclear safety (NNS) interfaces.

Inspections, Tests, Analyses and Acceptance Criteria

In a letter dated August 28, 2014, the applicant provided a supplemental response to RAI 108, Question 08-1 that proposed to revise COL application Part 10, Appendix B, to include two new inspections, tests, analyses and acceptance criteria (ITAAC), numbered 4.g and 7, in order to address Bulletin 2012-01, "Design Vulnerability in Electric Power System."

8.2.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of NRC regulations for the offsite power system are given in NUREG-0800, Sections 8.1 and 8.2.

The regulatory bases for acceptance of the COL information and supplementary information items are established as follows:

- 10 CFR Part 50, “Domestic licensing of production and utilization facilities,” Appendix A, “General Design Criteria [GDC] for Nuclear Power Plants,” Criterion 17, “Electric power systems”
- GDC 5, “Sharing of structures, systems, and components”
- GDC 18, “Inspection and testing of electrical power systems”
- 10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants”
- RG 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition)”
- Generic Letter (GL) 2006-02, “Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power”

8.2.4 Technical Evaluation

The staff reviewed WLS COL FSAR Section 8.2 and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The staff’s review confirmed that the information in the COL application and the DCD information incorporated by reference addresses the required information relating to the offsite power system. The results of the staff’s evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff’s findings on standard content that were documented in the SER for the reference COL application (Vogtle Electric Generating Plant (VEGP) Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.
- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed by the WLS COL applicant.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and finds the evaluation performed for the standard content to be directly applicable to the WLS application. This standard content material is identified in this report by use of italicized, double-indented formatting. Any confirmatory items in the standard content material retain the numbers assigned in the VEGP SER. Confirmatory items that are first identified in this report section have a WLS designation (e.g., Confirmatory Item WLS 8.2-1).

The staff reviewed the information in the WLS COL FSAR:

AP1000 COL Information Item

- WLS COL 8.2-1

The applicant provided additional information in WLS COL 8.2-1 to resolve COL Information Item 8.2-1, which states:

> Combined License applicants referencing the AP1000 certified design will address the design of the ac power transmission system and its testing and inspection plan (DCD Section 8.2.5).

The commitment was also captured as COL Action Items 8.2.3-1 and 8.2.3.3-1 in NUREG-1793, Appendix F, which states:

> The operating voltage for the high side of the AP1000 transformer and transmission switchyard, as well as the frequency decay rate are site specific and, therefore, will be addressed in the COL application. The COL applicant will provide analysis of these matters, including transient stability, voltage operating range, and preservation of the grid connections, in the COL application (COL Action Item 8.2.3-1).
>
> Combined License applicants referencing the AP1000 certified design will address the design of the ac power transmission system and its testing and inspection plan (COL Action Item 8.2.3.3-1).

The staff reviewed the resolution to the WLS COL 8.2-1 related to the transmission system design, testing, and inspection included under WLS COL FSAR Section 8.2. The staff's evaluation is as follows:

WLS Units 1 and 2, receive offsite ac power from a common 525/230-kV switchyard, which is connected to the DE transmission network. The two switchyards are connected by two 230-kV to 525-kV autotransformers. Unit 2 is connected to the 525-kV switchyard, and Unit 1 is connected to the 230-kV switchyard. The power from Unit 2 is transmitted via overhead transmission line to the 525-kV switchyard. Similarly, the power from Unit 1 is transmitted via overhead transmission lines to the 230-kV switchyard.

There are four transmission lines connected to the 230-kV switchyard, and two transmission lines connected to the 525-kV switchyard. Each transmission line is tied into a DE transmission line or switchyard located between 19 and 95 miles from the station.

Unit 1 is connected to the DE Transmission System via the Roddey East and Roddey West 230-kV lines. The Roddey lines consist of a section of line, 55 km (34 miles (mi)) in length, from WLS to Catawba Nuclear Station and a section of line, 31 km (19 mi) in length, from WLS to Pacolet Tie. The 230-kV line is constructed on a 46 m (150 foot (ft)) wide right-of-way with double circuit lattice steel towers; varying in height from 37 m to 58 m(120 ft to 190 ft) with a nominal height of 46 m (150 ft). Unit 2 is connected to the DE Transmission System via the Asbury 525-kV line. This line consists of a section of line, 66 km (41 mi) in length, from WLS to Newport Tie and a section of line, 153 km (95 mi) in length, from WLS to Oconee Nuclear Station. The 525-kV line is constructed on a 61 m (200 ft) wide right-of-way with single circuit lattice steel towers, varying in height from 37 m to 46 m (120 ft to 150 ft) with a nominal height of 43 m (140 ft). Conductors are two per phase in a horizontal bundle. All lines are designed to meet or exceed the requirements of the American National Standards Institute (ANSI) C2, "National Electric Safety Code."

The staff reviewed the layout of transmission lines and concluded that at least one offsite power source will be available to both Units 1and 2. The staff finds that the above satisfies the requirements of GDC 17.

In WLS COL FSAR Section 8.2.1.4, the applicant addressed the switchyard and transmission lines testing and inspection plan. In RAI 5, Question 08.02-7, the staff questioned the statement made in WLS COL FSAR Section 8.2.1.4, which states, "PD follows its own field test manuals, vendor manuals and drawings, industry's maintenance practices and observes Federal Energy Regulatory Commission (FERC) requirements and North American Electric Reliability Corporation (NERC) reliability standards." The staff requested the applicant explain whether the statement is intended to indicate that DE NGD will follow the FERC and NERC standards for switchyard maintenance and testing. In a September 5, 2008, response, the applicant stated that this statement was intended to indicate that DE follows the applicable NERC reliability standards associated with switchyard maintenance and testing. The applicant stated that it will revise the WLS COL FSAR as follows, for purposes of clarity:

> For performance of maintenance, testing, calibration, and inspection, TSO follows its own field test manuals, vendor manuals and drawings, and industry's maintenance practices to comply with applicable NERC reliability standards.

The staff has verified that the WLS COL FSAR has been updated to include the above-mentioned revised paragraph. Since the goal of GDC 18 is to assure testability of, in this case, the switchyard and transmission lines and the goal of GDC 17 is, among other things, to assure a high reliability of the offsite power system, full conformance to the applicable NERC reliability standards acceptably addresses the staff's concern. Therefore, the staff finds the issue in RAI 5, Question 08.02-7 resolved.

In COL 8.2-1 (WLS COL FSAR Section 8.2.1.1), the applicant also provided a brief summary of the FMEA performed on the WLS switchyard. In RAI 1, Question 08.02-2, the staff requested that the applicant describe in detail how each event (a breaker not operating during a fault on an offsite line; fault on a switchyard bus; fault on an auto bank; a spurious relay trip; a loss of control power; and other cases discussed in the WLS COL FSAR). In a September 5, 2008, response, the applicant provided the requested information and stated that the WLS COL FSAR will be revised to include the detailed FMEA.

The staff verified that WLS COL FSAR Section 8.2.1.1 has been updated to include the detailed FMEA results. The staff reviewed the FMEA of the WLS switchyard and concludes that a single initiating event, such as (1) a breaker not operating during a fault condition; (2) a fault on a switchyard bus; (3) a spurious relay trip; (4) or a loss of control power supply would not cause failure of more than one single offsite line, or a loss of offsite power to either WLS Unit 1 or 2. Therefore, the staff finds that the issue in RAI 1, Question 08.02-2 has been adequately addressed and resolved.

Additionally, the applicant provided the site-specific voltage and frequency variations expected at the WLS Units 1 and 2, switchyard during transient and steady state operating conditions and the site-specific frequency decay rate to satisfy COL 8.2-1.

- WLS COL 8.2-2

The applicant provided additional information in WLS COL 8.2-2 to resolve COL Information Item 8.2-2, which states:

> The Combined License applicant will address the technical interfaces for this nonsafety-related system listed in Table 1.8-1 and Section 8.2.2. These technical interfaces include those for ac power requirements from offsite and the analysis of the offsite transmission system and the setting of protective devices.

The staff's evaluation of the technical interfaces is addressed under "Interface Requirements" in this section of the report.

The commitment was also captured as COL Action Items 8.2.3.1-1, 8.2.3.1-2, and 8.2.3.1-3 in NUREG-1793, Appendix F, which states:

> The COL applicant will perform a site-specific grid stability analysis to show that, with no electrical system failures, the grid will remain stable and the reactor coolant pump bus voltage will remain above the voltage necessary to maintain the flow assumed in the Chapter 15 analyses for a minimum of 3 seconds following a turbine trip (COL Action Items 8.2.3.1-1 and 8.2.3.1-3).
>
> The COL applicant will set the protective devices controlling the switchyard breakers in such a way as to preserve the grid connection following a turbine trip (COL Action Item 8.2.3.1-2).

The staff reviewed the resolution to the COL Information Item WLS COL 8.2-2, related to the transmission system stability analysis and switchyard circuit breaker protective device settings included under WLS COL FSAR Section 8.2. The staff's evaluation is as follows:

WLS COL 8.2-2 was provided by the applicant describing details of: (1) the 525-kV and 230-kV switchyard arrangements and protective relaying schemes; and (2) a transmission system study performed regularly to verify grid stability, switchyard voltage, and frequency to maintain RCP operation for three seconds following a turbine trip as specified in AP1000 DCD Section 8.2.2. WLS COL 8.2-2 is addressed in WLS COL FSAR Sections 8.2.1.2.2 and 8.2.2.

The WLS 525-kV and 230-kV switchyards each have two main buses for each voltage level. All of the 525-kV and 230-kV lines and each of the GSU transformers are connected to both buses. The switchyards have breaker-and-a-half scheme. This arrangement is used for reliability and flexibility. This arrangement allows for isolation of components and buses, while preserving the plant’s connection to the grid.

The transmission line relay protection circuits continuously monitor the conditions of the offsite power system and are designed to detect and isolate the faults with maximum speed and minimum disturbance to the system. Each of the 525-kV and 230-kV lines is protected by two independent pilot systems to clear a fault anywhere on the line. The two autotransformers each have primary and secondary protecting relaying. The primary and secondary relaying use separate instrument current transformers for monitoring, and use separate direct current (dc) power supplies.

In the event of a breaker failure, the breaker failure relays operate after a preset time delay. Should a breaker fail to trip within the time setting, the associated breaker failure trip relay will trip and lock out all breakers necessary to isolate the failed breaker from all local sources. A breaker failure relay operation for 230-kV and or 525-kV switchyard breakers that are connected to a GSU, RAT, and auto bank transformers will also isolate the appropriate remote sources through a direct transfer trip operation.

The staff finds that the switchyard breaker arrangement, the protection of lines by independent high speed relay schemes, and the breaker failure scheme would preserve the WLS connection to the grid following a turbine trip. The staff finds this satisfies COL Action Item 8.2.3.1-2.

With regard to grid stability, the applicant stated that the WLS grid stability analysis confirms that, with no electrical system failures, the grid will remain stable and the RCP bus voltage will remain above the voltage necessary to maintain the flow assumed in the Chapter 15 analyses for a minimum of 3 seconds following a turbine trip as specified in AP1000 DCD Section 8.2.2 (COL Action Item 8.2.3.1-3). This requirement is also met when there is transmission element out of service, including the largest generator or most critical transmission line. Also, the grid stability analysis has confirmed that the interface requirements for steady state load, inrush kilovolt amp (kVA) for motors, nominal voltage, allowable voltage regulation, nominal frequency, allowable frequency fluctuation, maximum frequency decay rate and limiting under-frequency value for RCP have been met. Based upon the staff’s review of this information, the staff finds that the design will maintain acceptable voltage and frequency at the RCP buses for a minimum of 3 seconds in accordance with the Chapter 15 safety analyses, and is therefore acceptable.

However, based upon the wording in WLS COL FSAR Section 8.2, the staff was concerned that at WLS, the grid voltage could drop up to 20 percent on the high side of the GSU and RATs. This voltage drop could damage auxiliary and safety-related equipment. Therefore, in RAI 1, Question 08.02-1, the staff requested that the applicant clarify: (a) if the 20 percent voltage drop was based on worst expected switchyard voltage; (b) if the 20 percent voltage drop criteria is consistent with NERC criteria (or those of a local reliability council); and (c) what effect this voltage drop will have on the operation of the onsite auxiliary power system equipment including the Class 1E battery chargers and uninterruptible power supplies (UPSs). Subsequent clarification of this issue was provided by the applicant as discussed in the following three paragraphs.

In a September 5, 2008, letter supplemented by a September 14, 2011, letter, the applicant stated that in the WLS grid stability evaluation, switchyard equipment, including the transformers, were modeled to confirm required voltage would be available at the generator bus or high side of the transformer being used for bus supply. In addition, the applicant stated that the TSP/TSO maintains switchyard voltage such that steady state voltage on the 26-kV isophase bus is within 0.95-1.05 per unit (pu) of its nominal value. Based on the analysis, the expected voltage at the generator terminals is 1.00 pu for Unit 1 and 1.01 pu for Unit 2. In addition, the applicant stated that there were several different pre-contingency cases created with various generation outages that established initial conditions; then contingencies were applied. The base case, with all lines in service, and the combinations of generation outages and grid contingencies simulate different grid configurations that create several different pre-trip steady state voltages. Therefore, a series of different pre-trip voltages were studied. Steady state studies showed that, for Unit 2, a turbine trip with an Asbury West 525-kV line outage caused a 4.35-kV decrease on the 525-kV bus and a 1.18-kV decrease on the 230-kV bus, both less than a 1 percent change. For Unit 1, an outage of the 525/230-kV autotransformer caused a 3.29-kV decrease on the 230-kV bus and a 1.93-kV decrease on the 525-kV bus, both less than a 2 percent change. The voltage changes from the worst case contingency on each unit satisfy the voltage requirement for the RCP.

With regard to item (b) above, the applicant stated that the DE Bulk Electric System is designed to meet NERC reliability standards. The NERC standards do not give specific voltage or voltage drop criteria, but require that the system remain stable and consistent with the voltage requirements of the control area. However, maintaining switchyard voltage such that steady state voltage on the 26-kV isophase bus is within 0.95-1.05 pu of its nominal value would be considered to be consistent with the NERC requirement for system stability. Additionally, the criterion that the voltage cannot drop below a level that provides less than 80 percent of the nominal voltage at the RCP is consistent with DE practices to supply sufficient voltage at the nuclear switchyards or notify the plant operator when the minimum voltage may not be available.

With regard to item (c) above, the applicant clarified that the 80 percent voltage level discussed in the WLS COL FSAR referred to the equipment ratings and was not part of their description of the stability study results. The steady state studies showed changes of less than 2 percent and 1 percent for WLS Units 1 and 2, respectively, based on contingencies considered to be sufficiently extensive and at an appropriate severity level to bound the reasonably expected voltages. Based upon the above clarification and detailed response of the applicant to RAI 1, Question 08.02-1, the staff has no further concerns on this item and considers RAI 1, Question 08.02-1 resolved.

In reference to the Class 1E battery chargers, the applicant stated that the battery chargers are a qualified Class 1E isolation device. The battery charger function is to provide isolation between input ac and the safety-related dc system and to provide dc source power when ac power is available. Safe shutdown of the plant does not require the support of the battery chargers. The battery charger is designed to allow the battery to support the dc loads during times of ac input undervoltage. This could occur during the 3-second turbine trip transient discussed above, during which the RCP must remain above 80-percent stall voltage. The battery charger supply breaker at the ac motor control center is not designed to trip on this undervoltage condition. Additionally, there is no design requirement in the AP1000 to lock out the battery charger on an ac input undervoltage condition.

The staff reviewed the above information and concludes that this information is sufficient to demonstrate that the grid will remain stable to maintain RCP operation for 3 seconds following a turbine trip. The staff finds that the applicant has satisfied the portion of COL Information

Item 8.2-2 to maintain the voltage at the RCP to ≥ 80 percent for at least 3 seconds following a turbine trip, to maintain the reactor coolant flow assumed in the Chapter 15 analyses.

The staff finds the applicant's response acceptable because the analysis meets the AP1000 design requirements, the requirements of GDC 17 and the guidelines of RG 1.206. Therefore, the staff considers the issues in RAI 1, Question 08.02-1 resolved.

In RAI 5, Question 08.02-4, the staff stated that to confirm that the single offsite power circuit provided from the transmission network satisfies the requirements of GDC 17, the applicant should provide the voltage and frequency variations expected at the 525-kV and 230-kV switchyards and confirm that these voltage and frequency limits are acceptable for auxiliary power system equipment operation during different operating conditions.

Confirmation that these voltage and frequency limits are acceptable was shown by the following calculations: load flow analysis (bus and load terminal voltages of the station auxiliary system); short circuit analysis; equipment sizing studies; protective relay setting and coordination; motor starting with minimum and maximum grid voltage conditions. A separate set of calculations also was performed for each available connection to the offsite power supply. In addition, the applicant provided a discussion of how the results of the calculations will be verified before fuel load.

In a September 26, 2008, response to RAI 5, Question 08.02-4, the applicant stated that there is no requirement for functionality of the offsite power to accomplish safe shutdown of the AP1000 and that the design is partially exempted from the GDC 17. The applicant also stated that the 525-kV switchyard voltage was set to 523-kV (525-kV nominal) and the 230-kV switchyard was set to 233-kV. This is the anticipated voltage and is consistent with standard practice for grid studies at DE. For an AP1000 turbine trip event, adequate grid voltage is required for 3 seconds. The unit's electric generator will motor immediately following a turbine trip, providing megavolt ampere reactive (MVAR) power to support this voltage and, therefore, the generator bus voltage remains relatively stable.

In addition, the applicant stated that the above grid voltage evaluation results are verified during the preoperational testing identified in AP1000 DCD Section 14.2.10, which includes the following tests:

- 100 Percent Load Rejection (DCD Section 14.2.10.4.21)
- Plant Trip from 100 Percent Power (DCD Section 14.2.10.4.24)
- Loss of Offsite Power (DCD Section 14.2.10.4.26)

In a public meeting with the Nustart Consortium on April 7, 2009, there was an agreement that portions of BLN RAI 177, Question 08.02-3 (equivalent WLS RAI 5, Question 08.02-4) were not within the scope of the BLN COL, but rather within the scope of the AP1000 DC. This is considered a standard item applicable to all COL applications including WLS. Therefore, the staff finds that the relevant portions of RAI 5, Question 08.02-4 are resolved for WLS.

In RAI 5, Question 08.02-3, the staff requested that the applicant provide a discussion as to how single offsite power circuits complied with 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 17, "Electric Power Systems"; and GDC 18, "Inspection and Testing of Electric Power Systems," as well as with guidance in NUREG-0800, Section 8.2.II, and how the applicant intends to meet the requirements of 10 CFR 50.65. In a September 26, 2008, response, the applicant stated that there is no portion of the single offsite circuit required to comply with GDC 2, GDC 4, GDC 5, and GDC 18 and that these GDC are for structures, systems, and components (SSCs) important to safety. Based upon this response, the staff agrees that GDC 2 and GDC 4 do not apply to the AP1000 design.

With respect to GDC 5, the applicant stated that the transmission lines and switchyards are designed so the full output of the plants can be carried out to the network; the capacity is more than sufficient for any incoming power requirements. Based on the above, the staff concludes that since WLS Units 1 and 2 UATs and RATs are not shared among the units and the capacity of the offsite power system is more than sufficient compared to the minimal safety-related loads powered by the offsite power (battery chargers and UPS), the WLS Units 1 and 2 offsite power system design meets the requirements of GDC 5 and, therefore, the staff considers this item resolved.

With respect to GDC 17, the staff finds that the results of the grid stability analysis demonstrate the offsite source capacity and capability to power plant components during normal, shutdown, startup, and turbine trip conditions. The results of the failure modes and effects analysis demonstrate the reliability of the offsite source, which minimizes the likelihood of its failure under normal, abnormal and accident conditions. Therefore, the staff concludes that the WLS Units 1 and 2 offsite power systems design meets the requirements of GDC 17, as it is applicable to AP1000 design; therefore, the staff considers this item resolved.

With regard to GDC 18, NUREG-1793, Section 8.2.3.2 identifies COL Action Item 8.2.3.3-1 to demonstrate that the testing and inspection capability of the offsite power system be in conformance with GDC 18; therefore, this interface item must also be satisfied by the applicant.

The staff verified that WLS COL FSAR Section 8.3.1.4 has been revised to include implementation of procedures for periodic verification of proper operation of the onsite ac power system capability for automatic and manual transfer from the preferred power supply to the maintenance power supply and return from the maintenance power supply to the preferred power supply. The staff finds that the above satisfies the requirements of GDC 18 and is therefore acceptable.

WLS COL FSAR Section 17.6 describes implementation of the requirements of 10 CFR 50.65. As indicated therein, implementation of the Nuclear Energy Institute (NEI) 07-02A, "Generic FSAR Template Guidance for Maintenance Rule Program Description for Plants Licensed under 10 CFR Part 52," program description will determine the applicability of the maintenance requirements for the offsite power circuit. NEI 07-02A provides a template for presenting this information that also has been endorsed by the staff in a January 24, 2008, letter to NEI. The staff verified that the reference to this NEI report is in WLS COL FSAR Table 1.6-201. Since the scope of SSCs covered by the maintenance rule (MR) program is determined using the scoping procedures defined in the MR program description in accordance with NEI 07-02A, the offsite power system and its components will be evaluated for inclusion in the MR program in accordance with these scoping procedures during program implementation. The staff notes that NEI 07-02A, Section 17.X.1.5, "Risk Assessment and Risk Management per 10 CFR 50.65(a)(4)," addresses risk assessment and risk management from maintenance activities in accordance with 10 CFR 50.65(a)(4), and includes consideration of the issues associated with grid/offsite power system reliability as identified in NRC GL 2006-02, Items 5 and 6. Therefore, although detailed maintenance risk assessment is not anticipated in advance of the schedule defined in WLS COL FSAR Table 13.4-201, performance of "grid-risk-sensitive" maintenance activities is a necessary consideration of the program in accordance with NEI 07-02A guidance. Based on the above, the staff finds this item resolved.

With regard to the submerged or inaccessible electrical cable recommendations in GL 2007-01, "Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients," and the guidance in NUREG/CR-7000, and NUREG-0800 Section 8.2.III.1.L., WLS identified the standard content related to this item in its WLS COL FSAR Section 17.6.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 8.2.4:

> <u>*Submerged/Inaccessible Electrical Cables*</u>
>
> *In RAI 8.2-14, the staff asked the applicant to describe the inspection, testing and monitoring program to detect degradation of inaccessible or underground control and power cables that support equipment and other systems that are within the scope of 10 CFR 50.65. The description should include the frequency of testing and inspection. Guidance on the selection of electric cable condition monitoring can be found in Sections 3 and 4.5 of NUREG/CR-7000, "Essential Elements of an Electric Cable Condition Monitoring Program."*
>
> *In a letter dated May 6, 2010, the applicant stated that the Maintenance Rule (MR) program will not be implemented until prior to fuel load; as such, specific information necessary to determine appropriate inspections, tests and monitoring is not available at this time. In order to determine the method and frequency, a review of detailed design and procurement information is needed. The applicant also stated that the latest industry experience and other available information, including NUREG/CR-7000, will be followed in developing a cable condition monitoring program as part of the MR program. The applicant also committed to revise its FSAR to include condition monitoring of underground or inaccessible cables in its MR program. The commitment will be reflected in the COL application Part 2, FSAR Chapter 17, Section 17.6 as shown below.*
>
> > *Condition monitoring of underground or inaccessible cables is incorporated into the maintenance rule program. The cable condition monitoring program incorporates lessons learned from industry operating experience, addresses regulatory guidance, and utilizes information from detailed design and procurement documents to determine the appropriate inspections, tests and monitoring criteria for underground and inaccessible cables within the scope of the maintenance rule (i.e., 10 CFR 50.65). The program takes into consideration Generic Letter 2007-01.*
>
> *Based on the above, the staff concludes that the applicant's condition monitoring program for underground or inaccessible cables satisfies the recommendations of GL 2007-01, "Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients," and the guidance in*

> *NUREG/CR-7000 and NUREG-0800 Section 8.2.III.1.L. Therefore, this item is resolved subject to the verification that the VEGP COL FSAR has been updated to include applicable portions of the RAI response. This is identified as Confirmatory Item 8.2-3.*
>
> *Resolution of Standard Content Confirmatory Item 8.2-3*
>
> *Confirmatory Item 8.2-3 is an applicant commitment to revise its FSAR Section 17.6 to address condition monitoring of underground or inaccessible cables. The staff verified that the VEGP COL FSAR was appropriately revised. As a result, Confirmatory Item 8.2-3 is now closed.*

Supplemental Information

WLS SUPs 8.2-1 through 8.2-5 address information on the offsite power system requested by RG 1.206. RG 1.206 includes the request for detailed information about the offsite power system to allow the staff to determine the relative quality of the design. The requested information includes: a description of the overall grid configuration, interfaces with the grid operator, results of grid reliability studies, protocols for operation and pertinent historical grid reliability data.

- WLS SUP 8.2-1

With regard to WLS SUP 8.2-1, the applicant provided the following information:

> DE is a regulated, vertically integrated utility with regards to its electric generation and transmission operations. DE's NGD has a formal agreement titled Nuclear Switchyard Interface Agreement with the TSO which is DE's PD department. The PD department includes the Transmission Planning and Control Center (TCC), transmission System Operation Center (SOC), and Planning and Grid Operations (PGO). The Nuclear Switchyard Interface Agreement and the associated Department Directives serve as a communications protocol with the TSO.
>
> DE is also the transmission system provider. The TSP/TSO establishes a voltage schedule for the 525-kV and 230-kV switchyards. The nuclear power plant, while generating, is expected to supply or absorb reactive power to help regulate voltage in the 525/230-kV switchyard in accordance with TSP/TSO voltage schedule criteria. The TSP/TSO also maintains switchyard voltage such that voltage on the 26-kV isophase bus is within 0.95–1.05 pu of its nominal value.
>
> The plant's operator workstations monitor switchyard voltage, frequency, and other offsite power system parameters. The operator workstations are set to alert the nuclear plant operator if the grid may not be able to supply offsite power of sufficient voltage. Procedures direct the plant operators to contact the TSO and request a status of the most current contingency analysis for existing grid conditions. If the results of the contingency analysis indicate that insufficient voltage would exist in the switchyard, the procedures direct the plant operators to take appropriate actions.

The staff reviewed the information provided by the applicant on the functions of the TSO that establishes a voltage schedule for the WLS 500 kV switchyard and also maintains switchyard voltage such that steady state voltage on the 26 kV isophase bus is within 0.95-1.05 pu of nominal value. Based on the information provided by the applicant on the functions of the TSP/TSO and the detailed voltage and other requirements to be maintained by the TSP/TSO, the staff finds that the applicant has demonstrated that protocols are in place for WLS to remain cognizant of grid vulnerabilities in order to make informed decisions regarding maintenance activities critical to the electric system. This is, in part, consistent with GL 2006-2 in which one provision is to reduce the likelihood of losing offsite power. The staff finds that the information provided meets the requirements of GDC 17 and GDC 18 also conforms to the guidelines of RG 1.206. Therefore, the staff considers WLS SUP 8.2-1 acceptable.

- WLS SUP 8.2-2

With regard to WLS SUP 8.2-2, the applicant provided the following information:

> The Nuclear Switchyard Interface agreement between NGD and PD sets the requirements for transmission grid studies and analyses. These analyses demonstrate the capability of the offsite power system to support plant start up and shutdown.

The staff reviewed the information provided by the applicant on the Nuclear Switchyard Interface agreement. Based on the information provided by the applicant on the agreement, the staff finds that the applicant has demonstrated that analyses have been completed that demonstrate the capability of the grid to support operations, such as plant shutdown and startup. This is, in part, consistent with GL 2006-2 of which one provision is to reduce the likelihood of losing offsite power. The staff finds that the information provided complies with the requirements of GDC 17 and conforms to the guidelines of RG 1.206. Therefore, the staff considers WLS SUP 8.2-2 acceptable.

- WLS SUP 8.2-3

> DE’s Power Delivery (PD) Department is the approving grid organization for reliability studies performed on the area bulk electric system. PD conducts planning studies of the transmission grid on an ongoing basis. Model data used to perform simulation studies of projected future conditions is maintained and updated as load forecasts and future generation/transmission changes evolve. Studies are performed annually to assess future system performance in accordance with NERC reliability standards. These studies form a basis for identifying future transmission expansion needs.
>
> New large generating units requesting to connect to the area bulk electric system are required to complete the Large Generator Interconnection Procedure. The studies performed by DE TSO as part of this procedure, examine the generating unit (combined turbine-generator-exciter) and the main step-up transformer(s).

The staff reviewed the information provided by the applicant on planning and reliability studies and simulation studies. Based on the information provided by the applicant, the staff finds that the information provided complies with the requirements of GDC 17 and conforms to the guidelines of RG 1.206 and GL 2006-2. Therefore, the staff considers WLS SUP 8.2-3 acceptable.

- WLS SUP 8.2-4

With regard to WLS SUP 8.2-4, the applicant provided the following information:

> The Nuclear Switchyard Interface Agreement between NGD and PD demonstrates protocols in place for the plant to remain cognizant of grid vulnerabilities and make informed decisions regarding maintenance activities critical to the electrical system.
>
> In the operations horizon, the DE TSO continuously monitors real-time power flows and assesses contingency impacts. Operational planning studies are also performed using offline power flow study tools to assess near term operating conditions under varying load, generation, and transmission topology patterns.

Based on the information provided by the applicant on the functions of TSO, the staff finds that the applicant has demonstrated that protocols are in place for WLS to remain cognizant of grid vulnerabilities in order to make informed decisions regarding maintenance activities critical to the electric system. Therefore, the staff finds that the application complies with the requirements of GDC 17 and GDC 18, 10 CFR 50.60 and conforms to the guidelines of GL 2006-2 of which one of the provisions is to reduce the likelihood of losing offsite power.

- WLS SUP 8.2-5

With regard to WLS SUP 8.2-5, the applicant provided the last 12 years of average outage data available on the DE transmission system as follows:

1. The Momentary Average Interruption Frequency Index is 0.28 for the 230-kV system and 0.78 for the 525-kV system.
2. The Transmission System Average Interruption Frequency Index for sustained (>1 minute) outages is 0.08 for the 230-kV system and 0.37 for the 525-kV system.
3. The Transmission System Average Interruption Duration Index (minutes) is 31.8 for the 230-kV system and 210 for the 525-kV system.

The staff reviewed the supplemental information provided regarding the grid availability historical data and finds that the information complies with the requirements of GDC 17 and conforms to the guidelines of RG 1.206. Therefore, the staff considers WLS SUP 8.2-5 acceptable.

- WLS SUP 8.2-6

With regard to WLS SUP 8.2-6, the applicant provided the following information in WLS COL FSAR Section 8.2.1.2.3.

> Section 8.2.1.2.3 Plant Response to High Voltage Open Phase Condition
>
> A monitoring system is installed on the credited GDC 17 offsite power circuit that provides continuous open phase condition monitoring of the MSU transformer HV input power supply (see Reference 201). The system detects an open phase condition (with or without a concurrent high impedance ground on the HV side of the transformer) on one or more phases under all transformer loading conditions.

The open phase condition monitoring system provides an alarm to the operators in the control room should an open phase condition occur on the HV source to the MSU transformers. The system design utilizes commercially available components including state of the art digital relaying equipment and input parameters as required to provide loss of phase detection and alarm capability.

Additionally, a high-voltage open phase condition with or without a ground fault can manifest itself as an unacceptable voltage on the 6.9 kV medium voltage ES-1 and ES-2 buses during normal loading conditions. The presence of unacceptable voltages on the ES-1 and ES-2 buses results in isolation of the affected medium voltage bus from the offsite power supply and enables the onsite standby diesel generators to start and restore AC power to the ES-1 and ES-2 buses and associated defense-in-depth loads. The onsite AC power system is described in DCD Section 8.3.1.

Motor management relays for the medium voltage motors on ES-1 and ES-2 provide detection of unacceptably high negative sequence currents. High negative sequence current motor trips or other running load trips provide alarms in the MCR, which can assist in the detection of a high-voltage open phase condition with or without a ground fault. Electric circuit protection for the medium voltage system and equipment is described in DCD Section 8.3.1.1.1.1.

A high-voltage open phase condition with or without a ground fault can also manifest itself as an unacceptable voltage on the 480 VAC low-voltage buses powered from ES-1 and ES-2. The safety-related IDS battery chargers are powered from the low-voltage buses and continue to charge the IDS batteries unless the battery charger input or output monitored electrical parameters are unacceptable. If the monitored electrical parameters degrade to the point that the battery charger no longer provides sufficient DC bus voltage, the Class 1E electrical system DC bus receives power from the applicable IDS battery and the battery charger maintains isolation between the Non-Class 1E AC and Class 1E DC power systems which generates alarms in the MCR. The onsite AC power system is described in DCD Section 8.3.1 and the Class 1E DC power system is described in DCD Section 8.3.2.1.1.

Operator actions and maintenance and testing activities are addressed in procedures, as described in Section 13.5. Plant operating procedures, including off-normal operating procedures associated with the monitoring system will be developed prior to fuel load. Maintenance and testing procedures, including calibration, surveillance testing, setpoint determination and troubleshooting procedures associated with the monitoring system will be developed prior to fuel load.

Control Room operator and maintenance technician training associated with the operation and maintenance of the monitoring system will be conducted in accordance with the milestones for Non Licensed Plant Staff and Reactor Operator Training Programs in Table 13.4.201.

This supplement was added to the FSAR to address the issues in Bulletin 2012-01. The staff’s review and evaluation follows directly below.

NRC Bulletin 2012-01, "Design Vulnerability in Electric Power System"

In light of recent operating experience that involved the loss of one of the three phases of the offsite power circuit (i.e., loss of a single-phase) at Byron Station, Unit 2, the NRC issued Bulletin 2012-01, "Design Vulnerability in Electric Power System," on July 27, 2012, to all holders of operating and combined licenses requesting information about the facilities' electric power system designs. The above operating event resulted in neither the onsite nor the offsite electric power system being able to perform its intended safety functions (i.e., to provide electric power to the important to safety buses with sufficient capacity and capability to permit functioning of structures, systems, and components important to safety). NRC Bulletin 2012-01 was issued to operating and new reactor licensees to affirm compliance with GDC 17 requirements and to evaluate whether further NRC action is warranted to address this design vulnerability. Subsequently, the staff also issued RAI 108, Question 08-1, to the applicant for WLS Units 1 and 2, to address the matters described in NRC Bulletin 2012-01 and to ensure that the WLS design meets GDC 17.

In an October 23, 2012, letter, the applicant provided its response to RAI 108, Question 08-1, "Single-Phase Open Circuit Condition," for WLS Units 1 and 2. The proposed design utilized existing undervoltage relays on the ES-1 and ES-2 buses as well as existing undervoltage relays on the loads, on or downstream of the ES-1 and ES-2 buses. Based on review of this response, the staff could not determine whether the WLS Units 1 and 2 existing protection schemes would detect open circuit conditions on the high voltage side of a transformer connecting a GDC 17 offsite power circuit to the transmission system for all operating electrical system configurations and loading conditions.

On November 1, 2013, the NRC conducted a public meeting with representatives from the Nuclear Energy Institute and other industry representatives to discuss the industry initiative associated with resolving NRC Bulletin 2012-01. During the meeting, industry representatives provided feedback regarding their review of an offsite power two-phase open circuit event that occurred at Forsmark Nuclear Power Plant in Sweden (see NRC Information Notice 2006-18, Supplement 1: "Significant Loss of Safety-Related Electrical Power at Forsmark Unit 1 in Sweden"). The industry representatives informed the staff that their detailed analyses of this condition indicated that the proposed single-open phase detection system may not be sensitive enough to detect a two-phase open circuit condition. Therefore, the industry has taken the position that a two-phase open circuit condition must be considered when developing a resolution for the NRC Bulletin open phase issue.

GDC 17 requires, in part, "An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure: (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences, and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents." For AP1000 reactors, the main ac power system is non-Class 1E and not safety-related. During a loss of offsite power, ac power is supplied by the onsite standby diesel generators, which are also not safety-related. However, the ac power system is designed such that plant auxiliaries can be powered from the grid under all modes of operation. Furthermore, the ac power systems do supply power to equipment that is important to safety since that equipment serves defense-in-depth functions. The offsite power supply system provides power to the safety-related loads through the battery chargers, and both the offsite power system and the standby diesel generators provide defense-in-depth functions to supplement the capability of the safety-related passive systems for reactor coolant makeup and decay heat removal. In this regard, offsite power is the preferred power source, and supports the first line of defense. In addition, the safety analyses take credit for the grid remaining stable to maintain reactor coolant pump operation for three seconds following a turbine trip in accordance with the guidance of RG 1.206. Accordingly, these electric power systems are important to safety, and subject to the requirements of GDC 17. Consequently, it was the staff's position that AP1000 COL applicants address the design vulnerability identified in NRC Bulletin 2012-01.

Furthermore, it is the staff's position that an acceptable approach for passive designs includes the following four elements: (1) a dedicated automatic detection for an offsite power system single-phase open circuit condition with, and without, a high impedance ground fault condition on the high voltage side of the main power transformer including two open phase conditions under all loading and operating configurations; (2) an alarm in the main control room for operators to take manual actions if necessary; (3) an inspection, test, analysis, and acceptance criteria (ITAAC) to confirm that the analyses for developing the proper set points were completed in accordance with the acceptance criteria and to perform testing to demonstrate that the design functions as described in the WLS COL FSAR; and (4) procedures and training for the operating and maintenance staff. Since the system is continuously monitoring for open phase conditions using digital relays, the Staff concluded that the monitoring system automatically detects an open phase for either of the one or two open phase conditions (i.e. continuous monitoring meets the staff position of automatic detection). With regard staff position (4), Section 13.5 of this report states that the staff has reasonable assurance that the administrative procedures will be established to provide licensed operators and non-licensed plant staff with sufficient knowledge and operating experience to start up, operate, and maintain the plant in a safe manner. This approach ensures the required offsite ac power source with adequate capacity and capability is available to important safety equipment including safety-related battery chargers to meet their intended safety function in accordance with GDC 17 requirements.

In an August 28, 2014, supplemental response to RAI 108, Question 08-1, the applicant provided text that will be added to the next revision of the WLS COL FSAR, including but not limited to, ITAAC to confirm that the analyses for developing the proper set points were completed in accordance with the acceptance criteria and to perform testing to demonstrate that the design functions as described in the WLS COL FSAR. These proposed additions to the WLS COL FSAR and the ITAAC acceptably address the staff's position as that which is necessary to protect a plant with regard to an open phase condition as described in NRC Bulletin 2012-01, and that the WLS design meets GDC 17. Therefore, the staff considers this issue resolved and RAI 108, Question 08-1, closed pending the staff's confirmation that the revisions to the WLS COL FSAR noted above are incorporated in the WLS Units 1 and 2 application. RAI 108, Question 08-1 was being tracked as Confirmatory Item 8.2-1.

<u>Resolution of Confirmatory Item 8.2-1</u>

Confirmatory Item 8.2-1 is an applicant commitment to update its WLS COL FSAR and ITAAC to include details necessary to protect a plant with regard to an open phase condition, described in NRC Bulletin 2012-01. The applicant added the requested design description under WLS SUP 8.2-6 (above). The staff verified that the WLS COL FSAR and ITAAC were appropriately updated and that the WLS design meets GDC 17. Accordingly, the staff considers Confirmatory Item 8.2-1 related to RAI 108, Question 08-1, resolved.

Interface Requirements

The plant interfaces for the standard design of the AP1000 are discussed in AP1000 DCD Tier 2, Section 8.2.5 and DCD Table 1.8-1, Items 8.1, 8.2, and 8.3, where they are identified as "NNS" interfaces.

The applicant incorporated by reference AP1000 DCD Section 1.8. This section of the AP1000 DCD identifies certain interfaces with the standard design that have to be addressed in accordance with 10 CFR 52.47(a)(1)(vii).[2] As required by 10 CFR 52.79(d)(2), the COL application must demonstrate how these interface items have been met.

To satisfy AP1000 DCD, Tier 2, Table 1.8-1, Interface Item 8.1, the applicant provided a listing of the design criteria, regulatory guides, and IEEE standards in WLS COL FSAR Section 8.1.4.3 to which the applicant commits to as interface requirements for the offsite power system. The staff finds this information consistent with NUREG-0800, Section 8.1 as well as AP1000 DCD Table 1.8-1. Accordingly, the staff finds that this interface item for the offsite power system has been met.

With regard to plant AP1000 DCD, Tier 2, Table 1.8-1, Interface Item 8.2, the staff observed that in WLS COL FSAR Section 8.2.2, the applicant stated that the grid stability study has confirmed that the interface requirements for steady state load, nominal voltage, allowable voltage regulation, nominal frequency, allowable frequency fluctuation, and maximum frequency decay rate, have been met. In RAI 5, Question 08.02-6, the staff requested that the applicant provide the summary of the grid stability analysis results, the assumptions made, and the acceptance criteria for each case analyzed. Additionally, the staff requested that the applicant provide the nominal frequency, allowable frequency fluctuation, maximum frequency decay rate, and the limiting under-frequency values used for the RCP in the analysis. In a September 26, 2008, response to RAI 5, Question 08.02-6, the applicant provided the required parameter values and the associated analysis results. Additionally, the applicant stated that the WLS COL FSAR would be revised to include the parameter values. The staff has verified that Revision 4 to the WLS COL FSAR includes the proposed change. The staff has verified the change has been actualized in the WLS COL FSAR and concludes that the parameter values and the analysis results meet the AP1000 design requirements, the requirements of GDC 17 and the guidelines of RG 1.206. Accordingly, the staff considers this RAI resolved and AP1000 DCD Tier 2, Table 1.8-1, Interface Item 8.2 satisfied.

AP1000 DCD Tier 2, Table 1.8-1, Plant Interface Item 8.3 calls for a design feature that provides that "the protective devices controlling the switchyard breakers are set with consideration given to preserving the plant grid connection following a turbine trip." In the absence of a description of such a design feature, the staff asked the applicant to provide a reference as to where this issue is discussed in the application, or to provide a proposed revision to the application to address the issue (WLS RAI 74, Question 08.02-9).

[2] Following the update to 10 CFR Part 52 (72 *Federal Register* (FR) 49517), this provision has changed to 10 CFR 52.47(a)(25).

In an August 20, 2009, response to WLS RAI 74, Question 08.02-9, the applicant identified a proposed revision to WLS COL FSAR Section 8.2.1.2.2 that states, “The protective devices controlling the switchyard breakers are set with consideration given to preserving the plant grid connection following a turbine trip.” The staff verified that the WLS COL FSAR was updated to include this change and concludes that the switchyard arrangement, the protection of lines by independent high speed relaying, and the breaker failure scheme would preserve the WLS connection to the grid following a turbine trip satisfying the requirements of GDC 17. Therefore, the staff finds this interface has been met and the issue in WLS RAI 74, Question 08.02-9 resolved. On this basis, the staff considers COL Information Item 8.2.3.1-2, also resolved.

The staff reviewed the information supplied by the applicant and concludes that the applicant has adequately addressed AP1000 DCD Tier 2, Table 1.8-1, Interface Items 8.1, 8.2, and 8.3.

Inspections, Tests, Analyses and Acceptance Criteria

As part of the applicant’s resolution of electrical power issues, discussed in the subsection NRC Bulletin 2012-01, “Design Vulnerability in Electric Power System*,”* above, the applicant made changes to the WLS COL application, Part 10, Appendix B, “Inspections, Tests, Analysis and Acceptance Criteria.” The applicant proposed the following site-specific ITAAC for the Main ac Power System (ECS) to be added as new Item 4.g in Table 2.6.1-4, in the application, to that already required by AP1000 DCD Tier 1, Section 2.6.1. This ITAAC was not required by the staff to make the acceptability finding for resolution of NRC Bulletin 2012-01.

Design Commitment	Inspections, Tests, and Analyses	Acceptance Criteria
4.g) The ECS provides an alarm in the MCR [main control room] and automatic protection actuation if an undervoltage condition is detected on any one or more AC phases of either switchgear ECS-ES-1 or ECS-ES-2.	i) Testing of the as-built ECS will be conducted by simulating an undervoltage condition on ECS-ES-1 and ECS-ES-2 to confirm that an MCR alarm is generated when one or more ECS bus phase voltages is below setpoint on either switchgear ECS-ES-1 or ECS-ES-2.	i) Undervoltage relays on ECS-ES-1 and ECS-ES-2 provide alarm when one or more AC phases on the 6.9 kV buses are below setpoint.
	ii) Testing of the as-built ECS will be conducted by simulating an undervoltage condition on ECS-ES-1 and ECS-ES-2 to confirm that loss of one or more ECS bus phases automatically actuates the electrical protection function logic.	ii) Undervoltage relays on ECS-ES-1 and ECS-ES-2 initiate protective action when one or more AC phases on the 6.9 kV buses are below setpoint.

The applicant proposed the following site-specific ITAAC for the offsite power system to be added in the WLS application, Part 10, Appendix B, as new line item 7 in AP1000 DCD Tier 1, Table 2.6.12-1.

Design Commitment	Inspections, Tests, and Analyses	Acceptance Criteria
7. The credited GDC 17 off-site power source is monitored by an open phase condition monitoring system that can detect the following at the high voltage terminals of the transformer connecting to the off-site source, over the full range of transformer loading from no load to full load: (1) loss of one of the three phases of the offsite power source a. with a high impedance ground fault condition, or b. without a high impedance ground fault condition; or (2) loss of two of the three phases of the offsite power source a. with a high impedance ground fault condition, or b. without a high impedance ground fault condition. Upon detection of any condition described above, the system will actuate an alarm in the main control room.	i) Analysis shall be used to determine the required alarm set points for the open phase condition monitoring system to indicate the presence of open phase conditions described in the design commitment. ii) Testing of the credited GDC-17 off-site power source open phase condition monitoring system will be performed using simulated signals to verify that the as-built open phase condition monitoring system detects open phase conditions described in the design commitment and at the established set points actuates an alarm in the main control room.	i) Alarm set points for the open phase condition monitoring system to indicate the presence of open phase conditions as described in the design commitment have been determined by analysis. ii) Testing demonstrates the credited GDC 17 off-site power source open phase condition monitoring system detects open phase conditions described in the design commitment and at the established set points actuates an alarm in the main control room.

The evaluation of the applicant-proposed site-specific ITAAC item 7 is presented in the subsection, NRC Bulletin 2012-01, "Design Vulnerability in Electric Power System," above.

8.2.5 Post Combined License Activities

As discussed in the technical evaluation section above, the staff finds acceptable ITAAC items 4g and 7 as defined in the WLS COL application, Part 10, Appendix B.

8.2.6 Conclusion

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to the offsite power system, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the requirements of GDC 17 and GDC 18. The staff based its conclusion on the following:

- WLS COL 8.2-1 is acceptable because the applicant provided sufficient information involving the design details of the plant site switchyard, its interface with the local transmission grid, protective device settings, its testing and inspection plan and details of a FMEA performed for the plant site switchyard and meets the requirements of GDC 17 and GDC 18 and the guidelines of RG 1.206.

- WLS COL 8.2-2 is acceptable because the applicant provided sufficient information to demonstrate that the grid will remain stable to maintain RCP operation for three seconds following a turbine trip in accordance the requirements of GDC 17 and the guidelines of RG 1.206 and GL 2006-2.

- WLS CDI in WLS COL FSAR Section 8.2.1 is acceptable because the applicant provided sufficient information concerning the transformer area located next to each unit's turbine building in accordance with the requirements of GDC 17 and conforms to the guidelines of RG 1.206.

- WLS SUP 8.2-1 is acceptable because the applicant provided sufficient information on TSP/TSO, and the detailed voltage and other requirements to be maintained by TSP/TSO in accordance with the requirements of GDC 17 and GDC 18 and conforms to the guidelines of RG 1.206.

- WLS SUP 8.2-2 is acceptable because the applicant provided sufficient information describing the formal agreement between the NGD and PDs, which is the TSO, setting the requirements for transmission system studies and analyses in accordance with the requirements of GDC 17 and conforms to the guidelines of RG 1.206 and GL 2006-2.

- WLS SUP 8.2-3 is acceptable because the applicant provided sufficient information to describe the PD's responsibility for maintaining area bulk transmission system reliability and demonstrating, by power system simulation studies, projections, and analyses, the current and future reliability of the system in accordance with the requirements of GDC 17 and conforms to the guidelines of RG 1.206, and GL 2006-2.

- WLS SUP 8.2-4 is acceptable because the applicant provided sufficient information to demonstrate that protocols are in place for WLS to remain cognizant of grid vulnerabilities in order to make informed decisions regarding maintenance activities critical to the electric system in accordance with the requirements of GDC 17 and GDC 18, and 10 CFR 50.65, and conforms to the guidelines of RG 1.206 and GL 2006-2.

- WLS SUP 8.2-5 is acceptable because the applicant provided sufficient information regarding the 525-kV and 230-kV transmission lines outage data available over the past 12 years in accordance with the requirements of GDC 17 and GDC 18, 10 CFR 50.65, and conforms to the guidelines of RG 1.206.

- WLS SUP 8.2-6 is acceptable because the applicant provided sufficient information on a monitoring system for the offsite power circuit that provides continuous open phase condition monitoring that complies with the requirements of GDC 17 and conforms to the guidelines of RG 1.206 and GL 2006-2. Therefore, the staff finds this acceptable and considers the issue resolved.

- The applicant's addressing NRC Bulletin 2012-01, "Design Vulnerability in Electric Power System," is acceptable because the proposed additions to the WLS COL FSAR and the ITAAC acceptably address the staff position as that which is necessary to protect a plant with regard to an open phase condition as described in NRC Bulletin 2012-01, and that the WLS design meets GDC 17.

- The applicant provided sufficient information regarding the interfaces for standard design from the generic AP1000 DCD Table 1.8-1, Items 8.1, 8.2, and 8.3 in accordance with the requirements of GDC 17 for the staff to find acceptable.

- ITAAC – The applicant proposed the site-specific ITAAC for the Main ac Power System to be added to the WLS COL application, Part 10, Appendix B as new Item 4.g in Table 2.6.1-4. In addition, the applicant proposed site-specific ITAAC for the offsite power system to be added as new line Item 7 in AP1000 DCD Tier 1, Table 2.6.12-1. The staff finds that the proposed ITAAC items acceptably address the staff position with regard to an open phase condition as described in NRC Bulletin 2012-01, and that the WLS design meets the requirements of GDC 17.

8.2.A Site-Specific ITAAC for Offsite Power Systems

8.2.A.1 *Introduction*

This section specifically addresses the site-specific inspections, tests, analyses and acceptance criteria (SS-ITAAC), that the applicant proposed related to the offsite power system that are necessary and sufficient to provide reasonable assurance that the facility has been constructed and will operate in conformance with the COL, the provisions of the Atomic Energy Act of 1954, as amended, and NRC regulations.

8.2.A.2 Summary of Application

WLS COL FSAR, Revision 11, Section 14.3, incorporates by reference AP1000 DCD, Revision 19, Section 14.3, and contains the following supplemental information.

Supplemental Information

- STD SUP 14.3-1

The applicant provided supplemental information related to the offsite power system in STD SUP 14.3-1 in WLS COL FSAR Section 14.3.2.3.3.

8.2.A.3 *Regulatory Basis*

The regulatory basis for the review of the information incorporated by reference is addressed in NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," and its supplements. In addition, the acceptance criteria associated with the relevant requirements of NRC regulations for ITAAC are given in NUREG-0800, Section 14.3.

The applicable regulatory requirements for electrical SS-ITAAC are in 10 CFR 52.80(a), "Contents of applications; additional technical information."

8.2.A.4 Technical Evaluation

The staff reviewed WLS COL FSAR Section 14.3 and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to SS-ITAAC for offsite power systems. The results of the staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the design certification and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the report for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this report by use of italicized, double-indented formatting. Section 1.2.3 of this report provides an explanation of why the standard content material from the report for the reference application (VEGP) contains evaluation material from the report for the BLN Units 3 and 4, application.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 8.2.A.4:

<u>Supplemental Information</u>

- *STD SUP 14.3-1, addressing SS-ITAACs*

ITAAC Screening Summary Table 14.3-201 of the BLN FSAR identified the transmission switchyard and offsite power system as a site-specific system and selected them for ITAAC, but the table indicated "title only, no entry for COLA." Consequently, Section 2.6.12 of Part 10 of Appendix B, "License Conditions and ITTAC" of the BLN COL application (COLA) provided no ITAAC information for the transmission switchyard and offsite power system. The COL applicant must provide this site-specific ITAAC for compliance with 10 CFR 52.79(d) and 10 CFR 52.80(a). In RAI 14.3-1, the NRC staff stated that RG 1.206, CIII.7.2, Site-Specific ITAAC, recommends that applicants develop ITAAC for the site-specific systems that are designed to meet the significant interface requirements of the standard certified design, that is, the site-specific systems that are needed for operation of the plant (e.g., offsite power). Therefore, the applicant should justify why there is no ITAAC entry associated with offsite power, or revise Table 14.3-201 of the BLN FSAR to include ITAAC entries for the transmission switchyard and the offsite power system.

By letter dated June 24, 2008, the applicant stated that approved DCD Section 14.3 refers to the selection criteria and processes used for developing the AP1000 Certified Design Material (CDM) and identifies no interfaces (e.g., systems for storm drain, raw water, and closed circuit TV system, etc.) meeting this definition. Thus, according to the applicant, the CDM does not include ITAAC or a requirement for COL developed ITAAC for the offsite power interface system. The staff found the above response to be inconsistent with the requirements of 10 CFR 52.80(a), and guidance of NUREG-0800 Section 14.3 and RG 1.206.

Several discussions were held between the applicant and the NRC staff to discuss this issue. The staff pointed out that the offsite power system performs an important function in the passive designs as it provides power to the safety-related loads through battery chargers during normal, abnormal and accident conditions. It also provides power to those active systems that provide defense-in-depth capabilities for reactor coolant make-up and decay heat removal.

These active systems are the first line of defense to reduce challenges to the passive systems in the event of plant transients. The above function of the offsite power system in passive designs supports the need for ITAAC for these systems so that the staff can verify that (1) the designed and installed systems, structures, or components of the offsite power systems will perform as designed *and (2) the required single circuit from the transmission network satisfies the requirements of GDC 17.*

Subsequently, in a letter dated May 11, 2009, the applicant revised its response to RAI 14.3-1 and provided an ITAAC for the offsite power system to verify that the as-built offsite portion of the power supply from the transmission network to the interface with the onsite ac power system will satisfy the applicable provisions of GDC 17. Specifically, the ITAAC shall verify:

(1) A minimum of one offsite circuit supplies electric power from the transmission network to the interface with the onsite portions of the ac power system.

(2) Each offsite circuit interfacing with the onsite ac power system is adequately rated to supply assumed loads during normal, abnormal and accident conditions.

(3) During steady state operation, each offsite circuit is capable of supplying required voltage to the interface with the onsite ac power system that will support operation of assumed loads during normal, abnormal and accident conditions.

(4) During steady state operation, each offsite circuit is capable of supplying required frequency to the interface with the onsite ac power system that will support operation of assumed loads during normal, abnormal and accident conditions.

(5) The fault current contribution of each offsite portion circuit is compatible with the interrupting capability of the onsite ac power system fault current interrupting devices.

(6) The reactor coolant pumps continue to receive power from either the main generator or the grid for a minimum of 3 seconds following a turbine trip.

To ensure that the requirements of GDC 17 for the adequacy of the offsite power source within the standard design scope are met, the proposed ITAAC would verify the capacity and capability of the offsite source to feed the onsite power system. The proposed ITAAC provides for the inspection of the connection of the offsite source to the onsite power system.

Additionally, the applicant identified all associated changes that will be made in a future revision of the Bellefonte FSAR. On the basis of its review, the staff finds that the applicant has adequately addressed the site-specific ITAAC for the offsite power system so that the staff can verify that the designed and installed systems, structures, or components of the offsite power system will perform as designed. Therefore, the staff concludes that the applicant meets the requirements of 10 CFR 52.79(d) and 10 CFR 52.80(a), and the guidance of SRP 14.3 and RG 1.206. The applicant will revise the BLN COL FSAR to include the proposed ITAAC for offsite power system. This is identified as

Confirmatory Item 8.2A-1, pending NRC review and approval of the revised BLN COL FSAR.

<u>Resolution of Standard Content Confirmatory Item 8.2A-1</u>

The applicant proposed a license condition in Part 10 of the VEGP COL application, which will incorporate the ITAAC identified in Appendix B. Appendix B includes ITAAC for the offsite power system. The license condition's proposed text is evaluated in Chapter 1 of this SER.

Confirmatory Item 8.2A-1 required the applicant to update its FSAR to include proposed ITAAC for the offsite power system. The NRC staff verified that the VEGP COL application was appropriately updated. The ITAAC associated with the offsite power system are shown in VEGP COL Part 10, Appendix B, Table 2.6.12-1. Table 8.2A-1 of this SER reflects this table. As a result, Confirmatory Item 8.2A-1 is resolved. Therefore, the staff will include the ITAAC for the offsite power system in the license.

8.2.A.5 *Post Combined License Activities*

For the reasons discussed in WLS COL FSAR, Part 10, Appendix B, Table 2.6.12-4, "Offsite Power System," the staff finds the ITAAC proposed by the applicant acceptable.

8.2.A.6 *Conclusion*

The staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the requirements of GDC 17 and GDC 18.

8.3 Onsite Power Systems

8.3.1 AC Power Systems

8.3.1.1 *Introduction*

The onsite ac power system includes those standby power sources, distribution systems, and auxiliary supporting systems provided to supply power to safety-related equipment or equipment important to safety for all normal operating and accident conditions. In the AP1000 passive reactor design used at WLS, the onsite ac power system is a non-Class 1E system that provides reliable ac power to the various system electrical loads. It does not perform any safety-related functions. These loads enhance an orderly shutdown under emergency conditions when offsite power is not available. Additional loads for investment protection can be manually loaded on the standby power supplies. Diesel generator sets are used as the standby power source for the onsite ac power system.

8.3.1.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 8.3 incorporates by reference AP1000 DCD, Revision 19, Section 8.3, and contains COL information items and supplemental information to address any departures and/or supplements.

COL Information Items

- WLS COL 8.3-1

WLS COL 8.3-1 describes: (1) the grounding grid system design within the plant boundary; and (2) a lightning protection risk assessment and general system design for the switchyard and buildings comprising WLS Units 1 and 2. This COL information item is also referenced in WLS COL FSAR Section 8.3.3.

- STD COL 8.3-2

STD COL 8.3-2 describes the details of: (1) Class 1E and non-Class 1E battery maintenance, clearing of faults, and testing of chargers and voltage regulating transformers; (2) the bases of operation, inspection, and maintenance procedures for the onsite standby diesel generators; and (3) procedures for the periodic testing of penetration overcurrent protective devices. This COL information item is also referenced in WLS COL FSAR Section 8.3.3.

Supplemental Information

- WLS SUP 8.3-1

WLS SUP 8.3-1 states that the site conditions provided in WLS COL FSAR Sections 2.1 and 2.3 are bounded by the standard site conditions used to rate the diesel engine and the associated generator in AP1000 DCD Section 8.3.1.1.2.3.

8.3.1.3 *Regulatory Basis*

The regulatory basis for the review of the information incorporated by reference is addressed in NUREG-1793, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design,” and its supplements. In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the ac power systems are given in NUREG-0800, Section 8.3.1.

The regulatory basis for the review of WLS COL 8.3-1, addressing the grounding and lightning protection systems are the guidelines of:

- RG 1.204, “Guidelines for Lightning Protection of Nuclear Power Plants”
- IEEE Standard 80, “Guide for Safety in AC Substation Grounding”
- IEEE Standard 665, “Guide for Generating Station Grounding”
- IEEE C.62.23, “Application Guide for Surge Protection of Electric Generating Plants”
- National Fire Protection Association (NFPA) 780, “Standard for the Installation of Lightning Protection Systems”

The regulatory bases for the review of the part of STD COL 8.3-2 addressing the recommendations in operation, inspection, and maintenance procedures for the non-Class 1E onsite standby diesel generators are the manufacturer’s guidelines. The regulatory bases for the review of the part of STD COL 8.3-2 addressing procedures for penetration protective device testing are the guidelines of RG 1.63, Revision 3, "Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants." The regulatory basis for the review of WLS SUP 8.3-1 was the standard site conditions outlined in NUREG-1793. The regulatory bases for acceptance of STD SUP 8.3-2 is the requirements of GDC 18, "Inspection and Testing of Electric Power Systems."

8.3.1.4 *Technical Evaluation*

The staff reviewed WLS COL FSAR Section 8.3.1 and checked the referenced DCD to ensure that the combination of the DCD information incorporated by reference and the information included in WLS COL application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the ac power systems. The results of the staff's evaluation of the information incorporated by reference in the application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the design certification and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS application. This standard content material is identified in this report by use of italicized, double-indented formatting. Section 1.2.3 of this report provides an explanation of why the standard content material from the SER for the reference application (VEGP) contains evaluation material from the SER for the BLN Units 3 and 4 application.

The staff reviewed the information in the WLS COL FSAR:

<u>AP1000 COL Information Items</u>

- COL 8.3-1

The staff reviewed COL 8.3-1 related to the AP1000 DCD COL Information Item 8.3-1 that states:

> Combined License applicants referencing the AP1000 certified design will address the design of grounding and lightning protection.

The commitment was also captured as COL Action Item 8.3.1.6-1 in NUREG-1793, Appendix F, which states:

> The COL applicant will provide the design of the site-specific grounding and lightning protection.

The staff reviewed the resolution to COL Information Item 8.3-1, related to the ground grid system and lightning protection included under WLS COL FSAR Section 8.3. The staff's evaluation is described below.

The applicant stated that a grounding system calculation was performed to establish a ground grid design within the plant boundary resulting in step and touch potentials near equipment that are within the acceptable limit for personnel safety. Computer analysis utilized actual resistivity measurements from soil samples taken at the plant site, and were used to create a soil model for the plant site. The ground grid conductor size was then determined using the methodology outlined in IEEE Standard 80, and a grid configuration for the site was created. The grid configuration was modeled in conjunction with the soil model. The resulting step and touch potentials were calculated, and found to be within the acceptable limit. Based on the above, the staff concludes that IEEE Standard 80 provides an acceptable method for determining the right size for ground conductors; therefore, the COL information item provided by the applicant on station grounding grid is acceptable.

In reference to lightning protection, the applicant states that in accordance with IEEE 665, a lightning protection risk assessment for the buildings comprising the WLS was performed based on the methodology in NFPA 780. The tolerable lightning frequency for each of the buildings was determined to be less than the expected lightning frequency; therefore, lightning protection is required for WLS buildings in accordance with NFPA 780 and IEEE C.62.23. The zone of protection is based on the elevations and geometry of the structures. It includes the space covered by a rolling sphere having a radius sufficient enough to cover the building to be protected. The zone of protection method is based on the use of ground masts, air terminals and shield wires. Either copper or aluminum is used for lightning protection. Lightning protection grounding is interconnected with the station or switchyard grounding system. Based on the above, the staff concludes that the applicant adequately addressed the guidance of IEEE Standard 80, IEEE Standard 665, IEEE C.62.23, and NFPA 780, and provided an acceptable method for lightning protection; therefore, the supplemental information provided by the applicant on lightning protection is acceptable.

With regard to the protection of electrical penetration assemblies and the onsite standby diesel generator inspection and maintenance, WLS identified the standard content related to these two items in its WLS COL FSAR Section 8.3. The following portion of this technical evaluation section is reproduced from VEGP SER Section 8.3.1.4.

- STD COL 8.3-2

The staff reviewed STD COL 8.3-2 related to AP1000 DCD COL Information Item 8.3-2, which states, in part:

> *The Combined License applicant will establish plant procedures as required for:*
>
> – *Periodic testing of penetration protective devices*

– *Diesel generator operation, inspection and maintenance in accordance with manufacturers' recommendations*

The commitment was also captured as COL Action Items 8.3.1.2-1 and 8.4.1-1 in the staff's FSER for the AP1000 DCD, Appendix F (NUREG-1793), which state:

> *The COL applicant will establish plant procedures for preoperational testing to verify proper operation of the ac power system. (COL Action Item 8.3.1.2-1).*
>
> *The COL applicant will establish plant procedures for periodic testing of penetration protective devices. (COL Action Item 8.4.1-1).*

A part of standard information item, STD COL 8.3-2, was provided by the applicant describing the bases of the recommendations in operation, inspection, and maintenance procedures for the onsite standby diesel generators. This part of STD COL 8.3-2 is addressed in BLN COL FSAR Section 8.3.1.1.2.4.

A part of standard information item, STD COL 8.3-2, was provided by the applicant describing procedures for the testing of penetration protective devices. This portion of STD COL 8.3-2 is addressed in BLN COL FSAR Section 8.3.1.1.6.

The NRC staff reviewed the resolution to COL information item, STD COL 8.3-2, related to testing procedures for standby diesel generators and electrical penetrations included under Section 8.3 of the BLN COL FSAR. The NRC staff's evaluation follows.

For the operation, inspection and maintenance for diesel generators, the applicant's procedures will consider both the diesel generator manufacturer and industry diesel working group recommendations.

In RAI 8.3.1-2, the NRC staff stated that COL Action Item 8.3.1.2-1 in the NRC's FSER for the AP1000 DCD (NUREG-1793), contains the following discussion:

> *Preoperational tests are conducted to verify proper operation of the ac power system. The preoperational tests include operational testing of the diesel load sequencer and diesel generator capacity testing. The diesel generators are not safety-related and will be maintained in accordance with the requirements of the overall plant maintenance program. This program will cover the preventive, corrective, and predictive maintenance activities of the plant systems and equipment and will be presented in the COL application. This COL information is discussed in DCD Tier 2, Section 8.3.3, "Combined License Information for Onsite Electrical Power."*

In RAI 8.3.1-2, the applicant was asked to provide a reference to where the preoperational testing program and the preventive, corrective, and predictive *maintenance activities for the diesel generators are discussed in the application, or provide a proposed revision to the application to address this issue.*

In a letter dated April 6, 2009, the applicant stated that COL Action Item 8.3.1.2-1 in Appendix F of the FSER does not indicate that "pre-operational testing" of the diesel generators has been addressed in the DCD. Pre-operational testing of the ac power system is described in FSER Section 14, DCD Section 14, and BLN COL FSAR Chapter 14. Specifically, DCD Sections 14.2.9.2.15 and 14.2.9.2.17 address the onsite ac power system and diesel generator testing, including diesel generator capacity and sequencer tests. BLN COL FSAR Section 14.2.9.4.23 describes testing of the offsite power system. The NRC staff agrees that pre-operational testing of the diesel generators is addressed in DCD Section 14.2.9.2.17 and was found acceptable by the staff as indicated in FSER NUREG-1793 Section 14.2.9. Based on the above, the NRC staff finds that the applicant's response to the portion of the RAI regarding COL areas of responsibility is acceptable.

In addition, the applicant stated that BLN COL FSAR Section 8.3.1.1.2.4 will be revised to include inspection and maintenance (including preventive, corrective, and predictive maintenance) procedures considering both the diesel generator manufacturer's recommendations and industry diesel working group recommendations.

The NRC staff concludes that following the manufacturer and industry diesel generator working group recommendations for onsite standby diesel generator inspection and maintenance including preventive, corrective, and predictive maintenance provides reasonable assurance that the diesel generators will be adequately maintained. Therefore, DCD COL Information, Item 8.3-2 and FSER COL Action Item 8.3.1.2-1 are resolved subject to the verification that the BLN COL FSAR has been updated to include applicable portions of the RAI response. This is identified as Confirmatory Item 8.3.1-1.

With regard to establishing plant procedures for periodic testing of protective devices that provide penetration overcurrent protection, the applicant will implement procedures to periodically test a sample of each different type of overcurrent device. Testing includes:

- *Verification of thermal and instantaneous trip characteristics of molded case circuit breakers*
- *Verification of long time, short time, and instantaneous trips of medium voltage air circuit breakers*
- *Verification of long time, short time, and instantaneous trips of low voltage air circuit breakers*

Because the above testing is consistent with the recommendation of RG 1.63, the NRC staff concludes that the above information satisfies COL Information Item 8.3-2 and FSER COL Action Item 8.3.1.6-1, and that these items are resolved.

<u>*Resolution of Standard Content Confirmatory Item 8.3.1-1*</u>

Confirmatory Item 8.3.1-1 required the applicant to update its FSAR to specify that onsite standby diesel generator inspection and maintenance (including preventive, corrective, and predictive maintenance) procedures will consider both the diesel generator manufacturer's recommendations and industry diesel working group recommendations. The NRC staff verified that the VEGP COL FSAR was appropriately updated. As a result, Confirmatory Item 8.3.1-1 is resolved.

<u>Supplemental Information</u>

- WLS SUP 8.3-1

In WLS SUP 8.3-1, the applicant stated that its site conditions provided in WLS COL FSAR Sections 2.1 and 2.3 were bounded by the standard site conditions in AP1000 DCD Section 8.3.1.1.2.3 used to rate the diesel engine and the associated generator. The staff agrees that the WLS site conditions are bounded by the standard site conditions used to determine the rating.

- STD SUP 8.3-2

The applicant provided information in STD SUP 8.3-2 to include implementation of procedures for periodic verification of proper operation of the onsite ac power system capability for automatic and manual transfer from the preferred power supply to the maintenance power supply and return from the maintenance power supply to the preferred power supply. The staff finds that the above satisfies the requirements of GDC 18 and is acceptable.

8.3.1.5 *Post Combined License Activities*

There are no post COL activities related to this section.

8.3.1.6 *Conclusion*

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to onsite ac power systems, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the application are documented in NUREG-1793 and its supplements.

In addition, the staff compared the COL information items, the supplemental information, the interfaces for standard design, and the proposed design changes and corrections in the application to the relevant NRC regulations, guidance in NUREG-0800, Section 8.3.1, and other NRC regulatory guides and concludes that the applicant is in compliance with NRC regulations. The staff based its conclusion on the following:

- COL 8.3-1 is acceptable because the applicant provided sufficient information related to the grounding grid system design and lightning protection consistent with the recommendations of RG 1.204, IEEE Std. 80, IEEE Std. 665, IEEE Std. C.62.23, and NFPA 780.

- STD COL 8.3-2 is acceptable because the applicant demonstrated conformance to preoperational testing of the diesel generators and periodic testing of the penetration overcurrent protective devices consistent with manufacturers' guidelines for non-Class 1E onsite diesel generators and the recommendations of RG 1.63.

- WLS SUP 8.3-1 is acceptable because the applicant demonstrated conformance to site-specific conditions that are bounded by the standard site conditions in the AP1000 DCD for rating the diesel generator.

- STD SUP 8.3-2 is acceptable because the applicant established procedures for periodic verification of offsite power system capacity for automatic and manual transfer from the preferred power supply to maintenance power supply and vice versa, demonstrating compliance with the requirements of GDC 18.

8.3.2 Direct Current Power Systems

8.3.2.1 *Introduction*

The dc power systems include those dc power sources and distribution systems provided to supply motive or control power to safety-related equipment. Batteries and battery chargers serve as the power sources for the dc power system and convert power from the dc distribution system to ac instrumentation, and control power, as required. These components can provide a UPS that furnishes a continuous, highly reliable source of ac supply.

The AP1000 dc power system is comprised of independent Class 1E and non-Class 1E dc power systems. Each system consists of ungrounded stationary batteries, dc distribution equipment, and UPS.

8.3.2.2 *Summary of Application*

WLS COL FSAR, Revision 11, Section 8.3 incorporates by reference AP1000 DCD, Revision 19, Section 8.3. AP1000 DCD Section 8.3 also includes Section 8.3.2. AP1000 DCD, Revision 19 includes a standard COL information Item (STD COL 8.3-2) and a standard departure (STD DEP 8.3-1). However, the applicant proposed a COL information item, supplemental information, and a site-specific AP1000 FSAR Tier 2 departure (WLS DEP 8.3-1).

Tier 2 Departure

- WLS DEP 8.3-1

The applicant added Departure WLS DEP 8.3-1 to the WLS COL FSAR, revising AP1000 DCD Section 8.3.2.2 to add information on the Class 1E battery chargers' voltage regulating.

AP1000 COL Information Item

- STD COL 8.3-2

STD COL 8.3-2 describes the details of: (1) procedures for inspection, maintenance, and testing of Class 1E batteries; (2) the clearing of ground faults on the Class 1E dc power system; and (3) information related to periodic testing for the battery chargers and voltage regulating transformers.

Supplemental Information

- STD SUP 8.3-1

The applicant provided supplemental information stating that there are no site-specific non-Class 1E dc loads connected to the Class 1E dc system.

8.3.2.3 *Regulatory Basis*

The regulatory basis for the review of the information incorporated by reference is addressed in NUREG-1793, and its supplements. In addition, the acceptance criteria associated with the relevant requirements of NRC regulations for the dc power systems are given in NUREG-0800, Section 8.3.2.

The regulatory basis for review of WLS DEP 8.3-1 is established in:

- GDC 17, "Electric Power Systems"
- GDC 18, "Inspection and Testing of Electric Power Systems"
- RG 1.129, Revision 2, "Maintenance, Testing, and Replacement of Vented Lead-Acid Storage Batteries for Nuclear Power Plants
- RG 1.75, "Criteria For Independence Of Electrical Safety Systems," Revision 3
- IEEE Standard 384, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuit,"
- IEEE Standard 450, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications"

The regulatory basis for the review of WLS COL 8.3-1 is 10 CFR Part 52, Appendix D, Section VIII.B.5 and the guidance of IEEE 384, and RG 1.75.

The regulatory basis for the review of STD COL 8.3-2 is 10 CFR Part 52, Appendix D, Section VIII.B.5 and manufacturer's recommendations.

8.3.2.4 *Technical Evaluation*

The staff reviewed WLS COL FSAR Section 8.3.2 and checked the referenced DCD to ensure that the combination of the DCD information incorporated by reference and the information in the COL application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the COL application and the DCD information incorporated by reference addresses the required information relating to the dc power systems. The results of the staff's evaluation of the information incorporated by reference in the application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the design certification and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP

Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews.

- The staff compared the VEGP COL FSAR to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff completed its review and found the evaluation performed for the standard content to be directly applicable to the application. This standard content material is identified in this report by use of italicized, double-indented formatting. Section 1.2.3 of this report provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the BLN Units 3 and 4, application. However, the applicant took an exception and included the following FSAR Tier 2 departure WLS DEP 8.3-1 from the AP1000 DCD:

Tier 2 Departure

- WLS DEP 8.3-1 and STD COL 8.3-2

The AP1000 DCD states that the Class 1E battery chargers and Class 1E voltage regulating transformers are designed to limit the input (ac) current to an acceptable value under faulted conditions on the output side. In a December 5, 2011, letter, the applicant took a departure from AP1000 DCD Section 8.3.2.2 and added information on the Class 1E voltage regulating transformers current limiting features. The applicant added Departure (WLS DEP 8.3-1) to the COLA, Part 7, Section A, “Summary of FSAR Departures from the DCD,” and added a new Section 8.3.2.2 in WLS COL FSAR.

The departure relates to the AP1000 DCD Class 1E voltage regulating transformers current limiting features. Specifically, the applicant added a Tier 2 departure in WLS COL FSAR Section 8.3.2.2 where the isolation and protection function is provided by built-in circuit breakers between Class 1E loads and the non-Class 1E ac power source. The AP1000 voltage regulating transformers do not have active components to limit current. The applicant stated that the information in the similar departure for the reference COL is slightly different due to a certain supplier Deficiency Report (SDR) not being available at the time of the reference COL submittal; therefore, the WLS departure is site-specific. The applicant further stated that this Tier 2 departure does not impact the required design function (i.e., isolation) and, therefore, does not require NRC approval pursuant to 10 CFR Part 52, Appendix D, Section VIII.B.5. The staff finds the applicant’s assessment consistent with IEEE 384 and RG 1.75 and, therefore, acceptable.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 8.3.2.4:

<u>AP1000 COL Information Item</u>

- *STD COL 8.3-2, involving the inspection, maintenance, and testing of Class 1E batteries and clearing of ground faults on the Class 1E dc system.*

The NRC staff reviewed STD COL 8.3-2 related to COL Information Item 8.3-2. COL Information Item 8.3-2 states (in part):

> *The Combined License applicant will establish plant procedures as required for:*
>
> – *Clearing ground fault on the Class 1E dc system*
>
> – *Checking sulfated battery plates or other anomalous conditions through periodic inspections*
>
> – *Battery maintenance and surveillance (for battery surveillance requirements, refer to DCD Chapter 16, Section 3.8)*

The commitment was also captured as COL Action Item 8.4.1-1 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:

> *The COL applicant will establish plant procedures for periodic testing of penetration protective devices. (COL Action Item 8.4.1-1)*

The Class 1E 125 volts direct current (Vdc) system components undergo periodic maintenance tests to determine the condition of the system. The applicant has established procedures for inspection and maintenance of Class 1E batteries and non-Class 1E batteries. Class 1E battery maintenance and service testing is performed in conformance with RG 1.129. Batteries are inspected periodically to verify proper electrolyte levels, specific gravity, cell temperature and battery float voltage. Cells are inspected in conformance with IEEE 450 and vendor recommendations. In addition, the applicant has established procedures for clearing of ground faults on the Class 1E dc system. The battery testing procedures are written in conformance with IEEE 450 and the Technical Specifications. The NRC staff concludes that the applicant has established procedures for inspection and maintenance of Class 1E and non-Class 1E batteries to satisfy COL Information Item 8.3-2; therefore, this item is resolved.

With regard to periodic testing of electrical penetration protective devices (COL Action Item 8.4.1-1) for dc systems, the applicant has not addressed periodic testing of the penetration over load protective devices related to dc systems. In RAI 8.3.1-1, the staff requested that the applicant address the periodic testing of the electrical penetration primary and backup protective devices protecting Class 1E and non-Class 1E dc circuits. In a letter dated January 2, 2009, the applicant stated that the BLN COL FSAR will be revised in the next COLA submittal to include periodic testing of the electrical penetration primary and backup protective devices protecting Class 1E and non-Class 1E dc circuits, as well as control of protective devices. The staff has reviewed the *information in the applicant's response, which provided for the testing of Class 1E and non-Class 1E dc penetration overload protection devices. The staff also reviewed the proposed change to BLN COL FSAR Section 8.3.1.1.6 and concludes that COL Action Item 8.4.1-1 is resolved subject to the verification that the BLN COL FSAR has been updated to include portions of the RAI response. This is identified as Confirmatory Item 8.3.2-1.*

Resolution of Standard Content Confirmatory Item 8.3.2-1

Confirmatory Item 8.3.2-1 required the applicant to update its FSAR to provide for the testing of Class 1E and non-Class 1E dc penetration overload protection devices. The NRC staff verified that the VEGP COL FSAR was appropriately updated. As a result, Confirmatory Item 8.3.2-1 is resolved.

WLS Resolution of Standard Content Confirmatory Item 8.3.2-1

The staff verified that WLS COL FSAR Section 8.3.1.1.6 was appropriately updated. As a result, standard content Confirmatory Item 8.3.2-1 is resolved for WLS.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 8.3.2.4:

Evaluation of Tier 2 Departure STD DEP 8.3-1 and Revised STD COL 8.3-2

In a letter dated June 18, 2010, Westinghouse provided a response to Open Item OI-SRP8.3.2-EEB-09, Revision 3, related to the periodic testing of battery chargers and voltage regulating transformers. The response included a COL information item to be added to AP1000 DCD Section 8.3.3 to ensure that periodic testing is performed on the battery chargers and voltage regulating transformers. Specifically, this section will be revised to include the following COL information item:

> *The Combined License applicant will establish plant procedures as required for:*
>
> > *Combined License applicants referencing the AP1000 certified design will ensure that periodic testing is performed on the battery chargers and voltage regulating transformers.*

In a letter dated October 15, 2010, the applicant submitted its response to address the above identified AP1000 DCD revision to the Section 8.3.3 COL information item regarding battery charger and voltage regulating transformer testing. The applicant stated that procedures are established for periodic testing of the Class 1E battery chargers and the Class 1E regulating transformers in accordance with the manufacturer recommendations. The battery chargers and regulating transformers are tested periodically in accordance with manufacturer recommendations. Circuit breakers in the Class 1E battery chargers and Class 1E voltage regulating transformers that are credited for an isolation function are tested through the use of breaker test equipment. This verification confirms the ability of the circuit to perform the designed coordination and *corresponding isolation function between Class 1E and non-Class 1E components. Circuit breaker testing is done as part of the MR program and testing frequency is determined by that program. Fuses/fuse holders that are included in the isolation circuit are visually inspected. Class 1E battery chargers are tested to verify current limiting characteristic utilizing manufacturer recommendation and industry practices. Testing frequency is in accordance with that of the associated battery.*

The applicant clarified that the voltage regulating transformers do not have active components to limit current and, therefore, the voltage regulating transformer in combination with fuses and/or breakers will interrupt the input or output (ac) current under faulted conditions on the output side. The NRC staff finds this to be inconsistent with AP1000 DCD Section 8.3.2.2, which states that Class 1E voltage regulating transformers are designed to limit the input (ac) current to an acceptable value under faulted conditions on the output side. As such the use of the breakers/fuses for regulating transformers for isolation function in lieu of current limiting characteristics as presented in the AP1000 DCD is a departure for VEGP. The applicant stated that Part 7 of the COL application will be revised to include a departure from AP1000 DCD Section 8.3.2.2 clarifying the current limiting feature of voltage regulating transformers. The applicant has included, in its response, the appropriate changes related to the above departure that will be included in VEGP COL FSAR Sections 8.3.2.1.4 and 8.3.2.2, in Chapter 1, Table 1.8-201 and in Part 7 of the VEGP COL application. These changes will be included in a future revision to the VEGP COL application.

The NRC staff has reviewed the proposed changes to the VEGP COL application and concludes that the applicant has provided sufficient information regarding the isolation function and the periodic inspection and testing of the isolating devices for the Class 1E battery chargers and Class 1E voltage regulating transformers. In addition, the staff finds that, although the use of the breakers/fuses for regulating transformers isolation function in lieu of current limiting characteristics as presented in the AP1000 DCD is a departure for VEGP, the departure is acceptable because the use of the breakers/fuses for regulating transformers for isolation function is consistent with the recommendations in IEEE-384, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits," endorsed by RG 1.75. Therefore, COL Information Item STD DEP 8.3-1 and the revised STD COL 8.3-2 are resolved subject to NRC staff verification of the revision to the VEGP COL FSAR sections discussed above. This is being tracked as Confirmatory Item 8.3.2-2.

Resolution of Standard Content Confirmatory Item 8.3.2-2

Confirmatory Item 8.3.2-2 is an applicant commitment to revise its FSAR Table 1.8-201 and Section 8.3.2.1.4 to address COL Information Item STD COL 8.3-2 and a departure, STD DEP 8.3-1. The staff verified that the VEGP COL FSAR was appropriately revised. As a result, Confirmatory Item 8.3.2-2 is now closed.

<u>WLS Resolution of Standard Content Confirmatory Item 8.3.2-2</u>

In a December 5, 2011, letter, the applicant proposed a revision to FSAR Table 1.8-201, "Summary of FSAR Departures," WLS COL FSAR Section 8.3.2.1.4, "Maintenance and Testing," and WLS COL FSAR Section 8.3.2.2, "Analysis." The change to WLS COL FSAR Section 8.3.2.1.4 included a left margin annotation for STD COL 8.3-2, to establish procedures for periodic testing of the Class 1E battery chargers and Class 1E voltage regulating transformers. The applicant also took exception to STD DEP 8.3-1 and provided a site-specific departure, WLS DEP 8.3-1. The staff confirmed that Standard Content Confirmatory Item 8.3.2-2 and Departure 8.3-1 related changes were included in WLS COL FSAR. Accordingly, the staff considers this item resolved.

The following portion of this technical evaluation section is reproduced from BLN SER Section 8.3.2.4:

> *<u>Supplemental Information</u>*
>
> - *STD SUP 8.3-1*
>
> *STD SUP 8.3-1 was provided by the applicant indicating that there are no site-specific non-Class 1E dc loads connected to the Class 1E dc system. The staff finds this acceptable because it is consistent with the guidance in RG 1.206.*
>
> *<u>Evaluation of Site-specific Response to Standard Content</u>*
>
> *In VEGP COL FSAR, Revision 2, the VEGP applicant changed the number of the supplemental information item from STD SUP 8.3-1 to STD SUP 8.3-3. The associated VEGP COL FSAR, Revision 2 text, which is identical to the BLN COL FSAR, Revision 1 text accepted by the staff, was not changed. Therefore, the staff concludes that this difference is not relevant and that the staff's evaluation of STD SUP 8.3-1 for BLN applies to STD SUP 8.3-3 for VEGP.*

8.3.2.5 *Post Combined License Activities*

There are no post COL activities related to this section.

8.3.2.6 *Conclusion*

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to dc power systems, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the application are documented in NUREG-1793 and its supplements.

In addition the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the relevant NRC regulations, guidance in NUREG-0800, Section 8.3.2, and other NRC regulatory guides and concludes that with closure of the confirmatory item discussed above, the applicant is in compliance with NRC regulations. The staff based its conclusion on the following:

- WLS DEP 8.3-1 is acceptable because the applicant provided sufficient information involving the use of breakers/fuses for regulating transformers for isolation function that is consistent with IEEE-384, endorsed by RG 1.75.

- STD COL 8.3-2 is acceptable because the applicant provided sufficient information involving the inspection, maintenance, and testing of Class 1E batteries, the clearing of ground faults on the Class 1E dc system, and periodic testing of the battery chargers and voltage regulating transformers.

- STD SUP 8.3-1 is acceptable because the applicant made a commitment that there are no site-specific non-Class 1E dc loads connected to the Class 1E dc system.

9.0 AUXILIARY SYSTEMS

The auxiliary systems provide support systems that support the safe shutdown of the plant or the protection of the health and safety of the public. This area covers a wide range of systems including fuel storage and handling, water systems, compressed air, process sampling, drains, heating, ventilation, and air conditioning (HVAC), fire protection, communications, lighting, and emergency diesel generator support systems.

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage (Related to RG 1.206, Section C.III.1, Chapter 9, C.I.9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling," and C.I.9.1.2, "New and Spent Fuel Storage")

The new fuel storage facilities include the fuel assembly storage racks, the concrete storage pit that contains the storage racks, and auxiliary components including the spent fuel handling crane and pit cover. The storage facilities must maintain the new fuel in subcritical arrays during all credible storage conditions. In addition, new fuel must remain subcritical during fuel handling.

Section 9.1 of the William States Lee III Nuclear Station (WLS) combined license (COL) final safety analysis report (FSAR), Revision 11, incorporates by reference, with no departures or supplements, Section 9.1.1, "New Fuel Storage," of Revision 19 of the AP1000 design control document (DCD). The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793, "Final Safety Evaluation Report [FSER] Related to Certification of the AP1000 Standard Design," and its supplements.

9.1.2 Spent Fuel Storage (Related to RG 1.206, Section C.III.1, Chapter 9, C.I.9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling," and C.I.9.1.2, "New and Spent Fuel Storage")

9.1.2.1 *Introduction*

The spent fuel storage facilities include the spent fuel storage racks, the spent fuel storage pool that contains the storage racks, and the associated equipment storage pits. The storage facilities must maintain the spent fuel in subcritical arrays during all credible storage conditions. In addition, spent fuel must remain subcritical during fuel handling.

[1] See Section 1.2.2 for a discussion of the staff's review related to verification of the scope of information to be included in a COL application that references a design certification (DC).

9.1.2.2 *Summary of Application*

Section 9.1 of the WLS COL FSAR, Revision 11, incorporates by reference Section 9.1 of the AP1000 DCD, Revision 19. Section 9.1 of the DCD includes Section 9.1.2.

In addition, in WLS COL FSAR Section 9.1.6, the applicant provided the following:

<u>AP1000 COL Information Item</u>

- STD COL 9.1-7

The applicant provided additional information in standard (STD) COL 9.1-7 to address COL Information Item 9.1-7.

<u>License Condition</u>

- Part 10, License Condition 2, Item 9.1-7

The applicant proposed a license condition related to STD COL 9.1-7 that sets the implementation milestone for the Metamic coupon monitoring program.

- Part 10, License Condition 6

The applicant proposed in WLS Part 10, Revision 4, a license condition to provide a schedule to support the NRC's inspection of operational programs and added the Metamic monitoring program to this list.

9.1.2.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the fuel storage and handling are given in Section 9.1.2 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: [Light-Water Reactor] LWR Edition."

The regulatory basis for acceptance of the COL information and supplementary information items are established in:

- Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic licensing of production and utilization facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criteria (GDC) 4, "Environmental and Dynamic Effects Design Bases"

- GDC 61, "Fuel Storage and Handling and Radioactivity Control"

9.1.2.4 *Technical Evaluation*

The NRC staff reviewed Section 9.1.2 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to spent fuel storage. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this safety evaluation report (SER) provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (Vogtle Electric Generating Plant [VEGP], Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from requests for additional information (RAIs).

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the Bellefonte Nuclear Station (BLN), Units 3 and 4 COL application. Any confirmatory items in the standard content material retain the numbers assigned in the VEGP SER. Confirmatory items that are first identified in this SER section have a WLS designation (e.g., **Confirmatory Item WLS 9.1-1**).

The following portion of this technical evaluation section is reproduced from Section 9.1.2.4 of the VEGP SER:

> *AP1000 COL Information Item*
>
> - *STD COL 9.1-7*
>
> *COL Information Item 9.1-7 states:*
>
> > *The Combined License holder will implement a spent fuel rack Metamic coupon monitoring program when the plant is placed into commercial operation. This program will include tests to monitor bubbling, blistering, cracking, or flaking; and a test to monitor for*

> *corrosion, such as weight loss measurements and or visual examination.*

STD COL 9.1-7 states:

> *A spent fuel rack Metamic coupon monitoring program is to be implemented when the plant is placed into commercial operation. This program includes tests to monitor bubbling, blistering, cracking, or flaking; and a test to monitor for corrosion, such as weight loss measurements and or visual examination.*

The NRC staff reviewed STD COL 9.1-7 related to the Metamic coupon monitoring program included under Section 9.1 of the BLN COL FSAR. No additional details on the Metamic Coupon Monitoring Program are provided in Section 9.1 of the FSAR.

Since the applicant's proposed resolution of COL Information Item 9.1-7 was a restatement of the text of the COL information item from the DCD, the staff required additional information to be able to evaluate the applicant's closure of the item. An additional Request for Additional Information (RAI) response related to AP1000 DCD Section 9.1.2 (ML091120720) proposed a modification to the text of COL Information Item 9.1-7. The modified wording added neutron attenuation and thickness testing to the list of tests to be included in the Metamic monitoring program to be implemented by the COL holder. In RAI 9.1.2-1, the NRC staff requested that the applicant describe in detail the implementation of the aspects of the Metamic coupon monitoring program that are listed in STD COL 9.1-7, as modified by the additional AP1000 RAI response. In response to RAI 9.1.2-1, the applicant proposed modified wording for STD COL 9.1-7 as follows:

> *STD COL 9.1-7*
>
> *A spent fuel rack Metamic coupon monitoring program is to be implemented when the plant is placed into commercial operation. This program includes tests to monitor bubbling, blistering, cracking, or flaking; and a test to monitor for corrosion, such as weight loss measurements and / or visual examination. The program will also include tests to monitor changes in physical properties of the absorber material, including neutron attenuation and thickness measurements.*

This proposed wording matches the proposed revised text for AP1000 COL Information Item 9.1-7. However, the proposed wording is still a restatement of the COL information item and does not contain the level of detail needed by the staff to evaluate the adequacy of the Metamic monitoring program. Therefore, in RAI 9.1.2-2, the staff requested that the applicant describe the methodology and acceptance criteria for the tests listed, provide the corrective action requirements and provide the administrative controls applicable to the program. Additionally, the applicant should confirm the number of coupons and the withdrawal schedule will be the same as recommended in the DCD or provide an alternative. The *staff has identified this as **Open Item 9.1-1** to track resolution of this issue and to ensure that the additional details are included in the BLN COL FSAR.*

Resolution of Standard Content Open Item 9.1-1

To resolve Open Item 9.1-1, the VEGP applicant provided additional information in a letter dated April 23, 2010, which superseded the original response to Open Item 9.1-1 provided in a letter dated December 30, 2009.

With respect to the number of coupons and the withdrawal schedule, the applicant confirmed that the number of coupons and the withdrawal schedule will be the same as stated in AP1000 DCD Revision 19, Section 9.1.2.2.1. The applicant further stated that since AP1000 DCD Section 9.1 is incorporated by reference into the FSAR, no additional FSAR change would be required. The staff finds the applicant's response regarding the number of coupons and withdrawal schedule acceptable, because the applicant has confirmed the number of coupons and schedule will be the same as described in the AP1000 DCD Revision 19.

With respect to methodology and acceptance criteria, corrective actions and administrative controls, the applicant stated that since the Metamic coupon monitoring program has not yet been established, the level of detail requested is not completely available. The applicant further stated, "As stated in FSAR Subsection 9.1.6, a Metamic monitoring program will be implemented when the plant is placed into commercial operation. This program will include methodology to be employed, acceptance criteria, corrective actions and a description of administrative controls based on vendor recommendations and industry operating experience."

The applicant additionally stated that the VEGP COL FSAR will be revised to add the following to the end of the STD COL 9.1-7 discussion:

> *The program will include the methodology and acceptance criteria for the tests listed and provide corrective action requirements based on vendor recommendations and industry operating experience. The program will be implemented through plant procedures.*
>
> *Metamic Monitoring Acceptance Criteria:*
>
> - *Verification of continued presence of the boron is performed by neutron attenuation measurement. A decrease of no more than 5 percent in Boron-10 content, as determined by neutron attenuation, is acceptable. This is equivalent to a requirement for no loss in boron within the accuracy of the measurement.*
> - *Coupons are monitored for unacceptable swelling by measuring coupon thickness. An increase in coupon*

> *thickness at any point of no more than 10 percent of the initial thickness at that point is acceptable.*

Changes in excess of either of the above two acceptance criteria are investigated under the corrective action program and may require early retrieval and measurement of one or more of the remaining coupons to provide validation that the indicated changes are real. If the deviation is determined to be real, an engineering evaluation is performed to identify further testing or any corrective action that may be necessary.

Additional parameters are examined for early indications of the potential onset of Metamic degradation that would suggest a need for further attention and possibly a change in the coupon withdrawal schedule. These include visual inspection for surface pitting, blistering, cracking, corrosion or edge deterioration, or unaccountable weight loss in excess of the measurement accuracy.

The NRC staff concludes that the above information to be added to the VEGP COL FSAR provides the necessary level of detail for the Metamic monitoring program, including the methodology and acceptance criteria for the tests listed, the corrective action requirements, and the administrative controls applicable to the program.

The applicant proposed a markup of the VEGP COL application, Part 10, License Condition 6, adding a line item for the Metamic Monitoring Program. After the addition of this line item, the version of License Condition 6 included in Part 10 of the COL application, Revision 2, would be:

> *The licensee shall develop a schedule that supports planning for and conduct of NRC inspection of the operational program listed in VEGP COL FSAR Table 13.4-201, "Operational Program Required by NRC Regulations." This schedule must be available to the NRC staff no later than 12 months after issuance of the COL. The schedule shall be updated every 6 months until 12 months before scheduled fuel load, and every month thereafter until the operational programs listed in VEGP COL FSAR Table 13.4-201 have been fully implemented or the plant has been placed in commercial service, whichever comes first. This schedule shall address:*
>
> *a. the implementation of site-specific Severe Accident Management Guidance.*
>
> *b. the reactor vessel pressurized thermal shock evaluation at least 18 months prior to initial fuel load.*
>
> *c. the approved preoperational and startup test procedures in accordance with FSAR Section 14.2.3.*

d. the flow accelerated corrosion (FAC) program implementation, including the construction phase activities.

#. the spent fuel rack Metamic coupon monitoring program implementation.

(Where # will be replaced with the next sequential number in the final version of this license condition.)

The inclusion of the Metamic Coupon Monitoring Program in License Condition 6 ensures that the program will be treated as an operational program with respect to providing a schedule to support the NRC's inspection; thus, the applicant must submit and update the schedule for program implementation following the issuance of the COL, in order to support planning of NRC inspections. The staff, therefore, finds the applicant's proposed resolution of ***Open Item 9.1-1*** *acceptable because the applicant will modify proposed License Condition 6 to ensure the appropriate information is available for the staff's review of the details of the Metamic Monitoring Program prior to the start of plant operation.* ***Open Item 9.1-1*** *is, therefore, resolved. Incorporation of the proposed revision to Chapter 9 of the VEGP COL FSAR and to License Condition 6 in the VEGP COL application is being tracked as* ***Confirmatory Item 9.1-1****.*

<u>Resolution of Standard Content Confirmatory Item 9.1-1</u>

Confirmatory Item 9.1-1 is an applicant commitment to revise its FSAR Section 9.1.6 to include a requirement for inclusion of methodology, acceptance criteria and corrective action in the Metamic Coupon Monitoring Program. The staff verified that the VEGP COL FSAR was appropriately revised. As a result, Confirmatory Item 9.1-1 is now closed.

9.1.2.5 *Post Combined License Activities*

For the reasons discussed in the technical evaluation section above, the staff finds the following license condition acceptable:

- License Condition (9-1) – Prior to initial fuel load, the licensee shall implement the spent fuel rack Metamic Coupon Monitoring Program. No later than 12 months after issuance of the COL, the licensee shall submit to the Director of the Office of New Reactors(NRO) a schedule that supports planning for and conduct of NRC inspections of the spent fuel rack Metamic Coupon Monitoring Program. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until the spent fuel rack Metamic Coupon Monitoring Program has been fully implemented.

9.1.2.6 *Conclusion*

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to spent fuel storage, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the

information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the guidelines given in Section 9.1.2 of NUREG-0800. The staff based its conclusion on the following:

- STD COL 9.1-7 is acceptable because the necessary level of detail for the Metamic monitoring program has been provided by the applicant, including the methodology and acceptance criteria for the tests listed, the corrective action requirements, and the administrative controls applicable to the program.

9.1.3 Spent Fuel Pool Cooling System (Related to RG 1.206, Section C.III.1, Chapter 9, C.I.9.1.3, "Spent Fuel Pool Cooling and Cleanup System")

The spent fuel pool cooling system (SFS) is designed to remove decay heat, which is generated by stored fuel assemblies from the water in the spent fuel pool (SFP). The safety-related portion of the SFS credits the water inventory in the pool and safety-related makeup water to remove the decay heat. The nonsafety-related portion of the system is an active system during normal operations that pumps the high temperature water from within the fuel pool through a heat exchanger, and then returns the water to the pool. The SFS heat exchangers are cooled by the component cooling water system (CCS). A secondary function of the SFS is clarification and purification of the refueling water and the SFP.

Section 9.1.3 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures, Section 9.1.3, "Spent Fuel Pool Cooling System," of Revision 19 of the AP1000 DCD. To address recommendations of the Fukushima Near-Term Task Force described in SECY-12-0025, "Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami," (ADAMS Accession No. ML12039A103), specifically Recommendation 7.1 related to reliable spent fuel pool instrumentation, the applicant provided additional information, including supplemental information (WLS SUP 9.1-1) in Section 9.1.3.7 of the FSAR and a proposed license condition. Section 20.2 of this SER presents the staff's evaluation of the application with respect to NTTF Recommendation 7.1.

The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staffs review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

9.1.4 Light Load Handling System (Related to RG 1.206, Section C.III.1, Chapter 9, C.I.9.1.4, "Light Load Handling System (Related to Refueling)")

9.1.4.1 *Introduction*

The light-load handling system (LLHS) consists of the equipment and structures needed for the refueling operation. This equipment is comprised of fuel assemblies, core component and reactor component hoisting equipment, handling equipment, and a dual basket fuel transfer system. The structures associated with the fuel handling equipment are the refueling cavity, the transfer canal, the fuel transfer tube, the SFP, the cask loading area, the new fuel storage area, and the new fuel receiving and inspection area.

9.1.4.2 *Summary of Application*

Section 9.1 of the WLS COL FSAR, Revision 11, incorporates by reference Section 9.1 of the AP1000 DCD, Revision 19. Section 9.1 of the DCD includes Section 9.1.4.

In addition, in WLS COL FSAR Section 9.1.4, the applicant provided the following:

AP1000 COL Information Items

- STD COL 9.1-5

The applicant provided additional information in STD COL 9.1-5 to address COL Information Item 9.1-5 (COL Action Item 9.1.6-5).

- STD COL 9.1-6

The applicant provided additional information in STD COL 9.1-6 to address COL Information Item 9.1-6 (COL Action Item 9.1.6-6).

9.1.4.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the LLHS are given in Section 9.1.4 of NUREG-0800.

The regulatory basis for acceptance of the COL information items are established in:

- GDC 61

- American National Standards Institute/American Nuclear Society (ANSI/ANS) 57.1-1992, “Design Requirements for LWR Fuel Handling Systems”

9.1.4.4 *Technical Evaluation*

The NRC staff reviewed Section 9.1.4 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff’s review confirmed that the information in the application and incorporated by reference addresses the required information relating to the LLHS. The results of the NRC staff’s evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the BLN Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from Section 9.1.4.4 of the VEGP SER:

> *<u>AP1000 COL Information Items</u>*
>
> - *STD COL 9.1-5*
>
> *COL Information Item 9.1-5 states:*
>
> > *The Combined License applicant is responsible for a program for inservice inspection of the light load handling system as specified in subsection 9.1.4.4 and the overhead heavy load handling system in accordance with ANSI B30.2, ANSI B30.9, ANSI N14.6, and ASME [American Society of Mechanical Engineers] NOG-1 as specified in subsection 9.1.5.4.*
>
> *The commitment was also captured as COL Action Item 9.1.6-5 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:*
>
> > *The Combined License applicant is responsible for a program for inservice inspection of the light load handling system as specified in DCD Tier 2, Section 9.1.4.4 and the overhead heavy load handling system in accordance with ANSI B30.2, ANSI B30.9, ANSI N14.6, and ASME NOG-1 as specified in DCD Tier 2, Section 9.1.5.4.*

STD COL 9.1-5 states:

> *The above requirements are part of the plant inspection program for the light load handling system, which is implemented through procedures. In addition to the above inspections, the procedures reflect the manufacturers' recommendations for inspection.*

The staff reviewed STD COL 9.1-5, which addresses COL Information Item 9.1-5 on the inservice inspection (ISI) program for the LLHS. The applicant stated that the inspection program for the LLHS is implemented through procedures and reflect the manufacturer's recommendations. RAI 9.1.4-1 requested that the applicant provide a copy of the procedures for verification by the staff or provide the schedule in relation to fuel loading for issuance of the procedures.

The applicant stated in its response to RAI 9.1.4-1, that an inspection and testing program will be developed to address the LLHS. Procedures defining the program will address the testing and inspection requirements outlined in Section 9.1.4.4, "Inspection and Test Requirements," of the AP1000 DCD and the procedures will include applicable manufacturer's recommendations and industry standards. The applicant stated that procedure development is tracked by the overall plant construction and test schedule. The applicant further stated that details of the implementation milestones for development of procedures are not currently available and are not expected to be available until a detailed construction schedule has been developed. When it becomes available, scheduling information will be provided to the NRC as necessary to support timely completion of NRC inspection and audit functions.

Although the response to RAI 9.1.4-1 states that the plant inspection program schedule information will be provided when available, BLN COL FSAR Table 1.8-202 lists STD COL 9.1-5 as having been completed by the applicant. The staff notes that STD COL 9.1-5 has not been fully addressed. The applicant is asked to revise BLN COL FSAR Table 1.8-202 to commit in the BLN COL FSAR to implementing the plant inspection program for the LLHS before receipt of fuel. This is ***Open Item 9.1-2****.*

- *STD COL 9.1-6*

COL Information Item 9.1-6 states:

> *The Combined License applicant is responsible to ensure an operating radiation monitor is mounted on any crane or fuel handling machine when it is handling fuel.*

The commitment was also captured as COL Action Item 9.1.6-6 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:

> *The COL applicant/holder will ensure that an operating radiation monitor is mounted on any crane or fuel handling machine when it is handling fuel.*

STD COL 9.1-6 states:

> *Plant procedures require that an operating radiation monitor is mounted on any machine when it is handling fuel. Refer to DCD Subsection 11.5.6.4, "Fuel Handling Area Criticality Monitors," for a discussion of augmented radiation monitoring during fuel handling operations.*

The NRC staff reviewed STD COL 9.1-6, which addresses COL Information Item 9.1-6 related to radiation monitoring included under Section 9.1.4 of the BLN COL FSAR. The proposed mounting of an operating radiation monitor on any crane or fuel handling machine during fuel handling is included under Section 9.1.4.3.8 of the BLN COL FSAR. The applicant committed to develop plant procedures that will specify that an operating radiation monitor be mounted on any fuel handling machine when it is handling fuel. DCD Section 11.5.6.4 specifies the need to augment area radiation monitoring during fuel handling operations by a portable radiation monitor on the machine handling fuel. The staff finds that with the addition of the portable radiation monitor to any fuel handling machine when it is handling fuel, the BLN COL FSAR meets the applicable requirements of 10 CFR Part 50, Appendix A, GDC 61 for the prevention of unacceptable radiation exposure.

The staff finds that the applicant has adequately addressed COL Information Item 9.1-6 which would ensure that an operating portable radiation monitor is mounted on any fuel handling machine in the LLHS when it is handling fuel.

<u>Resolution of Standard Content Open Item 9.1-2</u>

*To resolve **Open Item 9.1-2**, in a letter dated December 30, 2009, the applicant proposed a change to VEGP COL FSAR Section 9.1.4.4 in response to this open item instead of a revision to Table 1.8-202. The applicant proposed a revision to FSAR Section 9.1.4.4 to clarify that the LLHS, including system inspections, is implemented prior to receipt of fuel onsite. The staff finds this acceptable since the commitment provided will ensure that these procedures will be in place prior to fuel movement. Therefore, **Open Item 9.1-2** is resolved. Incorporation of the proposed revision in the VEGP COL FSAR is being tracked as **Confirmatory Item 9.1-2**.*

<u>Resolution of Standard Content Confirmatory Item 9.1-2</u>

Confirmatory Item 9.1-2 is an applicant commitment to revise its FSAR Section 9.1.4.4 to include an inspection of the LLHS prior to receipt of fuel. The *staff verified that the VEGP COL FSAR was appropriately revised. As a result, Confirmatory Item 9.1-2 is now closed.*

Correction of Error in the Standard Content Evaluation Text

The NRC staff identified an error in the text reproduced above from Section 9.1.4.4 of the BLN SER that requires correction. The BLN SER provides quoted material for COL Action Item 9.1.6-5, citing Appendix F of NUREG-1793 as the source. The source of the quoted material for COL Action Item 9.1.6-5 is in fact from Chapter 9 (Section 9.1.6) of NUREG-1793.

9.1.4.5 *Post Combined License Activities*

For the reasons discussed in the technical evaluation above, the following FSAR commitment is identified as the responsibility of the licensee:

- The light-load handling program, including system inspections, will be implemented prior to receipt of fuel onsite.

9.1.4.6 *Conclusion*

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to the LLHS and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes, that the relevant information presented in the WLS COL FSAR is acceptable and meets the guidelines given in Section 9.1.4 of NUREG-0800. The staff based its conclusion on the following:

- STD COL 9.1-5 is acceptable because the staff finds that the relevant information in the WLS COL FSAR provided clarification that ISI of the LLHS is part of the plant inspection program for the LLHS, which is implemented through procedures.

- STD COL 9.1-6 is acceptable because the staff finds that the relevant information in the WLS COL FSAR meets the applicable requirements of 10 CFR Part 50, Appendix A, GDC 61.

9.1.5 Overhead Heavy Load Handling Systems (Related to RG 1.206, Section C.III.1, Chapter 9, C.I.9.1.5, "Overhead Load Handling System")

9.1.5.1 *Introduction*

The overhead heavy-load handling system (OHLHS) is used to lift loads whose weight is greater than the combined weight of a single spent fuel assembly and its handling device. The principal equipment is the containment polar crane, equipment hatch hoist, maintenance hatch hoist, and the cask handling crane. The OHLHS is designed to ensure that inadvertent operations or equipment malfunctions, separately or in combination, will not cause a release of radioactivity, a criticality accident, an inability to cool fuel within the reactor vessel or SFP, or prevent safe shutdown of the reactor.

9.1.5.2 *Summary of Application*

Section 9.1 of the WLS COL FSAR, Revision 11, incorporates by reference Section 9.1 of the AP1000 DCD, Revision 19. Section 9.1 of the AP1000 DCD includes Section 9.1.5.

In addition, in WLS COL FSAR Section 9.1.5, the applicant provided the following:

Supplemental Information

- STD SUP 9.1-1

The applicant provided supplemental (SUP) information in Section 9.1.5.3, "Safety Evaluation," describing heavy-load lifts outside those already described in the AP1000 DCD.

- STD SUP 9.1-2

The applicant provided supplemental information in Section 9.1.5, "Overhead Heavy Load Handling Systems," describing key elements of the heavy-loads handling program and a quality assurance (QA) program.

- STD SUP 9.1-3

The applicant provided supplemental information in Section 9.1.5.5, "Load Handling Procedures," describing load handling operations for heavy loads in the vicinity of irradiated fuel and safe shutdown equipment.

AP1000 COL Information Items

- STD COL 9.1-5

The applicant provided additional information in STD COL 9.1-5 to address COL Information Item 9.1-5 (COL Action Item 9.1.6-5).

- STD COL 9.1-6

The applicant provided additional information in STD COL 9.1-6 to address COL Information Item 9.1-6 (COL Action Item 9.1.6-6).

9.1.5.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the OHLHS are given in Section 9.1.5 of NUREG-0800.

The regulatory basis for acceptance of STD SUP 9.1-1, STD SUP 9.1-2 and STD SUP 9.1-3 addressing planned heavy-load lift programs include the following:

- GDC 4
- GDC 61
- NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants"

The regulatory basis for acceptance of STD COL 9.1-5, addressing the ISI program for the OHLHS is based on GDC 4 and the guidelines of NUREG-0612, which references ANSI B30.2, "Overhead and Gantry Cranes"; ANSI N14.6, "Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More," ASME NOG-1, "Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)"; and ANSI B30.9, "Slings."

The regulatory basis for acceptance of STD COL 9.1-6, addressing operating radiation monitor on any crane handling fuel is based on the requirements of GDC 61.

9.1.5.4 *Technical Evaluation*

The NRC staff reviewed Section 9.1.5 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to OHLHS. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the BLN Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from Section 9.1.5.4 of the VEGP SER:

Supplemental Information

- *STD SUP 9.1-1, STD SUP 9.1-2, and STD SUP 9.1-3*

The staff reviewed the information provided by the applicant for STD SUP 9.1-1. The applicant stated that it did not provide an itemized list of heavy load lifts outside the scope of heavy loads described in the AP1000 DCD because no such heavy load lifts are currently planned. The applicant provided a general description for addressing heavy load movements outside the planned scope if needed in the future. However, the applicant did not address all the program elements and detail listed in NUREG-0612 Section 5.1.1 and NUREG-0800 Section 9.1.5, nor did it provide a schedule for implementation of the heavy load handling program. A heavy load handling program that meets the guidelines of NUREG-0612 and NUREG-0800 Section 9.1.5, needs to be in place at a time before there is a possibility that a load drop could cause a release of radioactivity, a criticality accident, inability to cool fuel within the reactor vessel or spent fuel pool, or prevent safe shutdown of the reactor. The staff asked the applicant in RAI 9.1.5-1 to provide the program elements specified in NUREG-0612 Section 5.1.1 and NUREG-0800 Section 9.1.5, and a schedule for implementation.

In BLN COL FSAR, Revision 1, the applicant provided the missing and necessary information specified in NUREG-0612 Section 5.1.1 and NUREG-0800 Section 9.1.5. The applicant provided a description of the key elements of the heavy load handling system program in BLN COL FSAR Section 9.1.5. The key elements are: 1) Listing of heavy loads; 2) Listing of handling equipment; 3) Safe load paths definition, location and evaluation; 4) Procedures and maintenance manuals; 5) Inspection and testing; 6) Personnel qualification and training; and 7) Quality Assurance (QA) program to monitor and implement the heavy loads program. Also, the BLN COL FSAR, Revision 1 Section 9.1.5 describes the heavy loads handling system procedures. Because Section 9.1.5 of the BLN COL FSAR includes the key elements identified in NUREG-0612, the staff finds the aspects of RAI 9.1.5-1 regarding the key elements of the heavy loads program resolved. Therefore, the staff finds the applicant meets the applicable requirements of 10 CFR Part 50, Appendix A, GDC 4.

In its response to RAI 9.1.5-1, the applicant stated that details of the implementation milestones for the development of heavy load handling procedures and related engineering documents are not currently available, nor are the implementation milestones expected to be available until after a detailed construction schedule has been developed. The applicant stated that appropriate scheduling information will be provided, when available, to the NRC as necessary to support timely completion of inspection and audit functions. The applicant did not provide any schedule for when the heavy load handling program will be completed for the implementation of an approved heavy load handling *program (including OHLHS procedures). The applicant is asked to revise BLN COL FSAR Table 1.8-202 to commit in the BLN COL FSAR to implementing the heavy load handling program before receipt of fuel. This is **Open Item 9.1-3**.*

AP1000 COL Information Items

- *STD COL 9.1-5*

The applicant provided additional information in STD COL 9.1-5 to address COL Information Item 9.1-5. COL Information Item 9.1-5 states:

> *The Combined License applicant is responsible for a program for inservice inspection of the light load handling system as specified in subsection 9.1.4.4 and the overhead heavy load handling system in accordance with ANSI B30.2, ANSI B30.9, ANSI N14.6, and ASME NOG-1 as specified in subsection 9.1.5.4.*

The commitment was also captured as COL Action Item 9.1.6-5 in Chapter 9 of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:

> *The Combined License applicant is responsible for a program for inservice inspection of the light load handling system as specified in DCD Tier 2, Section 9.1.4.4 and the overhead heavy load handling system in accordance with ANSI B30.2, ANSI B30.9, ANSI N14.6, and ASME NOG-1 as specified in DCD Tier 2, Section 9.1.5.4.*

The staff reviewed STD COL 9.1-5, which addresses COL Information Item 9.1-5 on the plant inspection program for the OHLHS. The applicant stated that the inspection program for the OHLHS is implemented through procedures and reflect the manufacturer's recommendations and the recommendations of NUREG-0612. The staff asked the applicant in RAI 9.1.5-2 to provide a copy of the procedures for verification by the staff.

In its response to RAI 9.1.5-2, the applicant stated that a plant inspection program for the OHLHS will be created using the manufacturer's recommendations and will meet the requirements outlined in applicable industry standards. The staff confirmed that BLN COL FSAR Section 9.1.5.4 was revised to provide additional information related to the description of implementing procedures. On the basis of its review, the staff finds the applicant adequately addressed that the OHLHS plant inspection program procedures will follow the equipment manufacturer's recommendations and will meet the requirements in applicable industry standards. With the addition to BLN COL FSAR Section 9.1.5.4 of a descriptive list of the minimum elements required to be addressed in the overhead heavy load handling equipment plant inspection program procedures, in addition to the other guidelines specified in Section 9.1.5 of NUREG-0800, the staff finds the applicant meets the applicable requirements of 10 CFR Part 50, Appendix A, GDC 4.

In the RAI response, the applicant stated that the schedule for issuing the procedures that implement the plant inspection program for the OHLHS are not yet available. The applicant also stated that implementation milestones are not expected to be available until after a detailed construction schedule has been developed, but will be provided to the NRC when available to support timely completion of inspection and audit functions. Although the response to RAI 9.1.5-2 states that the plant inspection program schedule information will be provided when available, BLN COL FSAR Table 1.8-202 lists STD COL 9.1-5 as having been completed by the applicant. The staff notes that STD COL 9.1-5 has not been fully addressed. The applicant is asked to revise BLN COL FSAR Table 1.8-202 to commit in the BLN COL FSAR to implementing the plant inspection program for the OHLHS before receipt of fuel. This is ***Open Item 9.1-4****.*

- *STD COL 9.1-6*

The applicant provided additional information in STD COL 9.1-6 to address COL Information Item 9.1-6. COL Information Item 9.1-6 states:

> *The Combined License applicant is responsible to ensure an operating radiation monitor is mounted on any crane or fuel handling machine when it is handling fuel.*

The commitment was also captured as COL Action Item 9.1.6-6 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:

> *The COL applicant/holder will ensure that an operating radiation monitor is mounted on any crane or fuel handling machine when it is handling fuel.*

The NRC staff reviewed STD COL 9.1-6, which addresses COL Information Item 9.1-6 related to radiation monitoring included under Section 9.1.5 of the BLN COL FSAR. The proposed mounting of an operating radiation monitor on any crane or fuel handling machine during fuel handling is included under Section 9.1.5.3 of the BLN COL FSAR. The applicant committed to develop plant procedures that will specify that an operating radiation monitor be mounted on any fuel handling machine when it is handling fuel. DCD Section 11.5.6.4 specifies the need to augment area radiation monitoring during fuel handling operations by a portable radiation monitor on the machine handling fuel. The staff finds that with the addition of the portable radiation monitor to any fuel handling machine when it is handling fuel, the BLN COL FSAR meets the applicable requirements of 10 CFR Part 50, Appendix A, GDC 61 for the prevention of unacceptable radiation exposure.

The staff finds that the applicant has adequately addressed COL Information Item 9.1-6 which would ensure that an operating portable radiation monitor is mounted on any crane when it is handling fuel.

<u>Resolution of Standard Content Open Items 9.1-3 and 9.1-4</u>

*The VEGP applicant responded to **Open Items 9.1-3 and 9.1-4** in a letter dated December 30, 2009. The letter proposed a change to VEGP COL FSAR Section 9.1.5.4 in response to these open items instead of revising Table 1.8-202. The applicant proposed a revision to FSAR Section 9.1.5.4 to clarify that the OHLHS, including system inspections, will be implemented prior to receipt of fuel onsite. The staff finds this acceptable since the commitment provided will ensure that the procedures will be in place and the plant inspection program will be implemented for the OHLHS prior to fuel movement. Therefore, **Open Items 9.1-3 and 9.1-4** are resolved. Incorporation of the proposed revision in the FSAR is being tracked as **Confirmatory Item 9.1-3**.*

<u>Resolution of Standard Content Confirmatory Item 9.1-3</u>

Confirmatory Item 9.1-3 is an applicant commitment to revise its FSAR Section 9.1.5.4 to include an inspection of the OHLHS prior to receipt of fuel. The staff verified that the VEGP COL FSAR was appropriately revised. As a result, Confirmatory Item 9.1-3 is now closed.

9.1.5.5 *Post Combined License Activities*

For the reasons discussed in the technical evaluation above, the following FSAR commitment is identified as the responsibility of the licensee:

- The overhead heavy-load handling program, including system inspections, will be implemented prior to receipt of fuel onsite.

9.1.5.6 *Conclusion*

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to OHLHS and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes, that the relevant information presented in the WLS COL FSAR is acceptable and meets the guidelines given in Section 9.1.5 of NUREG-0800. The staff based its conclusion on the following:

- STD SUP 9.1-1, STD SUP 9.1-2, and STD SUP 9.1-3 are acceptable because the staff finds that the applicant provided supplemental information in accordance with NUREG-0612, NUREG-0800 Section 9.1.5, and Regulatory Guide (RG) 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," Section C.I.9.1.5 guidance to describe the program and schedule for the implementation of the program governing heavy-load handling.

- STD COL 9.1-5 is acceptable because the staff finds that the relevant information in the WLS COL FSAR provided clarification that ISI of the OHLHS is part of the plant inspection program for the OHLHS, which is implemented through procedures.

- STD COL 9.1-6 is acceptable because the staff finds that the relevant information in the WLS COL FSAR meets the applicable requirements of 10 CFR Part 50, Appendix A, GDC 61.

9.2 Water Systems

9.2.1 Service Water System (Related to RG 1.206, Section C.III.1, Chapter 9, C.I.9.2.1, "Station Service Water System (Open, Raw Water Cooling Systems)")

9.2.1.1 *Introduction*

The service water system (SWS) is a nonsafety-related system that supplies cooling water to remove heat from the nonsafety-related CCS heat exchangers in the turbine building. The SWS is arranged into two trains of components and piping. Each train includes one service water pump, one strainer, and a cooling tower cell as its heat sink. The heat sink for both trains is provided by a single cooling tower with two cells and a divided basin. Each train is capable of providing 100-percent of the required SWS flow for normal full power operation.

9.2.1.2 *Summary of Application*

As provided in Section 9.2 of the WLS COL FSAR, Revision 11, Section 9.2.1, "Service Water System," is incorporated by reference. However, the AP1000 standard design is for a single unit and the following item concerning multiple units is a site-specific consideration that needs to be addressed:

Potential SWS Cooling Tower Interactions:

> The AP1000 DCD was approved for use as a single unit. The applicant proposes to install two units, and potential interactions between the two SWS cooling towers were not considered in the original design and need to be addressed to assure adequate cooling capability for each unit.

In addition, in WLS COL FSAR Section 9.2.1, the applicant provided the following:

Supplemental Information

- WLS SUP 9.2-2

The applicant provided supplemental information in Section 9.2.1.2.2, "Component Description," by adding additional text to address the SWS cooling tower potential interactions.

9.2.1.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

Although the SWS (including heat sink) is not safety-related, it supports the normal (defense-in-depth) capability of removing reactor and spent fuel decay heat, it is part of the first line of defense for reducing challenges to passive safety systems in the event of transients and plant upsets, and its cooling function is important for reducing shutdown risk when the reactor coolant system (RCS) is open (e.g., during mid-loop conditions). The risk importance of the SWS makes it subject to regulatory treatment of nonsafety-related systems (RTNSS) in accordance with the Commission's policy for passive reactor plant designs in SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs."

The NRC staff's evaluation of the SWS focuses primarily on confirming that the SWS is capable of performing its defense-in-depth and RTNSS functions; that it will not adversely impact safety-related structures, systems and components (SSCs); and that inspections, tests, analyses, and acceptance criteria (ITAAC), test program specifications, and RTNSS availability controls for the SWS are appropriate.

The regulatory basis for acceptance of WLS SUP 9.2-2, addressing the SWS cooling tower is the acceptance criteria in Sections 9.2.1 and 9.2.5 of NUREG-0800.

9.2.1.4 *Technical Evaluation*

The NRC staff reviewed Section 9.2.1 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the SWS. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the WLS COL FSAR:

Supplemental Information

- WLS SUP 9.2-2

The applicant provided supplemental information in WLS COL FSAR Section 9.2.1.2.2 by adding additional text to address the SWS cooling tower potential interactions.

Potential SWS Cooling Tower Interactions:

The cooling capability of the SWS cooling towers for the WLS units can be adversely affected by interactions that exist between the station's cooling towers. Adverse interactions can occur due to localized atmospheric influences caused by siting and relative proximity considerations. Because this is not a factor for single cooling towers, it is not addressed by the AP1000 DCD. Therefore, the staff generated **RAI Letter #008 Question 9.2.1-1** and **RAI Letter #093 Question 9.2.1-8** requesting additional information for the applicant to address potential adverse interactions between the cooling towers for the two units and to describe any additional design provisions that are necessary, as appropriate.

The applicant provided responses for these two questions in two letters, dated September 26, 2008, and March 14, 2011. In its responses, the applicant stated that the effects on cooling capacity of the SWS mechanical draft cooling towers from interactions with the circulating water system (CWS) mechanical draft cooling towers at the WLS site have been considered.

The WLS has three Unit 1 CWS cooling towers, clustered in a triangular configuration, and located west-southwest of the Units 1 and 2 SWS cooling towers. A cluster of three Unit 2 CWS cooling towers is located east-northeast of the Units 1 and 2 SWS cooling towers. The CWS cooling towers are mechanical induced draft towers and their plumes are directed upward by their fans and the buoyant effect of warm air. Because the three CWS cooling towers of each unit are in close proximity to each other, the individual plumes combine to form a single plume for each unit. Only 50 percent of the heat removal capacity of the SWS cooling tower is required to maintain the required tower function, thus providing substantial margin to accommodate any potential for adverse effects on tower performance due to an interference condition. When an SWS cooling tower of one unit is operating at the highest heat loads, the CWS cooling tower of that unit would be operating at a much reduced heat load. Therefore, the potential for adverse impacts is confined to plume interactions between the CWS cooling towers of one unit and the SWS cooling tower of the adjacent unit. Interaction between the Unit 2 CWS cooling towers and the Unit 1 SWS cooling tower would require a wind direction from the east-northeast, which is not a prevailing wind direction for the site. Winds from the east-northeast direction occur at a yearly frequency of less than 6 percent. Since the natural tendency of the plume would be to rise and disperse at higher elevations, this wind direction would also have to coincide with meteorological conditions that would maintain the Unit 2 CWS cooling towers plume at ground level. Finally, this ground level plume from the Unit 2 CWS cooling towers would have to travel a distance of approximately 365 meters (1200 feet) and circumvent the Unit 2 turbine building to interact with the Unit 1 SWS cooling tower. Even with the assumption of a worst-case plume condition, independent of CWS cooling tower fan speed or the number of fans operating, there is a minimal potential for adverse impacts to the Unit 1 SWS cooling tower. Interaction between the Unit 1 CWS cooling tower and the Unit 2 SWS cooling towers has a lower probability. This plume interaction requires wind direction from the west-southwest, which occurs at a yearly frequency of less than 5 percent. In addition to the favorable wind and meteorological conditions, the Unit 1 CWS cooling towers plume would also be required to travel a distance of approximately 640 meters (2100 feet) and circumvent both the Units 1 and 2 turbine building structures to interact with the Unit 2 SWS cooling tower.

In addition, there is a minimal probability that an SWS cooling tower plume could travel to the vicinity of a SWS tower on an adjacent unit. Interfering structures in the path of the plume would provide ample opportunity for plume dispersion, greatly minimizing any adverse effect on tower performance. Due to the power block separation requirements for a two-unit facility (approximately 243 meters (800 feet) of separation between SWS cooling tower), the SWS cooling tower is in much closer proximity to the buildings and structures within its own unit than to those located in an adjacent unit. There are no site-specific conditions that could result in adverse impacts from air restriction. During conditions where the SWS cooling tower is subject to RTNSS requirements , the tower is only operating at 50 percent of its operational heat load, leaving a substantial margin available to accommodate site-specific adverse interactions, if they exist.

The maximum normal wet bulb temperature for the site is over 4 degrees less than the wet bulb temperature used to size the tower, creating additional margin.

Based on the information that was provided in the applicant's responses, the staff considers the applicant's resolution of this issue to be acceptable since all of the mechanical draft cooling tower interactions at WLS have been considered and that there will be minimal cooling tower interaction effects and the cooling tower interactions will not adversely affect the cooling capacity of the SWS since the cooling towers have at least 243 meters (800 feet) of building separation and the large structure, the turbine building, being placed between the cooling towers. Therefore, there is reasonable reassurance to conclude that any postulated site-specific performance degradation resulting from an interaction with a second unit would be minimal and would be readily accommodated by the design margins available to support RTNSS capability. **RAI Letter #008 Question 9.2.1-1** and **RAI Letter #093 Question 9.2.1-008** are resolved.

9.2.1.5 *Post Combined License Activities*

There are no post-COL activities related to this section.

9.2.1.6 *Conclusion*

Information that the applicant incorporated by reference was previously reviewed and approved by the NRC as documented in NUREG-1793, including Supplement 2. Consequently, this information was not included within the scope of this evaluation. Therefore, the staff's evaluation of the WLS COL application was limited to plant-specific considerations that were not included within the scope of the generic AP1000 DCD approval.

The staff evaluated the potential for adverse interactions between the SWS cooling towers and CWS cooling towers for the WLS units. Based on the results of this evaluation, the staff determined that the applicant's RAI response related to CWS mechanical draft cooling tower and SWS cooling tower interactions has been adequately resolved. Therefore, the staff concludes that the WLS SWS, as described in Section 9.2.1 of the WLS COL FSAR is acceptable.

9.2.2 Component Cooling Water System (Related to RG 1.206, Section C.III.1, Chapter 9, C.I.9.2.2, "Cooling System for Reactor Auxiliaries (Closed Cooling Water Systems")

9.2.2.1 *Introduction*

The CCS provides a closed loop of cooling water for reactor system components, reactor shutdown equipment, ventilation equipment, and components of the emergency core cooling system.

Section 9.2 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 9.2.2, "Component Cooling Water System (CCS)," of Revision 19 of the AP 1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

9.2.3 Demineralized Water Treatment System

The demineralized water treatment system provides the required supply of reactor coolant purity water to the demineralized water transfer and storage system. This system does not perform any safety-related function or accident mitigation, and its failure would not reduce the safety of the plant.

Section 9.2 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 9.2.3, "Demineralized Water Treatment System," of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

9.2.4 Demineralized Water Transfer and Storage System

The demineralized water transfer and storage system supplies demineralized water to fill the condensate storage tank and to the plant systems that demand a demineralized water supply. This system has no safety-related function other than containment isolation, and its failure does not affect the ability of safety-related systems to perform their safety-related functions.

Section 9.2 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 9.2.4, "Demineralized Water Transfer and Storage System," of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

9.2.5 Potable Water System (Related to RG 1.206, Section C.III.1, Chapter 9, C.I.9.2.4, "Potable and Sanitary Water Systems")

9.2.5.1 *Introduction*

The potable water system (PWS) supplies clean water from the raw water system (RWS) for domestic use and human consumption. The PWS has no safety-related functions other than to prevent in-leakage into the main control room envelope during main control room emergency habitability system (VES) operation. A loop seal in the safety-related PWS piping that penetrates the main control room envelope boundary prevents unfiltered air in-leakage into the main control room envelope

9.2.5.2 *Summary of Application*

Section 9.2 of the WLS COL FSAR, Revision 11, incorporates by reference Section 9.2 of the AP1000 DCD, Revision 19. Section 9.2 of the AP1000 DCD includes Section 9.2.5, "Potable

Water System," which addresses Section 9.2.4, "Potable and Sanitary Water Systems," of NUREG-0800.

In addition, in WLS COL FSAR Section 9.2.5, the applicant provided the following:

AP1000 COL Information Item

- WLS COL 9.2-1

The applicant provided additional information in WLS COL 9.2-1 to address COL Information Item 9.2-1 in WLS COL FSAR Sections 9.2.5.2.1, "General Description," 9.2.5.3, "System Operation, and 9.2.12.1,"Potable Water, by providing information concerning the source of water for the PWS.

9.2.5.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the PWS are given in Section 9.2.4 of NUREG-0800.

The regulatory basis for the review of the COL information item is established in 10 CFR Part 50, Appendix A, GDC 60, "Control of Releases of Radioactive Materials to the Environment."

9.2.5.4 *Technical Evaluation*

The NRC staff reviewed Section 9.2.5 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the PWS. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the WLS COL FSAR:

AP1000 COL Information Item

- WLS COL 9.2-1

The applicant provided additional information in WLS COL 9.2-1 to resolve COL Information Item 9.2-1. COL Information Item 9.2-1 states:

> The Combined License applicant will address the components of the potable water system outside of the power block, including supply source required to meet design pressure and capacity requirements, specific chemical selected for use as a biocide, and any storage requirements deemed necessary. A biocide such as sodium hypochlorite is recommended. Toxic gases such as chlorine are

not recommended. The impact of toxic gases on the main control room habitability is addressed in Section 6.4.

The NRC staff reviewed the resolution to COL Information Item 9.2-1 on the source of water for the potable water system included under Sections 9.2.5.2.1, 9.2.5.3, and 9.2.12.1 of the Lee COL FSAR. In these sections the applicant proposes to use the Draytonville Water District as the source of potable water. The water supply meets DCD Section 9.2.5 regarding pressure, capacity, and quality requirements. Because the applicant is using a municipal water supply, no biocide is necessary; therefore there is no impact of toxic gases on main control room habitability from this system. The staff finds this an acceptable resolution of COL Information Item 9.2-1 because the pressure, capacity, and quality requirements from the DCD are met. In DCD Revision 19, Westinghouse states that no interconnections exist between the PWS and any potentially radioactive system or any system using water for purposes other than domestic water service. The site specific information provided in WLS COL 9.2-1 is outside the power block and not potentially contaminated by radioactive water. Because no interconnections exist between the PWS and any potentially radioactive system, the staff finds that GDC 60 is satisfied, with respect to preventing contamination by radioactive water.

9.2.5.5 *Post Combined License Activities*

There are no post-COL activities related to this section.

9.2.5.6 *Conclusion*

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to PWS, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the guidance in Section 9.2.4 of NUREG-0800. The staff based its conclusion on the following:

- WLS COL 9.2-1 is acceptable because the applicant has provided sufficient information on the source of water for the PWS to satisfy GDC 60, with respect to preventing contamination by radioactive water.

9.2.6 Sanitary Drains (Related to RG 1.206, Section C.III.1, Chapter 9, C.I.9.2.4, "Potable and Sanitary Water Systems")

9.2.6.1 *Introduction*

The sanitary drain system collects sanitary wastes from plant restrooms and locker room facilities. The sanitary drainage system has no safety-related function other than main control room envelope isolation. Redundant safety-related isolation valves are provided in the vent line penetrating the main control room. Therefore, there are no single active failures that would prevent isolation of the main control room envelope. The system design ensures that there is no possibility for radioactive contamination of the sanitary drains.

9.2.6.2 *Summary of Application*

Section 9.2 of the WLS COL FSAR, Revision 11, incorporates by reference Section 9.2 of the AP1000 DCD, Revision 19. Section 9.2 of the AP1000 DCD includes Section 9.2.6, "Sanitary Drains," which addresses Section 9.2.4, "Potable and Sanitary Water Systems," of NUREG-0800.

In addition, in WLS COL FSAR Section 9.2.6, the applicant provided the following:

Departures

- WLS DEP 6.4-1

The applicant provided additional information in Section 9.2.6 of the WLS COL FSAR about WLS DEP 6.4-1 related to design changes affecting habitability of the main control room and changes to the calculated doses to control room operators. This information, as well as related WLS DEP 6.4-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.2 of this SER.

Supplemental Information

- WLS SUP 9.2-1

The applicant provided supplemental information by adding text to the end of Section 9.2.6.2.1, "General Description," to state that the sanitary drainage systems (SDS) collects sanitary waste from plant restrooms and locker room facilities in the turbine building, auxiliary building, and annex building, and carries this waste off-site to Gaffney Board of Public works treatment plant where it is processed.

9.2.6.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for WLS SUP 9.2-1 are given in Section 9.2.4 of NUREG-0800.

The regulatory basis for acceptance of the supplementary information is established in:

- GDC 60, as it relates to sanitary drains

9.2.6.4 *Technical Evaluation*

The NRC staff reviewed Section 9.2.6 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to sanitary drains. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the WLS COL FSAR:

Supplemental Information

- WLS SUP 9.2-1

The NRC staff reviewed the location of the waste treatment plant included under Section 9.2.6.2.1 of the WLS COL FSAR. In Section 9.2.6.2.1 of the WLS COL FSAR, the applicant proposes the Gaffney Board of Public Works sewage treatment plant for the treatment of sanitary waste which is located Off-site. The AP1000 DCD states that there are no interconnections between the sanitary drainage system and systems having the potential for containing radioactive material, and the sanitary drainage system does not service facilities in radiologically controlled areas. Therefore, the staff finds the proposed location of the waste treatment plant acceptable as it does not affect compliance with GDC 60, with respect to preventing contamination by radioactive water.

9.2.6.5 *Post Combined License Activities*

There are no post-COL activities related to this section.

9.2.6.6 *Conclusion*

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to sanitary drains, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the requirements of NRC regulations, and the acceptance criteria in NUREG-0800, Section 9.2.4. The staff based its conclusion on the following:

- WLS DEP 6.4-1, related to design changes affecting habitability of the main control room and changes to the calculated doses to control room operators, is reviewed and found acceptable by the staff in Section 21.2 of this SER.

- WLS SUP 9.2-1 is acceptable because the applicant has provided sufficient information on the location of the waste treatment plant to satisfy GDC 60, with respect to preventing contamination by radioactive water.

9.2.7 Central Chilled Water System (Related to RG 1.206 Section C.III.1, Chapter 9, C.I.9.2.2, "Cooling System for Reactor Auxiliaries (Closed Cooling Water Systems)")

The central chilled water system is a nonsafety system that provides chilled water to the cooling coils of the supply air handling units and unit coolers of several radiologically controlled areas of the plant.

Section 9.2 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 9.2.7, "Central Chilled Water System," of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review. The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the VEGP COL application are documented in NUREG-1793 and its supplements.

9.2.8 Turbine Building Closed Cooling Water System

9.2.8.1 *Introduction*

The turbine building closed cooling water system (TCS) is a nonsafety system that provides closed-loop cooling for the removal of heat from heat exchangers in the turbine building and rejects the heat to either the CWS or the RWS. The system consists of two 100-percent capacity pumps, three 50-percent capacity heat exchangers (connected in parallel), one surge tank, one chemical addition tank, and associated piping, valves, controls, and instrumentation. Backwashable strainers are provided upstream of each TCS heat exchanger. System piping is made of carbon steel, except that nonmetallic piping may be used in accordance with ASME B31.1, "Power Piping," if justified by evaluation.

9.2.8.2 *Summary of Application*

Section 9.2 of the WLS COL FSAR, Revision 11, incorporates by reference Section 9.2 of the AP1000 DCD, Revision 19. Section 9.2 of the DCD includes Section 9.2.8.

In addition, in WLS COL FSAR Section 9.2.8, the applicant provided the following:

<u>*Site-Specific Information Replacing Conceptual Design Information*</u>

- WLS CDI

The applicant provided additional information to replace conceptual design information (CDI) in the AP1000 DCD with site-specific information identifying the source of cooling water for the WLS TCS heat exchangers.

9.2.8.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the TCS are given in Section 9.2.2 of NUREG-0800.

9.2.8.4 *Technical Evaluation*

The NRC staff reviewed Section 9.2.8 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the TCS. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the WLS COL FSAR:

Site-Specific Information Replacing Conceptual Design Information

- WLS CDI

The AP1000 standard plant allows the use of either circulating water or raw water for removing heat from the TCS heat exchangers. The AP1000 DCD leaves it up to the COL applicant to specify a specific source of cooling water for plant-specific applications. The WLS design specifies the use of both circulating water and raw water for this purpose. This arrangement was reviewed and approved by the NRC during its evaluation of the AP1000 DCD. Consequently, the WLS design is consistent with the AP1000 licensing basis as approved by the staff, which includes conformance with NUREG-0800 Section 9.2.2 (as applicable). Therefore, the supplementary design information that was provided for the WLS TCS is acceptable.

The COL applicant modified FSAR Section 9.2.8.2.3, "System Operations - Startup," in Revision 1 to eliminate the provision that the CWS must be placed in operation prior to placing the TCS in operation. Based on the staff's review, this appeared to be an apparent departure from the description provided in the AP1000 DCD, but it was not recognized and evaluated as such. Therefore, in **RAI 038 Question 9.2.2-1**, the staff requested that the applicant revise FSAR Section 9.2.8.2.3 to address the need to properly address this change to the startup description provided in the AP1000 DCD Tier 2, Revision 16, Section 9.2.8.2.3. Based on the applicant's response, the applicant provided clarification on how the TCS system is to be placed in operation during startup. The FSAR markup states that after cooling water flow from the CWS, or RWS when applicable, is established but prior to the operation of systems that required turbine building closed cooling water flow, then the TCS is placed into operation. The staff considers the licensee's resolution of this issue to be acceptable since it clarifies the cooling water sources as either CWS or RWS. In addition, the staff verified this item is not considered a departure. The staff verified that the WLS COL FSAR markup that was provided with the applicant's RAI responses was added to Revision 1 of the COL; therefore, **Question 9.2.2-1** is resolved.

9.2.8.5 *Post Combined License Activities*

There are no post-COL activities related to this section.

9.2.8.6 *Conclusion*

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to TCS, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the acceptance criteria given in Section 9.2.2 of NUREG-0800. The staff based its conclusion on the following:

- WLS CDI is acceptable because the design of the TCS meets the guidance in Section 9.2.2 of NUREG-0800, with respect to the source of cooling water for the removing heat from the TCS heat exchangers.

9.2.9 Waste Water System (Related to RG 1.206 Section C.III.1, Chapter 9, C.I.9.3.3, "Equipment and Floor Drainage System"

9.2.9.1 *Introduction*

The waste water system (WWS) has no safety-related function other than main control room envelope isolation. A normally closed safety-related isolation valve is provided in the drain line penetrating the main control room. The drain line is safety related up to the isolation valve to ensure that the main control room habitability pressure boundary is maintained. The waste water system collects and processes the waste water from the equipment and floor drains in the nonradioactive building areas during plant operations and outages. The waste water from the turbine building sumps flows to a waste water retention basin, if required, for settling of suspended solids and treatment before discharge. The waste water retention basin transfer pumps discharge the basin effluent to the blowdown sump prior to discharge to the Parr Reservoir via the plant outfall piping. The design of the system precludes inadvertent discharge of radioactively contaminated drainage

9.2.9.2 *Summary of Application*

Section 9.2 of the WLS COL FSAR, Revision 11, incorporates by reference Section 9.2 of the AP1000 DCD, Revision 19. Section 9.2 of the AP1000 DCD includes Section 9.2.9, "Waste Water System," which addresses Section 9.3.3, "Equipment and Floor Drainage System," of NUREG-0800.

In addition, in WLS COL FSAR Section 9.2, the applicant provided the following:

<u>*AP1000 COL Information Item*</u>

- WLS COL 9.2-2

The applicant provided additional information in WLS COL 9.2-2 to address COL Information Item 9.2-2, by including additional design information to the waste water retention basin portion of AP1000 DCD Section 9.2.9.2.2.

9.2.9.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the WWS are given in Section 9.3.3 of NUREG-0800.

The regulatory basis for acceptance of the COL information item is established in:

- GDC 4
- GDC 60

9.2.9.4 *Technical Evaluation*

The NRC staff reviewed Section 9.2.9 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the WWS. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the WLS COL FSAR:

<u>*AP1000 COL Information Item*</u>

- WLS COL 9.2-2

The applicant provided additional information in WLS COL 9.2-2 to resolve COL Information Item 9.2-2. COL Information Item 9.2-2 states:

> The Combined License applicant will address the final design and configuration of the plant waste water retention basins and associated discharge piping, including piping design pressure, basin transfer pump size, basin size, and location of the retention basins.

The NRC staff reviewed the resolution to WLS COL 9.2-2 with respect to the design of the plant waste water retention basin (WWRB) and associated components included under

Section 9.2.9.2.2, ”Component Description” of the WLS COL FSAR. To address WLS COL 9.2-2, details were provided for the location of the WWRB and routing configuration.

The waste water from the WWRB is discharged to the Ninety-Nine Islands on the Broad River through a common blowdown sump with inputs from the Unit 1 and 2 WWRB and CWS cooling tower blowdown. There is one WWRB per unit. The method for forwarding the waste water from the basin to the blowdown sump is by use of two transfer pumps.

In order to meet GDC 60, the applicant must demonstrate suitable control of the release of radioactive materials in liquid effluent. Upon review of WLS COL 9.2-2, the staff requested the applicant, in RAI 9.3.3-1, to provide a discussion on whether all site-specific potentially radioactive fluid draining into and downstream of the WWRB will be monitored prior to distribution or provide a justification for not providing radiation monitoring. The staff also requested that the applicant provide the additional details of the associated components (i.e., transfer pumps, size of basin, etc.) as requested in the COL item.

The applicant responded to RAI 9.3.3-1 in a letter dated May 4, 2010. The response provided detailed information on radiation monitoring, level instrumentation and components for the WWS. Each unit's WWRB is divided into two separate compartments, which allows one compartment to be out of service while the other compartment is available. Each WWRB is constructed such that its contents dissolved or suspended, do not penetrate the liner and leach into the ground. The configuration and size of the WWRB allows settling of solids larger than 10 microns, which may be suspended in the waste water stream. The applicant confirmed that the potentially contaminated fluids entering the WWRB from the turbine building sumps are monitored with a radiation monitor on the common discharge piping. As indicated in the RAI response, there is several effluent lines within the scope of the certified design that bypass this radiation monitor. These include the diesel fuel area sumps, SWS cooling tower blowdown, SWS strainer blowdown, and CWS strainer backwash. The RAI response clarified that these lines do not come in contact with radioactive sources or contain radiation monitoring prior to discharge into the WWRB. The applicant indicated that for WLS Units 1 and 2, there are no additional site–specific influent streams to the WWRB outside of those associated with the certified design. Waste water can also be sampled prior to discharge from the WWRB.

Two 100 percent capacity submersible type pumps send waste water from the WWRB to the common blowdown sump. The transfer pumps have 750 gallons per minute (gpm) capacity and the discharge piping has a design pressure of 150 pounds per square inch gauge (psig).

The blowdown sump is a concrete structure and is open to the atmosphere. The blowdown sump is located to the east of Units 1 and 2, outside the protected area. The blowdown sump, common to both WLS Units 1 and 2, receives input from the WWRB and mixes with high volume CWS stream. The RWS provides water for an alternate dilution source to the blowdown sump when the CWS blowdown is not sufficient or not available for that purpose. As discussed in WLS COL FSAR Section 9.2.11.4, the RWS comes directly from the make-up pond A intake and does not interact with any recognized radioactive sources.

The combined dilution flow gravity drains from the blowdown sump flows through an outfall pipe to the Ninety-Nine Islands dam on the Broad River. The blowdown sump outfall is sized to prevent sump overflow during maximum inlet flow to the sump. At the dam, the dilution flow is mixed with liquid radwaste effluent from each unit and discharged to the environment through a diffuser mounted on the upstream side of the dam. The elevation difference between the sump and the river prevents liquid radwaste cross-contamination of the blowdown sump. The liquid radwaste is monitored and sampled for radiation and is addressed in detail in WLS COL FSAR Section 11.2.

Based on the content in WLS COL FSAR Section 9.2.9 and the RAI 9.3.3-1 response, the staff concludes that the design of the WWS complies with GDC 60, with respect to control of radiation release to environment because as discussed above, the WWS does not normally interact with any potential radioactive sources and any influent streams with the potential to become contaminated are monitored.

To protect against flooding, the WWRB will be equipped with level instrumentation used to control the WWRB transfer pumps and to alarm when the basin level reaches a point where operator action is required. Each WWRB is located approximately 850 feet north of the associated power block. The normal WWRB water level in the basin is at or below grade. Site grading and the distance between the basins and the power block ensures that there will be no adverse impact on safety-related or RTNSS SSCs in the unlikely event of an overflow of the WWRB.

Waste water and blowdown effluent from the blowdown sump drains by gravity to the Broad River via the plant outfall piping. The blowdown sump outfall pipe is sized with adequate capacity to gravity drain the blowdown sump at the highest anticipated influent flow rate. Therefore, no level instrumentation is provided at the blowdown sump. The blowdown sump is located well away from the power block (approximately 1125 feet). Site drainage features ensure that there will be no impact on safety-related or RTNSS SSCs in the unlikely event of an overflow of the sump. Based on the content in WLS COL FSAR Section 9.2.9 and the RAI 9.3.3-1 response, the staff concludes that the design of the WWS complies with GDC 4, with respect to flood protection because the WWRB are designed with two 100-percent pumps controlling level by use of WWRB level instrumentation.

Based on the information provided in WLS COL FSAR Section 9.2.9 and in the response to RAI 9.3.3-1, the staff finds that the applicant has adequately addressed COL information item WLS COL 9.2-2. The staff finds that GDC 4 is met based on the WWS arrangement to prevent flooding that could adversely affect safety-related SSCs and GDC 60 is met based on the requirements for controlling the release of radioactive materials by preventing the inadvertent transfer of contaminated fluids to system portions for noncontaminated drainage. Therefore, RAI 9.3.3-1 is closed and incorporation of the proposed markup into a future revision of the WLS COL FSAR is identified as **Confirmatory Item 9.2-3**.

Resolution of Confirmatory Item 9.2-3

Confirmatory Item 9.2-3 is an applicant commitment to revise its FSAR to include the information proposed in their response to RAI 9.3.3-1. The staff verified that the WLS COL FSAR was appropriately revised. As a result, Confirmatory Item 9.2-3 is now closed.

9.2.9.5 *Post Combined License Activities*

There are no post-COL activities related to this section.

9.2.9.6 *Conclusion*

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to the WWS, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that, the relevant information presented in the WLS COL FSAR is acceptable and meets the guidelines given in Section 9.3.3 of NUREG-0800. The staff based its conclusion on the following:

- WLS COL 9.2-2 is acceptable because the staff finds that the relevant information in the WLS COL FSAR meets the applicable requirements of GDC 4 and GDC 60.

9.2.10 Hot Water Heating System

The hot water heating system is a nonsafety-related system that supplies heated water to selected nonsafety-related air handling units and unit heater in the plant during cold weather operation, and to the containment recirculation fan coil units during plant outages in cold weather.

Section 9.2 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 9.2.10 of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

9.2.11 Raw Water System

9.2.11.1 *Introduction*

WLS COL FSAR Section 9.2.11, "Raw Water System," describes the RWS for WLS Units 1 and 2. The RWS is a nonsafety-related system that pumps water from the Broad River for use by the WLS units. The RWS supplies raw water for makeup to the CWS cooling tower basins, makeup for the SWS cooling tower basins, WWS alternate dilution flow, the demineralizer water treatment system (DTS), the fire protection system (FPS); and serves as an alternate source of cooling water for the TCS heat exchangers. The RWS also provides alternate sources of make-up water to the SWS cooling towers during a loss of offsite power from the secondary fire water storage tank clearwell, the clarified water subsystem, and the raw water supply subsystem. The RWS consists of the river water subsystem, the raw water supply subsystem (which includes make-up pond A), the make-up pond B subsystem, refill subsystem, make-up pond C system (offsite supply), clarifier subsystem, and the clarified water supply subsystem. The RWS is shared by the two WLS units.

Make-up pond A serves as a central repository for raw water and contains the intake structure for the station. During normal Broad River flow conditions, withdrawal from the river is used to maintain a normal level in make-up pond A and, if required, store water in make-up ponds B and C. When permit conditions limit withdrawal from the Broad River, withdrawal from make-up ponds B and C, and if allowed, the Broad River is used to maintain a normal level in make-up pond A. The water inventory required to support the power generation design basis is provided by the raw water supply subsystem and maintained in make-up pond A. The river water, refill, and make-up ponds B and C subsystems provide water storage and source diversity to adapt to Broad River flow conditions. Make-up pond A has a usable storage volume of 1200 acre-foot, which provides sufficient capacity to support a dual unit cooldown to cold shutdown conditions and maintain the station in this condition for longer than 7 days.

9.2.11.2 *Summary of Application*

Section 9.2.11 of the WLS COL FSAR provides information concerning the RWS design basis, system description, system operation, safety evaluation, tests and inspections, and instrumentation. The RWS was only vaguely referred to in the AP1000 DCD in relation to the CWS, SWS, DTS, and FPS, and a RWS section was not included in the AP1000 DCD for the NRC staff to evaluate. In addition, AP1000 DCD, Table 1.7-2, “AP1000 System Designators and System Diagrams,” indicates that the RWS is “wholly out of scope.” The RWS is needed in order to operate the WLS units and consequently, the applicant has provided a complete description of this system in the WLS COL FSAR.

In addition, in WLS COL FSAR Section 9.2.11, the applicant provided the following:

Tier 2 Departure

- STD DEP 1.1-1

The applicant proposed a Tier 2 departure (DEP) from the AP1000 DCD by adding a new Section 9.2.11, “Raw Water System” after DCD Section 9.2.10 and renumbering DCD Sections 9.2.11 and 9.2.12 as Sections 9.2.12 and 9.2.13, respectively.

Supplemental Information

- WLS SUP 9.2-4 - reference RAI response letter from Duke Energy dated May 15, 2009. This is supplemental description including FSAR sections addressing design basis, system description, component description, system operations, safety evaluation, testing and inspections, and instrumentation applications.

The applicant provided supplemental information by adding the new Section 9.2.11 after AP1000 DCD Section 9.2.10.

9.2.11.3 *Regulatory Basis*

In most cases, the regulatory bases for AP1000 systems are provided in NUREG-1793 and its supplements. However, because the RWS was not addressed in the AP1000 DCD, it was not evaluated by the staff in NUREG-1793 and a regulatory basis for this system was not established for the standard plant design. Consequently, the staff is unable to refer to

NUREG-1793 for the regulatory basis of the RWS and instead, the regulatory basis of the RWS for the WLS units is provided in this section.

The RWS pumps water from the Broad River for use in dissipating the heat necessary for normal power operation (among other things) and in this capacity, the RWS is somewhat similar to a CWS. Because large amounts of water are being pumped and stored by the RWS, flooding is a major consideration. The regulatory criteria that pertain to CWS are provided in NUREG-0800, Section 10.4.5, “Circulating Water System.” As specified in this section of NUREG-0800, the staff’s acceptance of the RWS is based upon compliance with GDC 4, by confirming that design provisions for minimizing the occurrence and accommodating the effects of discharging water that may result from RWS failures are adequate.

The RWS also provides makeup water for the SWS cooling tower basins and in this capacity supports the SWS and the nonsafety-related ultimate heat sink (UHS) functions. The regulatory criteria that pertain to the SWS and the UHS are provided in NUREG-0800, Section 9.2.1, “Station Service Water System,” and NUREG-0800, Section 9.2.5, “Ultimate Heat Sink,” respectively. As specified in these NUREG-0800 sections, the staff’s acceptance is based upon compliance with GDC 2, “Design Basis for Protection against Natural Phenomena.” The staff considers the RWS to be acceptable with respect to GDC 2 if it satisfies Position C.2 of RG 1.29, “Seismic Design Classification.” Position C.2 indicates that the design is acceptable if RWS failures do not adversely affect the control room occupants or safety-related SSCs.

9.2.11.4 *Technical Evaluation*

As discussed above in the Regulatory Basis section, the RWS was not specifically described for the AP1000 standard plant design and consequently, it was not evaluated by the staff in NUREG-1793. In addition, AP1000 DCD, Table 1.7-2, indicates that the RWS is “wholly out of scope.” The staff reviewed the information provided in Section 9.2.11 of the WLS COL FSAR, Revision 4 that describes the RWS for the WLS units, including the information provided by Figure 9.2-201, “Raw Water System,” through Figure 9.2-207. As discussed above, the staff’s evaluation in this section focuses primarily on RWS failure considerations and on the capability and reliability of the RWS to perform its cooldown function.

The following staff’s technical evaluation includes the Tier 2 departure and supplemental information addressed in Section 9.2.11.2 of this SER.

A. GDC 2, “Design Bases for Protection against Natural Phenomena”; RG 1.29, “Seismic Design Classification”; and GDC 4, “Environmental and Dynamic Effects Design Bases”

The staff’s review of the information in WLS COL Section 9.2.11 was to confirm that RWS failures will not adversely impact the control room occupants or adversely affect SSCs that are safety-related or designated for RTNSS. Although Section 9.2.11.1.1, “Safety Design Basis,” stated that failures of the RWS or its components will not affect the ability of safety-related systems to perform their intended functions, more detailed information was needed to adequately describe the consequences of RWS failures and to explain why safety-related SSCs are not affected. Likewise, additional information was needed to explain why a failure of the RWS will not adversely affect RTNSS systems and components or impact the control room occupants, or result in an unacceptable release of radioactive material to the environment. Because the applicant did not identify and address these considerations, the staff was unable to confirm compliance with GDC 2, GDC 4, and passive plant policy considerations. The staff issued RAI 9.2.1-1, dated September 23, 2008, asking the applicant to revise WLS COL FSAR Section 9.2.11 to address the impact of RWS failures accordingly, including development of plant-specific ITAAC and test program specifications as appropriate. The applicant's response to RAI 9.2.1-1, dated October 28, 2008, did not adequately address the question and subsequently, the staff expanded the question and issued supplemental RAI 9.2.1-5, dated January 28, 2009, to more fully address RWS failure considerations.

In its response, dated May 15, 2009, the applicant provided a detailed response to the GDC 2, GDC 4, ITAAC, and testing questions. In the response that follows, the applicant stated that failure of the RWS piping located in the yard and inside the turbine building were considered. In addition, the staff has determined that appropriate testing of the RWS was addressed in WLS COL FSAR Section 14.2.

> The potential failures of the RWS and the corresponding impact on SSCs that are safety-related or AP1000 equipment Class D are described below.
>
> Underground piping transfers water from the Broad River to the make-up ponds. The significant above ground portions of this piping are at the intake structures. This piping does not interface with Class D systems nor is it routed in close proximity to safety-related structures or Class D equipment. Underground piping from the make-up pond A intake structure supplies the water treatment equipment in the RWS clarifier subsystem, make-up to the CWS cooling towers, alternate dilution flow to the WWS and alternate make-up to the SWS. This piping is in relatively close proximity to the underground portions of the CWS, and a break in the RWS piping is bounded by a break in the CWS. As discussed in DCD Tier 2, Section 3.4.1.1.1, "Protection from External Flooding," a failure of the cooling tower or the SWS or the CWS piping under the yard could result in a potential flood source. The consequences of a failure in the yard would be enveloped by the analysis described in DCD Tier 2, Section 10.4.5, for failure of the CWS. Site grading will carry the water away from safety-related or important to safety SSCs.
>
> Underground piping from the RWS clarifier subsystem supplies the clarified water storage tank, where additional underground piping supplies the yard-located interface with the FPS, as well as the SWS, DTS and TCS interfaces located in the turbine building. This underground piping is also in close proximity to the underground portions of the CWS and is bounded by the analysis previously discussed.
>
> Short runs of RWS piping from the raw water supply and clarified water supply subsystems are routed inside the turbine building to provide normal and alternate supplies to SWS. Clarified water supply piping also supplies the interface points with DTS and TCS. The RWS to SWS interface is at the SWS make-up control valve V009 (refer to DCD Tier 2, Figure 9.2.1-1). The SWS piping is routed from the control valve to the top of the SWS cooling tower basin. There is an air gap between the piping discharge and the SWS cooling tower basin water level. This air gap ensures that any break in the raw water make-up flow path will not result in the draining of the SWS cooling tower basin by preventing backflow from the basin to the break. Therefore, any flooding will be from the RWS water that discharges through the break prior to securing the make-up supply. The RWS piping to DTS and TCS is on the 100'-0" elevation of the

turbine building (WLS Elevation 590') and the primary source of flooding in this scenario would be from the RWS water that discharges through a break prior to securing the clarified water supply subsystem. A break in the RWS piping to the SWS, DTS and TCS is bounded by a break in circulating water piping. As discussed in DCD Tier 2, Section 3.4.1.2.2.3, “Adjacent Structures Flooding Events,” the bounding flooding source inside the turbine building is a break in the circulating water piping. Flow from any postulated line breaks above elevation 100'-0" would flow down to elevation 100'-0" via floor gratings and stairwells and would run out of the building through a relief panel in the turbine building west wall. There is no safety-related equipment in the turbine building. The components cooling water and service water components on elevation 100'-0", which provide the RTNSS support for the RNS will remain functional following a flooding event in the turbine building since the pump motors and valve operators are above the flood level. Therefore, a failure of the RWS piping within the turbine building will not adversely impact any safety-related or important to safety SSCs.

The control room is located inside the AP1000 nuclear island. No RWS piping is located inside or outside in the vicinity of the nuclear island. Therefore, there are no RWS pipe breaks or flooding events which could impact the control room. Atmospheric releases of the pH adjustment and chlorination chemicals used in RWS treatment are bounded by the DCD Section 6.4 discussion, as the storage volume is much smaller than those used in the CWS treatment. All other chemicals used for RWS treatment are non-toxic, small in volume, and do not represent a hazard to the control room. Therefore, RWS flooding or postulated chemical releases will not adversely impact control room habitability.

Accidental releases of radioactive fluids in ground and surface waters are addressed in WLS COL FSAR Section 2.4.13. In accordance with this discussion, any radioactive fluids released from the AP1000 power block would follow the preferential path for groundwater movement. This flow path is generally northward towards the Broad River.

The RWS piping corridor between the intake on the Broad River and make-up pond A is the closest to the preferential groundwater flow path. This piping corridor is positioned above the water table and located approximately 600 to 2150 feet east of the preferential groundwater flow path. The remaining RWS underground piping corridors are located upgradient to the south and west of the preferential groundwater flow path. When the RWS system is in operation it is under positive pressure. Therefore, migration of any potential contamination from the power block to the piping is considered unlikely.

The RWS does not have the potential to be a flow path for radioactive fluids. The RWS operates at a higher system pressure than those systems that it directly interfaces with; therefore, in-leakage is not feasible when the system is in operation. During normal operations, the interfacing systems for RWS are CWS, WWS at the blowdown sump, SWS, FPS, DTS and TCS. None of these systems have interfaces with radioactive systems.

As discussed in WLS COL FSAR Section 9.2.11.3.5, “Raw Water Supply Subsystem,” the RWS supplies an alternate source of dilution water to the WWS for diluting the WLS effluent stream when the normal dilution source, CWS blowdown, is not available. This function is supported by routing branch lines from the raw water supply subsystem to the CWS blowdown sump. The blowdown sump is open to atmosphere and as shown on

> WLS COL FSAR Figure 1.1-202 is located on the east side of the site at an elevation approximately 60 feet above the Broad River. The CWS blowdown sump mixes CWS blowdown (and, if required, RWS) with discharge from WWS and the combined dilution flow gravity drains through an outfall pipe to the Ninety-Nine Islands dam on the Broad River. At the dam, the dilution flow is mixed with liquid radwaste effluent from each unit and discharged to the environment through a diffuser mounted on the upstream side of the dam. There are no valves on the outfall piping between the blowdown sump and the dam, so the elevation difference between the sump and the river prevents liquid radwaste cross-contamination of the blowdown sump. There is an air gap maintained between the RWS piping discharge into the sump and the sump level that provides additional assurance that cross-contamination is unlikely.

As described in WLS COL FSAR Section 14.2.9.4.24, “Raw Water System,” initial testing verifies that as installed components supply raw water to the CWS cooling tower basin, SWS cooling tower basin, FPS storage tanks, and other systems, as described in WLS COL FSAR Section 9.2.11. Testing shall consist of performance and functions of components and integrated systems.

The applicant’s response to RAI 9.2.1-5 was found acceptable based on the following evaluation. The staff determined failure of the RWS or its components will not affect the ability of any safety-related systems to perform their intended safety functions nor will it adversely impact any Class D systems. Postulated breaks in the RWS piping will not impact safety-related components because the RWS is not located in the vicinity of any safety-related equipment and the water from the postulated break will not reach any safety-related equipment, result in impact to the control room occupants, or result in a release of radioactivity to the environment. Testing of the RWS has been properly addressed, and the RWS instrumentation requirements have been satisfied. In addition, based on the staff’s review of the instrumentation application of the RWS as described in WLS COL FASR Section 9.2.11.7, the operators have sufficient indications of system alarms to identify component failures, such as traveling screens, strainers, and pumps. Since the RWS is not safety-related and its failure does not lead to the failure of any safety-related systems, the staff has concluded that the requirements of GDC 2 and GDC 4 have been satisfied; therefore, RAIs 9.2.1-1 and 9.2.1-5 are resolved.

B. Cold Shutdown

The RWS is relied upon for achieving and maintaining cold shutdown conditions, which is necessary for satisfying Technical Specification requirements. In particular, the RWS is relied upon for cooling the RCS from Mode 4 to Mode 5 conditions within 36 hours. The staff found that Section 9.2.11 did not provide a clearly defined design basis with respect to the RWS cooldown function, and the reliability and capability of the RWS to perform this function for the most limiting situations were not described and addressed in this regard. For example, the minimum RWS flow rate, water inventory, temperature limitations, and corresponding bases for providing SWS makeup for the two WLS units were not described. Also, the suitability of RWS materials for the plant-specific application and measures being implemented to resolve vulnerabilities and degradation mechanisms to assure RWS functionality over time were not addressed. Because the applicant did not adequately define and address RWS design-bases considerations with respect to its cooldown function, the staff was unable to confirm that the cooldown and policy considerations that apply to passive plant designs were satisfied. Therefore, in RAI 9.2.1-2, dated September 23, 2008, the staff asked the applicant to provide clarification. The applicant's response to RAI 9.2.1-2 dated October 28, 2008, did not adequately address the question and subsequently the staff expanded the question and issued supplemental RAI 9.2.1-6, dated January 28, 2009, to further fully address cold shutdown considerations.

In its response, dated May 15, 2009, the applicant stated that the RWS consists of several subsystems. Of these subsystems, only two supply water to support plant system functions during Modes 1 through 5. The clarified water supply subsystem supplies treated make-up, fill water to the DTS and FPS in both units, serve as the preferred make-up, and fill supply to the SWS. The raw water supply subsystem supplies untreated make-up and fill water to the CWS. The raw water supply subsystem also provides an assured make-up supply to SWS to ensure that the power generation design basis for SWS is maintained during abnormal conditions that could deplete the inventory in the clarified water supply subsystem.

This response specifically focuses on the clarified water supply and raw water supply subsystems interfaces with the SWS. This is because, as noted in the response to RAI 9.2.1-5, the other functions performed by the RWS do not have a direct interface with any other system identified within the AP1000, which is safety-related, designated for RTNSS, or designated as AP1000 Class D.

> The RWS clarified water supply and raw water supply subsystems provide a water fill/makeup interface with the SWS. The SWS has investment protection short-term availability controls as described in AP1000 DCD Table 16.3-2, which are applicable in Mode 5 with the RCS pressure boundary open, and in Mode 6 with the upper internals in place or cavity level less than full. Under these conditions, the SWS is directly providing active core cooling and, as noted in the response to RAI 9.2.1-5, was evaluated by Westinghouse and determined to meet the RTNSS criteria as documented in NUREG-1793 and Westinghouse Topical Report WCAP-15985, "AP1000 Implementation of the Regulatory Treatment of Nonsafety-Related System Process." Unlike the SWS, the RWS does not directly provide core cooling. As discussed in response to RAI 9.2.1-5, RWS support of the SWS cooling function was evaluated in WCAP-15985 and determined to not meet the RTNSS criteria and to not require investment protection short-term availability controls.
>
> In the unlikely event of a failure of the RWS to provide adequate make-up flow to the SWS cooling tower basins during the short time period in which the SWS is performing a RTNSS function as stated above, the remaining inventory in the service water cooling tower basins and the stored water, which is available in the upper region of the secondary fire water tank provide ample time (more than 24 hours) to restore the RWS make-up flow or take the procedural actions necessary to exit the conditions for applicability. Therefore, the RWS is not a RTNSS system or subject to investment protection short-term availability controls. However, as described below, the RWS is designed to be a highly reliable and robust system, capable of operating during a loss of normal alternating current (ac) power to provide RWS make-up flow under normal and abnormal conditions. Procedural controls, which provide for continued operation of the RWS or re-establishment of operations under off-normal conditions, will be in the operating procedures, where appropriate.

As described in AP1000 DCD Section 5.4.7.1.2.1, “Shutdown Heat Removal,” the RNS, in conjunction with its associated support systems, the CCS and the SWS, are used for shutdown heat removal. The RWS provides indirect support for this function by providing a source of make-up water to the SWS cooling tower basins to compensate for evaporation, drift, and blowdown. The RWS provides this make-up water to support the cooling requirements for the SWS. During a normal plant cooldown, the RNS and CCS reduce the temperature of the RCS from approximately 350 °Fahrenheit (F) to approximately 125 °F within 96 hours after shutdown. The RWS is designed to provide ample make-up flow to both units' cooling tower basins during these conditions using the clarified water supply pumps.

WLS COL FSAR Figure 2.4.1-202, “Water Balance Summary,” identifies the maximum make-up requirement for RWS to the SWSs in both units to be 1660 gpm (830 gpm per unit). This demand represents a design maximum make-up to the SWS cooling towers, occurring four hours after a simultaneous shutdown of both units, when the maximum SWS heat load is encountered at the beginning of cooldown. This flow rate is very conservative, as the decay heat load decreases during cooldown with an accompanying decrease in make-up requirements.

If cooldown to cold shutdown (Mode 5) is required within 36 hours to comply with a limiting condition for operation (LCO) in accordance with the Technical Specifications, heat will be transferred from the RCS via the steam generators to the main steam system for a longer period of time, allowing RNS to be placed in service at a lower temperature with lower decay heat levels. Because of the reduced RNS heat removal requirements associated with this cold shutdown sequence, the required RWS make-up flow to the SWS cooling towers is less than normal cooldown requirements.

For a loss of normal ac power scenario, Westinghouse AP1000 design data indicates an RWS flow of approximately 108 gpm will provide sufficient make-up to account for evaporation and drift losses from the SWS cooling tower following the first 28 hours of event initiation.

The clarified water supply subsystem is the normal make-up source for the SWS cooling towers in both units and the preferred source of water for the normal plant cooldown described in the power generation design basis. The clarified water storage tank, shared by both units, has a capacity of 2.7 million gallons. There are four clarified water supply pumps (two per unit) that take suction from the storage tank. Each pump has a design flow rate of 1500 gpm, so one pump can easily supply the maximum make-up requirements for the associated unit.

The raw water supply subsystem provides the assured make-up supply to the SWS to ensure that the power generation design basis for the SWS is maintained during abnormal conditions that could deplete the inventory in the clarified water supply subsystem. The inventory for the subsystem is make-up pond A, which has a useable storage volume of 1200 acre-foot and provides sufficient capacity to support a dual unit cooldown to cold shutdown conditions and maintain the station in this condition for longer than 7 days. There are six raw water supply pumps (three per unit) that take suction from make-up pond A. Each pump has a design flow rate of 15,000 gpm.

> The underground RWS piping will be high-density polyethylene (HDPE), which is not susceptible to corrosion or biological fouling and is designed to ASME B31.1; therefore, periodic inspections of the underground RWS piping are not required. Equipment that remains idle for extended periods of time (pumps, valves) will be operated periodically in accordance with vendor recommended maintenance practices.
>
> The lack of designation of the RWS as RTNSS or Class D indicates there is no performance requirement for the system in the event of a single active failure or during a loss of normal ac power. Nonetheless, the RWS is highly reliable based on its design, and a single failure of an active component in the RWS would not affect normal plant cooldown. Only one of the two clarified water supply pumps or one of the three raw water supply pumps are required to support make-up to the SWS cooling tower basins in each unit during all modes of SWS operation. Failure of an operating pump or electrically-operated valve in the make-up path to the SWS would not prevent the RWS from providing make-up to the cooling towers. The clarified water supply pumps are supplied from separate buses that are automatically loaded on the standby power supply during a loss of normal ac power. All RWS valves in the make-up to SWS are manual. Restoring make-up flow requires starting a pump from the control room. The raw water supply pumps are supplied by separate buses and two pumps in each unit are on buses that can be manually loaded on the standby power supply. If the clarified water supply subsystem is not available, operator actions will be taken to align the raw water supply subsystem as described in WLS COL FSAR Section 9.2.11.3.5. The water inventory in the SWS cooling tower basins provides adequate time to perform the manual actions needed to restore SWS make-up. The RWS, therefore, continues to maintain the capability to provide make-up water to the SWS cooling tower basins during loss of normal ac power events.
>
> The raw water screen wash pumps and traveling screen do not have backup power. The traveling screens are powered by the normal ac power system, which is backed by a standby power supply for occurrences of loss of normal ac power. RWS make-up requirements following a loss of normal ac power are a small fraction of the normal flow. In such condition, the intake screens act as passive screens.
>
> In the unlikely event that all RWS flow to the SWS cooling towers is lost, there is ample time to identify and correct the situation or to align alternate sources of water to provide that make-up flow, and the RWS is shown to not be a RTNSS system nor subject to investment protection short-term availability controls. It is also important to note that the RNS, CCS, SWS, nor RWS are required to establish and maintain the AP1000 plant in a safe shutdown condition, since passive safety-related systems perform that function. This is explicitly recognized throughout the AP1000 DCD and NUREG-1793.

The applicant’s response to RAI 9.2.1-6 was found acceptable based on the following evaluation. The staff finds that the RWS is designed with the provision of single failure since many of the raw water supply subsystem and clarified subsystem components can be supplied with backup power from the onsite diesel generators as necessary. During a loss of SWS make-up from the RWS supply subsystem or clarified water supply subsystem, make-up to the SWS is not required for 12 hours due to existing cooling tower basin inventory. After 12 hours, onsite make-up capacity from the fire protection storage tank is available for more than an additional 12 hours. In addition, the RWS is considered highly reliable and able to supply required water for the SWS for greater than 7 days due to the redundancies of pumps, strainers, screens, and screen wash components associated with make-up pond A. The raw water screen wash pumps and traveling screen do not have backup power; however, the staff finds the RWS make-up requirements following a loss of normal ac power are a small fraction of the normal flow. In such condition, the intake screens act as passive screens. This is acceptable because lower flow rates and limited duration reduce the potential for entrainment and impingement.

In addition, the staff finds the RWS underground material acceptable since buried HDPE will be designed and installed in accordance with industry Codes, such as ASME B31.1 and American Water Works Association (AWWA) C906, "Polyethylene (PE) Pressure Pipe and Fittings, 4 in (100mm) through 63 in (1,575mm), for Water Distribution and Transmission." This material is an industry proven material that is corrosion resistant inside and out, hydraulically smooth, and tends to resist buildup (bio-fouling) so the inner surface usually remains in this condition throughout the service life of the pipe. In addition, HDPE has a life expectancy of approximately 50 years. Ultraviolet protection is of no concern since the RWS HDPE piping will be buried. HDPE materials are well within the temperature and pressures ranges in which the RWS piping system will be exposed to during operations.

C. Regulatory Treatment of Non-Safety Related System (RTNSS)

The RWS supports the SWS cooling function by providing makeup water to the SWS cooling tower basins. The staff noted that while the SWS is designated for RTNSS during reduced reactor inventory conditions, the RWS is not needed to support the SWS cooling function when the reactor water inventory is reduced because RWS is not designated for RTNSS. However, there was no explanation in Section 9.2.11 as to why this was the case. Also, because the SWS cooling tower basins are very limited in their capacity, it was not clear why RWS makeup was not required for this situation. Therefore, the staff asked the applicant to explain this as part of RAI 9.2.1-2, dated September 23, 2008. The applicant's response to RAI 9.2.1-2, dated October 28, 2008, did not adequately address the question. Subsequently, the staff expanded the question and issued supplemental RAI 9.2.1-7, dated January 28, 2009, which asked the applicant to explain why RWS makeup is not needed during reduced reactor inventory conditions and in particular, to describe controls that will be implemented to ensure that assumptions remain valid.

In its response to RAI 9.2.1-7, dated May 15, 2009, the applicant referred to its RAI 9.2.1-6 response, which provided an explanation as to why the RWS is not designated as RTNSS and makeup from the RWS to the SWS cooling tower basins is not required during reduced reactor inventory conditions. The referenced RAI response also discusses that procedural control will be established to take the required actions to exit the conditions for applicability of the SWS as a RTNSS system, in the unlikely event of a failure to re-establish RWS makeup capability. Plant documentation, in the form of the system description for the RWS, will include the information addressed in the responses to RAI 9.2.1-6 and RAI 9.2.1-7.

In the response to RAI 9.2.1-6, the applicant also stated that the RWS does not have a direct interface with any other system identified within the AP1000, which is safety-related, designated for RTNSS, or designated as AP1000 Class D. The RWS provides a water fill/makeup function for the SWS, and the SWS has investment protection short-term availability controls as described in AP1000 DCD Table 16.3-2, "Investment Protection Short-Term Availability Controls," which are applicable in Mode 5 with the RCS pressure boundary open and in Mode 6 with the upper internals in place or cavity level less than full. Under these conditions, the SWS is directly providing active core cooling and was evaluated and determined to meet the RTNSS criteria as documented in NUREG-1793 and WCAP-15985. Unlike the SWS, the RWS does not directly provide core cooling and was evaluated in WCAP-15985 and determined to not meet the RTNSS criteria and to not require investment protection short-term availability controls. Neither the SWS nor the RWS are required to establish and maintain the AP1000 plant in a safe shutdown condition, since passive safety-related systems perform that function. This is explicitly recognized throughout the AP1000 DCD and NUREG-1793.

The applicant's response to RAI 9.2.1-6, also stated that in the unlikely event of a failure of the RWS to provide makeup flow to the SWS cooling tower basis during the short time period (as stated above) that SWS is performing a RTNSS function, the remaining inventory in the SWS cooling tower basins and water in the secondary fire tank will provide more than 24 hours to restore the RWS makeup flow or take procedural actions to exit the conditions for applicability.

In summary, the staff finds the applicant's response to RAI 9.2.1-7 acceptable because the NRC previously concluded in NUREG-1793 that the SWS meets the RTNSS criteria for provided active core cooling. The RWS does not directly provided core cooling. Therefore, the staff concludes the RWS need not be considered RTNSS and, RAIs 9.2.1-2, 9.2.1-6, and 9.2.1-7 are resolved and closed.

D. 10 CFR 20.1406, "Minimization of Contamination" Considerations

As specified by 10 CFR 20.1406, COL applicants are required to describe how facility design and procedures for operation will minimize the generation of radioactive waste and contamination of the facility and environment, and facilitate eventual plant decommissioning. Although the RWS has no interconnections with any systems that contain radioactive fluids, industry experience has shown that this alone may not be sufficient to prevent the RWS from becoming contaminated. For example, unplanned leaks or release of contaminated fluids as a result of component failures or transport, drainage problems in contaminated areas, and the migration of contamination through soils and other porous barriers over time have caused systems and areas of the plant that are not directly connected with contaminated systems to become contaminated. Also, because the RWS is used as a source of water for diluting liquid radwaste, this may create a potential for contaminating the RWS or for spreading contamination inadvertently. Therefore, the staff requested in RAI 9.2.1-3, dated September 23, 2008, that the applicant provide additional information to describe design provisions and other measures that will be implemented to satisfy the requirements specified by 10 CFR 20.1406, including measures that will be implemented to monitor the RWS for contamination and corrective actions that will be taken to eliminate any radioactive contamination that is identified (as appropriate). In a response dated October 26, 2008, it was stated that the RWS has no interconnection with systems that contain radioactive fluids. In addition, the applicant indicated that the groundwater monitoring program should minimize the possibility of contaminating the RWS from external subsurface sources. The applicant noted that the groundwater monitoring program is described in WLS COL FSAR Section 12AA.5.4.13, "Groundwater Monitoring Program." The staff's evaluation of the groundwater monitoring program is provided in the corresponding section of this SER. The staff considers the applicant's resolution of this issue to be acceptable, and RAI 9.2.1-3 is closed.

9.2.11.5 *Post Combined License Activities*

There are no post-COL activities related to this section.

9.2.11.6 *Conclusion*

The RWS was evaluated using the guidance referred to in the Regulatory Basis section as it pertains to these considerations and acceptability was based upon conformance with the NRC requirements and criteria that are specified in this regard.

The NRC staff has evaluated the RWS as described in WLS COL FSAR Section 9.2.11. The staff's evaluation focused primarily on confirming that: a) the RWS will not adversely affect safety-related SSCs, or impact the control room occupants; b) the RWS is capable of performing its intended function over the life of the plant; c) the RWS reliance for the support of SWS for achieving and maintaining cold shutdown conditions and RTNSS considerations; and d) the initial test program considerations have been adequately addressed and are appropriate. The RWS was evaluated using the guidance referred to in the regulatory basis section as it pertains to these considerations and acceptability was based upon conformance with the NRC requirements and criteria that are specified in this regard. Based upon the results of this evaluation, the staff concluded that the RWS, as described in WLS COL FSAR Section 9.2.11, is acceptable and all of the FSAR markups included in the above noted RAIs have been adequately incorporated into the FSAR.

9.3 <u>Process Auxiliaries</u>

9.3.1 Compressed and Instrument Air System (Related to RG 1.206, Section C.III.1, Chapter 9, C.I.9.3.1, "Compressed Air Systems")

9.3.1.1 *Introduction*

The compressed and instrument air system delivers instrument air, service air, and high-pressure air. The instrument air subsystem provides high quality instrument air for plant use. The service air subsystem supplies plant breathing air. The high-pressure air subsystem produces air for high-pressure applications.

9.3.1.2 *Summary of Application*

Section 9.3 of the WLS COL FSAR, Revision 11, incorporates by reference Section 9.3 of the AP1000 DCD, Revision 19. Section 9.3 of the AP1000 DCD includes Section 9.3.1.

In addition, in WLS COL FSAR Section 9.3, the applicant provided the following:

<u>Departures</u>

- WLS DEP 6.4-2

The applicant provided additional information in Section 9.3.1.1.2 of the WLS COL FSAR about WLS DEP 6.4-2 related to design changes affecting how the temperature and humidity in the main control room are maintained within the limits for reliable human performance. This information, as well as related WLS DEP 6.4-2 information appearing in other chapters of the FSAR, is reviewed in Section 21.3 of this SER.

AP1000 COL Information Item

- STD COL 9.3-1

The applicant provided additional information in STD COL 9.3-1 to address COL Information Item 9.3-1 (COL Action Item 9.3.1-1).

9.3.1.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the compressed and instrument air system are given in Section 9.3.1 of NUREG-0800.

The regulatory basis for STD COL 9.3-1 addressing Generic Safety Issue (GSI) 43, "Reliability of Air Systems," as part of training and procedures include the following:

- GDC 1, "Quality Standards and Records," as it relates to the reliability of safety-related equipment actuated or controlled by compressed air.

9.3.1.4 *Technical Evaluation*

The NRC staff reviewed Section 9.3.1 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the compressed and instrument air system. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside of the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the BLN Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from Section 9.3.1.4 of the VEGP SER:

> *<u>AP1000 COL Information Item</u>*
>
> - *STD COL 9.3-1 (COL Action Item 9.3.1-1), involving air systems (NUREG-0933, "Resolution of Generic Safety Issues," Issue 43)*
>
> *The NRC staff reviewed STD COL 9.3-1 related to COL Information Item 9.3-1. COL Information Item 9.3-1 states:*
>
>> *The Combined License applicant will address DCD 1.9.4.2.3, Issue 43 as part of training and procedures identified in section 13.5.*
>
> *The commitment was also captured as COL Action Item 9.3.1-1 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:*
>
>> *The COL applicant will address NUREG-0933, Issue 43 as part of training and procedures.*
>
> *The applicant proposed to resolve STD COL 9.3-1 by providing training and procedures for operations and maintenance of the instrument air subsystem and air operated valves. The methodology to develop system operating procedures, abnormal operating procedures, and alarm response procedures is reviewed in Section 13.5 of this SER. The training program for operators and maintenance personnel is reviewed in Section 13.2 of this SER. The applicant also stated that the compressed and instrument air system will be maintained and tested in accordance with the manufacturers' recommendations and procedures and that the system will be periodically tested to demonstrate conformance with the quality requirements of ANSI/ISA-7.3-1981.*
>
> *NUREG-0933, Issue 43 discusses that possible solutions for this issue, include better operator training, operator awareness of the importance of compress air systems, and periodic testing and inspection of the compressed air systems. The NRC staff reviewed the applicant's proposed resolution to STD COL 9.3-1 and determined that the BLN COL FSAR meets the guidance in NUREG-0933, Issue 43; therefore, the staff finds STD COL 9.3-1 resolved.*

9.3.1.5 *Post Combined License Activities*

There are no post-COL activities related to this section.

9.3.1.6 *Conclusion*

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to compressed and instrument air system, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the guidelines given in Section 9.3.1 of NUREG-0800.

- WLS DEP 6.4-2, related to design changes affecting how the temperature and humidity in the main control room are maintained within the limits for reliable human performance, is reviewed and found acceptable by the staff in Section 21.3 of this SER.

- STD COL 9.3-1, the staff evaluated Issue 43, "Reliability of Air Systems," as part of the training and procedures in accordance with the requirements of GDC 1, as it relates to the impact of a failure of the compressed and instrument air system on safety-related SSCs. Based on the results of this evaluation, the WLS COL FSAR meets the guidance in NUREG-0933, Issue 43 and is acceptable.

9.3.2 Plant Gas System (Related to RG 1.206 Section C.III.1, Chapter 9, C.I.9.3.1, "Compressed Air Systems")

The plant gas system is a nonsafety-related system that supplies hydrogen, carbon dioxide, and nitrogen gasses to plant systems as required. Failure of the system does not compromise any safety-related system nor does it prevent safe reactor shutdown.

Section 9.3 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 9.3.2, "Plant Gas System," of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

9.3.3 Primary Sampling System (Related to RG 1.206, Section C.III.1, Chapter 9, C.I.9.3.2, "Process and Postaccident Sampling Systems")

The primary sampling system is used to collect samples during normal operations and following an accident. The system collects for analysis samples from the reactor coolant, auxiliary primary process streams, and containment atmosphere. Both the normal operation and post accident requirements are carried out by this single system.

Section 9.3 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 9.3.3, “Primary Sampling System,” of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff’s review confirmed that there is no outstanding issue related to this section. The results of the NRC staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

9.3.4 Secondary Sampling System (Related to RG 1.206, Section C.III.1, Chapter 9, C.I.9.3.2, “Process and Postaccident Sampling Systems”)

The secondary sampling system delivers representative samples of fluids from secondary systems to sample analyzer packages. Continuous online secondary chemistry monitoring detects impurity ingress and provides early diagnosis of system chemistry excursions in the plant.

Section 9.3 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 9.3.4, “Secondary Sampling System,” of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff’s review confirmed that there is no outstanding issue related to this section. The results of the NRC staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

9.3.5 Equipment and Floor Drainage Systems (Related to RG 1.206, Section C.III.1, Chapter 9, C.I.9.3.3, “Equipment and Floor Drainage System”)

The equipment and floor drainage system collects liquid wastes from equipment and floor drains during normal operation, startup, shutdown, and refueling. The equipment and floor drainage system consists of two subsystems, radioactive waste drains and nonradioactive waste drains.

Section 9.3 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 9.3.5, “Equipment and Floor Drainage Systems,” of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff’s review confirmed that there is no outstanding issue related to this section. The results of the NRC staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

9.3.6 Chemical and Volume Control System (Related to RG 1.206, Section C.III.1, Chapter 9, C.I.9.3.4, “Chemical and Volume Control System (PWR) Including Boron Recovery System”)

The CVS maintains the required water inventory and quality in the RCS, provides pressurizer auxiliary spray, controls the boron neutron absorber concentration in the reactor coolant, provides a means for filling and pressure testing the RCS, controls the primary water chemistry and reduces coolant radioactivity level. Further, the system provides recycled coolant for demineralized water makeup for normal operation and provides borated makeup flow to the RCS in the event of some accidents, such as a small break loss-of-coolant accident.

Section 9.3 of the WLS COL FSAR, Revision 11, incorporates by reference, Section 9.3.6, “Chemical and Volume Control System,” of Revision 19 of the AP1000 DCD. In addition, in the WLS COL FSAR, the applicant provided the following:

Departures

- WLS DEP 7.3-1

The applicant provided additional information in Section 9.3.6 of the WLS COL FSAR about WLS DEP 7.3-1 related to required design changes for the PMS source range neutron flux doubling logic to comply with the requirements of IEEE Std. 603-1991, Clause 6.6. This information, as well as related WLS DEP 7.3-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.5 of this SER.

The NRC staff reviewed Section 9.3.6 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this section. The NRC staff’s review confirmed that the applicant addressed the required information to satisfy the evaluation criteria. There is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

9.4 Air-Conditioning, Heating, Cooling, and Ventilation Systems

9.4.1 Nuclear Island Nonradioactive Ventilation System (Related to RG 1.206, Section C.III.1, Chapter 9, C.I.9.4.1, “Control Room Area Ventilation System”)

9.4.1.1 *Introduction*

The VBS, in conjunction with the MCR emergency habitability system described in Section 6.4, provides a controlled environment for the comfort and safety of control room personnel and assures the operability of control room and nearby components during normal operating, anticipated operational transient, and design-basis accident conditions.

9.4.1.2 *Summary of Application*

Section 9.4 of the WLS COL FSAR, Revision 11, incorporates by reference Section 9.4 of the AP1000 DCD, Revision 19. Section 9.4 of the DCD includes Section 9.4.1, describing the VBS.

In addition, in WLS COL FSAR Sections 9.4.1, 9.4.1.4, and 9.4.12, the applicant provided the following:

Departures

- WLS DEP 6.4-1

The applicant provided additional information in Section 9.4.1 of the WLS COL FSAR about WLS DEP 6.4-1 related to design changes affecting habitability of the main control room and changes to the calculated doses to control room operators. This information, as well as related WLS DEP 6.4-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.2 of this SER.

- WLS DEP 6.4-2

The applicant provided additional information in Section 9.4.1 of the WLS COL FSAR about WLS DEP 6.4-2 related to design changes affecting how the temperature and humidity in the main control room are maintained within the limits for reliable human performance. This information, as well as related WLS DEP 6.4-2 information appearing in other chapters of the FSAR, is reviewed in Section 21.3 of this SER.

AP1000 COL Information Items

- STD COL 9.4-1a

The applicant provided additional information in STD COL 9.4-1a to address the first part of COL Information Item 9.4-1 (COL Action Item 9.4.1-1), related to a program for inspections and testing applicable to the VBS.

In addition, in WLS COL FSAR Section 9.4.12, the applicant provided the following:

- WLS COL 9.4-1b

The applicant provided additional information in WLS COL 9.4-1b to address the second part of COL Information Item 9.4-1 (COL Action Item 6.4-3). The local toxic gas services are evaluated to determine the need for monitoring for control room habitability.

9.4.1.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the VBS are given in Section 9.4.1 of NUREG-0800.

The applicable regulatory guidance for the VBS is as follows:

- RG 1.140, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants,” Revision 2

9.4.1.4 *Technical Evaluation*

The NRC staff reviewed Section 9.4.1 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the VBS. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the BLN Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from Section 9.4.1.4 of the VEGP SER:

> *<u>AP1000 COL Information Items</u>*
>
> - *STD COL 9.4-1a*
>
> *The applicant provided additional information in STD COL 9.4-1a to resolve COL Information Item 9.4-1. COL Information Item 9.4-1a states:*
>
> > *The Combined License applicants referencing the AP1000 certified design will implement a program to maintain compliance with ASME AG-1, ASME N509, ASME N510 and Regulatory Guide 1.140 for portions of the nuclear island nonradioactive ventilation system and the containment air filtration system identified in subsection 9.4.1 and 9.4.7.*

The commitment was also captured as COL Action Item 9.4.1-1 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:

> *The COL applicant will develop a program to maintain operability of the nuclear island nonradioactive ventilation system and the containment air filtration system.*

The NRC staff reviewed STD COL 9.4-1a related to COL Action Item 9.4-1 included under Section 9.4.1.4 of the BLN COL FSAR. The NRC staff reviewed the resolution to STD COL 9.4-1a on the proposed implementation of a program to maintain compliance with industry standards and RGs for the VBS included under Section 9.4.1.4 and Section 9.4.12 of the BLN COL FSAR, and concludes that this item has been resolved for the VBS because the applicant has referenced the applicable regulatory guide and industry standards.

Correction of Error in the Standard Content Evaluation Text

The NRC staff identified an error in the text reproduced above from Section 9.4.1.4 of the BLN SER that requires correction. The BLN SER includes the following statement: "The NRC staff reviewed STD COL 9.4-1a related to COL Action Item 9.4-1 included under Section 9.4.1.4 of the BLN COL FSAR." COL Action Item 9.4-1 does not exist and should be replaced with COL Information Item 9.4-1.

- WLS COL 9.4-1b

The applicant provided additional information in WLS COL 9.4-1b to resolve the second part of COL Information Item 9.4-1. The second part of COL Information Item 9.4-1 states:

> The Combined License applicant will also provide a description of the [Main Control Room/Technical Support Center] MCR/TSC HVAC subsystem's recirculation mode during toxic emergencies, and how the subsystem equipment isolates and operates, as applicable, consistent with the toxic issues, including conformance with Regulatory Guide 1.78 to be addressed by the Combined License applicant as discussed in DCD subsection 6.4.7.

The commitment was also captured as COL Action Item 6.4-3 in Appendix F of NUREG-1793, which states:

> The COL applicant will determine the amount and location of possible sources of toxic chemicals in or near the plant and for seismic Category I Class 1E toxic gas monitoring, using methods discussed in RG 1.78.

The commitment was also captured as COL Action Item 9.4.1-1 in Appendix F of NUREG-1793, which states:

> The COL applicant will develop a program to maintain operability of the nuclear island nonradioactive ventilation system and the containment air filtration system.

The NRC staff review of WLS COL 9.4-1b is addressed in Section 6.4 of this SER.

9.4.1.5 *Post Combined License Activities*

There are no post-COL activities related to this section.

9.4.1.6 *Conclusion*

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to the VBS, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The applicant has provided sufficient information for satisfying Section 9.4.1 of NUREG-0800 and RG 1.140 related to the applicable inspection and testing standards. This addresses STD COL 9.4-1a for VBS. The staff based its conclusion on the following:

- WLS DEP 6.4-1, related to design changes affecting habitability of the main control room and changes to the calculated doses to control room operators, is reviewed and found acceptable by the staff in Section 21.2 of this SER.

- WLS DEP 6.4-2, related to design changes affecting how the temperature and humidity in the main control room are maintained within the limits for reliable human performance, is reviewed and found acceptable by the staff in Section 21.3 of this SER.

- STD COL 9.4-1a, related to a program for inspections and testing applicable to the VBS, is adequately addressed by the applicant and is resolved.

- WLS COL 9.4-1b, addressing the local toxic gas services are evaluated to determine the need for monitoring for control room habitability, is reviewed by the staff in Section 6.4 of this SER.

9.4.2 Annex/Auxiliary Buildings Nonradioactive HVAC System (Related to RG 1.206, Section C.III.1, Chapter 9, C.I.9.4.3, "Auxiliary and Radwaste Area Ventilation System")

The annex/auxiliary building nonradioactive HVAC system maintains ventilation, permits personnel access, and controls the concentration of airborne radioactive material in the nonradioactive personnel and equipment areas, electrical equipment rooms, clean corridors, the ancillary diesel generator room and demineralized water deoxygenating room in the annex building, and the main steam isolation valve compartments, reactor trip switchgear rooms, and piping and electrical penetration areas.

Section 9.4.2 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 9.4.2, "Annex/Auxiliary Buildings Nonradioactive HVAC System," of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

9.4.3 Radiologically Controlled Area Ventilation System (Related to RG 1.206, Section C.III.1, Chapter 9, C.I.9.4.2, "Spent Fuel Pool Area Ventilation System," and C.I.9.4.3, "Auxiliary and Radwaste Area Ventilation System")

The radiologically controlled area ventilation system maintains ventilation permits personnel access, and controls the concentration of airborne radioactive material in the fuel handling area, the radiologically controlled areas of the auxiliary and annex buildings.

Section 9.4 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 9.4.3, "Radiologically Controlled Area Ventilation System," of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

9.4.4 Balance-of-Plant Interface

This section is not applicable to AP1000.

9.4.5 Engineered Safety Features Ventilation System

This section is not applicable to AP1000.

9.4.6 Containment Recirculation Cooling System (Related to RG 1.206, Section C.III.1, Chapter 9, C.I.9.4.5, "Engineered Safety Feature Ventilation System")

The containment recirculation cooling system provides a suitable and controlled environment for the containment building during normal plant operation and shutdown.

Section 9.4 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 9.4.6, "Containment Recirculation Cooling System", of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

9.4.7 Containment Air Filtration System (Related to RG 1.206, Section C.III.1, Chapter 9, C.I.9.4.5, "Engineered Safety Feature Ventilation System")

9.4.7.1 *Introduction*

The containment air filtration system (VFS) serves no safety function, except containment isolation. The system conditions and filters outside air for the containment, the fuel handling area and the other radiologically controlled areas of the auxiliary and annex buildings, except for the hot machine shop and health physics areas, which are served by a separate ventilation system.

9.4.7.2 *Summary of Application*

Section 9.4 of the WLS COL FSAR, Revision 11, incorporates by reference Section 9.4 of the AP1000 DCD, Revision 19. Section 9.4 of the DCD includes Section 9.4.7, "Containment Air Filtration System," which addresses Section 9.4.5, "Engineered Safety Feature Ventilation System," of NUREG-0800.

In addition, in WLS COL FSAR Section 9.4.7.4, the applicant provided the following:

<u>*AP1000 COL Information Item*</u>

- STD COL 9.4-1a

The applicant provided additional information in STD COL 9.4-1a to address COL Information Item 9.4-1 related to a program for inspections and testing applicable to the VFS included under Section 9.4.7.4 of the WLS COL FSAR.

9.4.7.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the VFS are given in Section 9.4.5 of NUREG-0800.

The applicable regulatory guidance for the VFS is as follows:

- RG 1.140

9.4.7.4 *Technical Evaluation*

The NRC staff reviewed Section 9.4.7 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the VFS. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the BLN Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from Section 9.4.7.4 of the VEGP SER:

> *AP1000 COL Information Item*
>
> - *STD COL 9.4-1a*
>
> *The applicant provided additional information in STD COL 9.4-1a to resolve COL Information Item 9.4-1. COL Information Item 9.4-1 states:*
>
> > *The Combined License applicants referencing the AP1000 certified design will implement a program to maintain compliance with ASME AG-1, ASME N509, ASME N510, and Regulatory Guide 1.140 for portions of the nuclear island nonradioactive ventilation system and the containment air filtration system identified in subsection 9.4.1 and 9.4.7. The Combined License applicant will also provide a description of the MCR/TSC HVAC subsystem's recirculation mode during toxic emergencies, and how the subsystem equipment isolates and operates, as applicable, consistent with the toxic issues, including conformance with Regulatory Guide 1.78, to be addressed by the Combined License applicant as discussed in DCD subsection 6.4.7.*
>
> *The commitment was also captured as COL Action Item 9.4.1-1 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:*
>
> > *The COL applicant will develop a program to maintain operability of the nuclear island nonradioactive ventilation system and the containment air filtration system.*
>
> *The NRC staff reviewed STD COL 9.4-1a related to COL Action Item 9.4-1 included under Section 9.4.7.4 of the BLN COL FSAR.*

The NRC staff reviewed the resolution to STD COL 9.4-1a on the proposed implementation of a program to maintain compliance with industry standards and RGs for the VFS included under Section 9.4.7.4 of the BLN COL FSAR, and concludes that this item has been resolved for the VFS because the applicant has appropriately referenced the applicable regulatory guide and industry standards.

Correction of Error in the Standard Content Evaluation Text

The NRC staff identified an error in the text reproduced above from Section 9.4.7.4 of the BLN SER that requires correction. The BLN SER includes the following statement: "The NRC staff reviewed STD COL 9.4-1a related to COL Action Item 9.4-1 included under Section 9.4.7.4 of the BLN COL FSAR." COL Action Item 9.4-1 does not exist and should be replaced with COL Information Item 9.4-1.

9.4.7.5 *Post Combined License Activities*

There are no post-COL activities related to this section.

9.4.7.6 *Conclusion*

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to the VFS, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In conclusion, the applicant has provided sufficient information for satisfying Section 9.4.7 of NUREG-0800 and RG 1.140 related to the applicable inspection and testing standards. This addresses STD COL 9.4-1a for the VFS.

9.4.8 Radwaste Building HVAC System

The radwaste building HVAC system serves the radwaste building, which includes the clean electrical/mechanical equipment room and the potentially contaminated HVAC equipment room, the packaged waste storage room, the waste accumulation room, and the mobile systems facility.

Section 9.4 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 9.4.8, "Radwaste Building HVAC System," of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

9.4.9 Turbine Building Ventilation System

The turbine building ventilation system operates during startup, shutdown, and normal plant operations. The system maintains acceptable air temperatures in the turbine building for equipment operation and for personnel working in the building.

Section 9.4 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 9.4.9, “Turbine Building Ventilation System,” of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff’s review confirmed that there is no outstanding issue related to this section. The results of the NRC staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

9.4.10 Diesel Generator Building Heating and Ventilation System

The diesel generator building heating and ventilation system serves the standby diesel generator rooms, electrical equipment service modules, and diesel fuel oil day tank vaults in the diesel generator building and the two diesel oil transfer modules located in the yard near the fuel oil storage tanks. Local area heating and ventilation equipment is used to condition the air to the stairwell and security room.

Section 9.4 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 9.4.10, “Diesel Generator Building Heating and Ventilation System,” of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff’s review confirmed that there is no outstanding issue related to this section. The results of the NRC staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

9.4.11 Health Physics and Hot Machine Shop HVAC System

The health physics and hot machine shop HVAC system serves the annex building stairwell, S02; the personnel decontamination area, frisking and monitoring facilities, containment access corridor, and health physics facilities on the 100′-0″ elevation of the annex building and the hot machine shop on the 107′-2″ elevation of the annex building.

Section 9.4 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 9.4.11, “Health Physics and Hot Machine Shop HVAC System,” of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff’s review confirmed that there is no outstanding issue related to this section. The results of the NRC staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection System (Related to RG 1.206, Section C.III.1, Chapter 9, C.I.9.5.1, Fire Protection Program)

9.5.1.1 *Introduction*

The FPS provides assurance, through a defense-in-depth philosophy, that the Commission's fire protection objectives are satisfied. These objectives are: 1) to prevent fires from starting; 2) to detect rapidly, control, and extinguish promptly those fires that do occur; and 3) to provide protection for SSCs important to safety so that a fire that is not promptly extinguished by the fire suppression activities will not prevent the safe shutdown of the plant. In addition, FPSs must be designed such that their failure or inadvertent operation does not adversely impact the ability of the SSCs important to safety to perform their safety functions. These objectives are stated in NUREG-0800, Section 9.5.1, "Fire Protection Program," and are identified as the Fire Protection Program goals and objectives in RG 1.189, "Fire Protection for Nuclear Power Plants."

9.5.1.2 *Summary of Application*

Section 9.5 of the WLS COL FSAR, Revision 11, incorporates by reference Section 9.5 of the AP1000 DCD, Revision 19. Section 9.5 of the AP1000 DCD includes Section 9.5.1.

In addition, in WLS COL FSAR Section 9.5.1, the applicant provided the following:

Departures

- WLS DEP 6.3-1

The applicant revised DCD Table 9.5.1-1, "AP1000 Fire Protection Program Compliance with BTP CMEB 9.5-1," Sheet 11 of 33, as new WLS COL FSAR Table 9.5.1-201, providing additional information about WLS DEP 6.3-1 related to quantifying the duration that the passive residual heat removal system heat exchanger can maintain safe shutdown conditions, changing the indefinite duration to greater than 14 days. This information, as well as related WLS DEP 6.3-1 information appearing in other chapters of the WLS COL FSAR, is reviewed in Section 21.1 of the SER.

- WLS DEP 18.8-1

The applicant provided this departure from the AP1000 DCD to address the relocation of the Operations Support Center (OSC). This departure is evaluated in this SER section and in Section 13.3 of this SER.

AP1000 COL Information Items

- STD COL 9.5-1 and STD COL 9.5-3

The applicant provided additional information in STD COL 9.5-1 and STD COL 9.5-3 to resolve COL Information Items 9.5-1 and 9.5-3 (COL Action Item 9.5.1-1(a) through 9.5.1-1(o)) by establishing the site-specific implementation of the fire protection program, including the organization, responsibility, qualification, and training for fire protection program personnel and fire brigade members in Section 9.5.1.8, “Fire Protection Program,” and in Appendix 9A of the WLS COL FSAR.

- STD COL 9.5-4

The applicant provided additional information in STD COL 9.5-4 to resolve COL Information Item 9.5-4 (COL Action Item 9.5.1-5) by establishing Table 9.5-201, “AP1000 Fire Protection Program Compliance with BTP CMEB 9.5-1,” and Table 9.5-202, “Exceptions to NFPA Standard Requirements,” of the WLS COL FSAR.

- STD COL 9.5-8

The applicant provided additional information in STD COL 9.5-8 to resolve COL Information Item 9.5-8 (COL Action Item 9.5.1-3) by establishing an administrative control procedure to address fire barrier breaches.

- STD COL 9.5-6

The applicant provided additional information in STD COL 9.5-6 to resolve COL Information Item 9.5-6 (COL Action Item 9.5.1-6) by specifying a preoperational testing program to verify field installed fire barriers are as tested, and to provide disposition for any deviation.

- WLS COL 9.5-2

The applicant provided additional information in WLS COL 9.5-2 to resolve COL Information Item 9.5-2 (COL Action Item 9.5.1-2) by providing site-specific fire hazard analysis of the yard areas and outlying buildings in WLS COL FSAR Appendix 9A, Section 9A.3.3.

Supplemental Information

- STD SUP 9.5-1

The applicant provided supplemental information in Section 9.5.1.2.1.3, “Fire Water Supply System,” by adding additional text to address the piping threads compatibility requirement between onsite hydrants, hose couplings, and standpipe risers and equipment used by the offsite fire department.

- WLS SUP 9.2-4

The applicant provided supplemental information in Section 9.5.1.2.1.3, “Fire Water Supply System,” by adding additional text to address the makeup water, which is provided to the fire water storage tanks by RWS as described in Section 9.2.11.

License Conditions

- Part 10, License Condition 3, Items C.2, D.1 and G.6

The applicant proposed a license condition in Part 10 of the WLS COL application addressing the Fire Protection Program implementation milestones.

- Part 10, License Condition 6

The applicant proposed a license condition in Part 10 of the WLS COL application to provide a schedule to support the NRC's inspection of operational programs, including the Fire Protection Program.

9.5.1.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the FPS are given in Section 9.5.1 of NUREG-0800.

The regulatory basis and guidance documents for acceptance of STD COL 9.5-1, STD COL 9.5-3, STD COL 9.5-4, STD COL 9.5-6, STD COL 9.5-8, and WLS COL 9.5-2 includes the following:

- RG 1.189, "Fire Protection for Nuclear Power Plants"
- Branch Technical Position (BTP) CMEB 9.5-1, in NUREG-0800, Revision 3
- 10 CFR 50.48, "Fire Protection"

The regulatory guidance for acceptance of STD SUP 9.5-1 and WLS SUP 9.2-2 includes the following:

- RG 1.189

9.5.1.4 *Technical Evaluation*

The NRC staff reviewed Section 9.5.1 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the fire protection system. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the BLN Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced[2] from Section 9.5.1.4 of the VEGP SER:

> *<u>Supplemental Information</u>*
>
> - *STD SUP 9.5-1 provided supplemental information within Section 9.5.1.2.1.3, "Fire Water Supply System," addressing compatibility of piping threads with equipment used by the off-site fire department.*
>
> *The NRC staff reviewed the information on the compatibility of piping threads with off-site equipment included under Section 9.5.1.2.1.3 of the BLN COL, and determined that the applicant conforms to the guidance of RG 1.189. In accordance with the applicant's response to RAI 14.2-9, the requirement to verify fire equipment hose thread compatibility, or alternatively, an adequate supply of readily available thread adapters will be verified. This was added to the Initial Test Program outlined in Section 14.2 of the BLN COL FSAR.*
>
> *<u>AP1000 COL Information Items</u>*
>
> - *STD COL 9.5-1 (COL Action Item 9.5-1(a)), involving qualification requirements for the fire protection program*
>
> *The applicant provided additional information in STD COL 9.5-1 to resolve COL Information Item 9.5-1. COL Information Item 9.5-1 states:*
>
> > *The Combined License applicant will address qualification requirements for individuals responsible for development of the*

[2] Only the BLN SER text relevant to WLS is reproduced here. For example, the BLN SER included a discussion of BLN SUP 9.5-2 after the discussion of STD SUP 9.5-1. Since BLN SUP 9.5-2 does not apply to WLS, it was not reproduced here. Also, the discussion of WLS COL 9.5-2 (corresponds to BLN COL 9.5-2) was moved to the end of this technical evaluation section.

> *fire protection program, training of firefighting personnel, administrative procedures and controls governing the fire protection program during plant operation, and fire protection system maintenance.*

The commitment was also captured as COL Action Item 9.5-1(a) in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:

> *The COL applicant will establish a fire protection program at the facility for the protection of structures, systems, and components (SSCs) important to safety. The COL applicant will also establish the procedures, equipment, and personnel needed to implement the program.*

The NRC staff reviewed the resolution to STD COL 9.5-1 on the qualification requirements for the Fire Protection Program included under Section 9.5.1.6, Section 9.5.1.8, and Section 9.5.1.9 of the BLN COL application, and determined that the above sections provided adequate details to ensure conformance with the regulatory positions contained in RG 1.189 regarding the implementation of the BLN Fire Protection Program. Such details include personnel qualifications and training, organization and responsibilities, fire brigade training, etc.

- *STD COL 9.5-4 (COL Action Item 9.5.1-5), involving NFPA exceptions*

The applicant provided additional information in STD COL 9.5-4 to resolve COL Information Item 9.5-4. COL Information Item 9.5-4 states:

> *The Combined License applicant will address updating the list of NFPA exceptions in the plant-specific DCD, if necessary.*

The commitment was also captured as COL Action Item 9.5.1-5 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:

> *The COL applicant is responsible for ensuring that any deviations from the applicable National Fire Protection Association (NFPA) codes and standards in addition to those in the DCD are incorporated into the final safety analysis report (FSAR) with appropriate technical justification.*

The NRC staff reviewed the resolution to STD COL 9.5-4 under Section 9.5.1.8.1.1 and Section 9.5.1.9.4 of the BLN COL. The applicant provided for BLN COL FSAR Table 9.5-202, Exceptions to NFPA Standard Requirement, to document and justify deviations from applicable NFPA codes and standards in addition to those identified in the DCD. This provision satisfies FSER Action Item 9.5.1-5. The staff also reviewed the exception to NFPA 804 related to the intake structure as documented in Table 9.5-202 although NFPA 804 is not formally endorsed by the NRC as a regulatory guidance document. Since the exception and the provided justification are consistent with the *guidance of RG 1.189, the staff finds it acceptable. Based on the above, the staff concludes that FSER Action Item 9.5.1-5 is resolved.*

- *STD COL 9.5-8 (COL Action Item 9.5.1-3), establishing procedures to minimize risk for fire areas breached during maintenance*

The applicant provided additional information in STD COL 9.5-8 to resolve COL Information Item 9.5-7. COL Information Item 9.5-7 states:

> *The Combined License applicant will establish procedures to minimize risk when fire areas are breached during maintenance. These procedures will address a fire watch for fire areas breached during maintenance.*

The commitment was also captured as COL Action Item 9.5.1-3 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:

> *The COL applicant will establish procedures to address a fire watch for fire areas breached during maintenance.*

The NRC staff reviewed the resolution to STD COL 9.5-8 on the establishment of procedures to minimize risk for fire areas breached during maintenance included under Section 9.5.1.8.1.2 and Section 9.5.1.9.7 of the BLN COL, and determined that the applicant has adequately included a provision to have procedures and administrative controls in place, including fire watches, when fire barriers are breached.

- *STD COL 9.5-6 (COL Action Item 9.5.1-6), involving verification of field installed fire barriers, also designated as a COL information item*

The applicant provided additional information in STD COL 9.5-6 to resolve COL Information Item 9.5-6. COL Information Item 9.5-6 states:

> *The Combined License applicant will address the process for identifying deviations between the as-built installation of fire barriers and their tested configurations.*

The commitment was also captured as COL Action Item 9.5.1-6 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:

> *The COL applicant will establish the process for identifying deviations between the as-built installation of fire barriers and their tested configurations.*

The NRC staff reviewed the resolution to STD COL 9.5-6 under Section 9.5.1.8.6 and Section 9.5.1.9.6. The applicant provided that new installation or modification of fire barriers not part of the AP1000 DCD will be controlled through administrative procedures. These procedures impose inspection and testing requirements to ensure that the as-built fire barrier configurations match tested *configurations. These procedures also describe the process for identifying and dispositioning deviations. Based on the above, the staff concluded that FSER Action Item 9.5.1-6 is resolved.*

- *STD COL 9.5-3 (COL Action Items 9.5.1-1(b) through 9.5.1-1(o)), addressing regulatory conformance*

The applicant provided additional information in STD COL 9.5-3 to resolve COL Information Item 9.5-3. COL Information Item 9.5-3 states:

> *The Combined License applicant will address BTP CMEB 9.5-1 issues. The acronym 'WA' is the identifier in Table 9.5.1-1 for "will address."*

The commitment was also captured as COL Action Items 9.5.1-1(b) through 9.5.1-1(o) in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:

> *9.5.1-1(b) – The COL applicant will implement the fire protection program prior to receiving fuel onsite for fuel storage areas, and for the entire unit prior to reactor startup.*
>
> *9.5.1-1(c) – The COL applicant will establish administrative controls to maintain the performance of the fire protection system and personnel.*
>
> *9.5.1-1(d) – The COL applicant will establish a site fire brigade that is trained and equipped for fire fighting to ensure adequate manual fire fighting capability for all plant areas containing SSCs important to safety.*
>
> *9.5.1-1(e) – The COL applicant will establish a quality assurance (QA) program to ensure that the guidelines for the design, procurement, installation, and testing, as well as the administrative controls for fire protection systems are satisfied.*
>
> *9.5.1-1(f) – The COL applicant is responsible for the inspection and maintenance of fire doors, access to keys for the fire brigade, and the marking of exit routes.*
>
> *9.5.1-1(g) – The COL applicant is responsible for the collection and sampling of water drainage from areas that may contain radioactivity.*
>
> *9.5.1-1(h) – The COL applicant is responsible for controlling the use of compressed gases inside structures.*
>
> *9.5.1-1(i) – The COL applicant is responsible for the use of portable radio communication by the plant fire brigade.*

9.5.1-1(j) – The COL applicant is responsible for fire protection inside containment during refueling and maintenance.

9.5.1-1(k) – The COL applicant is responsible for controlling combustible materials in the remote shutdown workstation.

9.5.1-1(l) – The COL applicant is responsible for fire protection for cooling towers.

9.5.1-1(m) – The COL applicant is responsible for the proper storage of welding gas cylinders.

9.5.1-1(n) – The COL applicant is responsible for the proper storage of ion exchange resins.

9.5.1-1(o) – The COL applicant is responsible for the proper storage of hazardous chemicals.

The NRC staff reviewed the resolution to STD COL 9.5-3 provided in Section 9.5.1.8, Fire Protection Program, and Table 9.5-201 of the BLN COL application. The staff determined that the applicant has incorporated the appropriate portions of RG 1.189 into the BLN Fire Protection Program, pending some changes to be included in Revision 2 to the BLN COL FSAR. The applicant provided the following clarifications related to the BLN Fire Protection Program:

(1) The applicant confirmed that no operator manual actions outside of the Main Control Room are credited or required for post-fire safe shutdown.

(2) The applicant stated that the wireless telephone system is credited as the portable communication system used by the fire brigade. In the applicant's response to RAI 9.5.1-12, the wireless telephone system was confirmed to be designed with multiple antennas (repeaters) throughout the plant to maintain communication capability if individual repeater(s) are damaged from fire. Also, preoperational and periodic testing during fire drills will be performed to verify that the fire brigade portable communication system operates without excessive interference at different locations inside and outside the plant.

(3) In its response to RAI 9.5.1-9, the applicant stated that a housekeeping program is provided in order to maintain cleanliness and minimize fire hazards in the Main Control Room areas.

(4) In its response to RAI 9.5.1-14, the applicant stated that no probabilistic risk assessment (PRA) or fire modeling results will be credited to demonstrate acceptable fire hazards or post-fire safe shutdown capability for specific fire areas or scenarios.

(5) In its response to RAI 9.5.1-15, the applicant confirmed that the supply of reserve air is sufficient to provide at least 6 hours of additional breathing air for "each" of the 10 self-contained breathing apparatus (SCBA) units.

(6) In its response to RAI 9.5.1-16, the applicant proposed a change to BLN COL FSAR Section 9.5.1.8.6 to clarify that testing and inspection of fire protection systems are to be performed per NFPA 25 and NFPA 72 as appropriate. This is ***Confirmatory Item 9.5-1****.*

(7) In its response to RAI 9.5.1-17, the applicant confirmed that the design pressure of the High Pressure Air Subsystem that is used to recharge fire brigade's SCBAs is 4000 psig, and that 2216 psig SCBAs are used to ensure that the cylinders are adequately charged to provide an operating life of at least 30 minutes.

License Conditions

- *License Condition 3, addressing the Fire Protection Program implementation milestones*
- *License Condition 6, addressing the Fire Protection Program implementation schedule*

In Part 10 of the BLN COL FSAR, License Condition 3, "Operational Program Implementation," the applicant proposed a license condition for the implementation of operational programs as described in Table 13.4-201 of the FSAR. This license condition included implementation milestones for the Fire Protection Program, namely D.1 and G.6. Specifically:

- *Milestone D.1 states that the applicable portions of the Fire Protection Program will be implemented prior to initial receipt of fuel onsite.*
- *Milestone G.6 states that the Fire Protection Program will be implemented prior to initial fuel load.*

In Part 10 of the BLN COL FSAR, proposed License Condition 6, "Operational Program Readiness," the applicant states:

> *The licensee shall submit to the appropriate Director of the NRC, a schedule, no later than 12 months after issuance of the COL, that supports planning for and conduct of the NRC inspection of the operational programs listed in the operation program FSAR Table 13.4-201. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until either the operation programs in the FSAR table have been fully implemented or the plant has been placed in commercial service.*

Based on the above, the staff concludes that the applicant satisfied the documentation and implementation requirements for the Fire Protection Program in accordance with RG 1.189 by identifying and providing the implementation schedule for each of the operational program aspects of the Fire Protection Program.

<u>Correction of Error in the Standard Content Evaluation Text</u>

The NRC staff identified an error in the text reproduced above from Section 9.5.1.4 of the BLN SER that requires correction. The BLN SER includes the following statement: "The applicant provided additional information in STD COL 9.5-8 to resolve COL Information Item 9.5-7. COL Information Item 9.5-7 states:" The reference to COL Information Item 9.5-7 should be to COL Information Item 9.5-8.

<u>Resolution of Standard Content Confirmatory Item 9.5-1</u>

To resolve Confirmatory Item 9.5-1, the VEGP applicant revised FSAR Section 9.5.1.8.6 to clarify that procedures governing the inspection, testing, and maintenance of fire protection alarm and detection systems, and water-based suppression and supply systems, use the guidance of NFPA 72, "National Fire Alarm and Signaling Code," and NFPA 25, "Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems," as appropriate. NFPA 25 standard is also added to VEGP COL FSAR Section 9.5.5. The staff determined that these documentation changes satisfy the requirement of standard content Confirmatory Item 9.5-1; therefore Confirmatory Item 9.5-1 is resolved.

<u>Proposed License Condition 3, Item C.2</u>

The VEGP applicant proposed to add another implementation milestone associated with the Fire Protection System to License Condition 3. Specifically, the applicant added Milestone C.2, which states that the applicable portions of the Fire Protection Program will be implemented prior to initial receipt of byproduct, source, or special nuclear materials onsite (excluding Exempt Quantities as described in 10 CFR 30.18). The staff concludes that the applicant satisfied the documentation and implementation requirements for the Fire Protection Program in accordance with RG 1.189 by identifying and providing the implementation schedule for each of the operational program aspects of the Fire Protection Program.

- WLS SUP 9.2-4

The applicant provided supplemental information in Section 9.5.1.2.1.3, "Fire Water Supply System," by adding additional text to address the makeup water, which is provided to the fire water storage tanks by RWS as described in Section 9.2.11.

The applicant stated that the makeup water is filtered, treated, and monitored in the clarification process to prevent or control biofouling or microbiologically induced corrosion. The NRC Staff reviewed the information provided under Section 9.5.1.2.1.3 of the WLS COL regarding the sampling and chemical treatment of the fire water as needed and determined that the applicant conforms to the guidance of RG 1.189.

AP1000 COL Information Items

- WLS COL 9.5-2

The applicant provided additional information in WLS COL 9.5-2 to resolve COL Information Item 9.5-2. COL Information Item 9.5-2 states:

> The Combined License applicant will provide site-specific fire protection analysis information for the yard area, the administration building, and for other outlying buildings consistent with Appendix 9A.

The commitment was also captured as COL Action Item 9.5.1-2 in Appendix F of NUREG-1793, which states:

> The COL applicant will provide site-specific fire protection analysis information for the yard area, the administration building, and other outlying buildings.

The NRC staff reviewed the resolution to WLS COL 9.5-2 on the site-specific fire protection analysis information included under Section 9.5.1.9.2 and Section 9A.3.3 of the WLS COL FSAR, and determined that the yard area, administration building and other outlying areas are adequately described in accordance with RG 1.189 in the fire hazard analysis, which is, therefore, acceptable.

Resolution of VCS DEP 18.8-1

The AP1000 Annex Building does not contain any system or equipment credited for achieving and maintaining post-fire safe shutdown. As such, the relocation of the OSC in the Annex Building as prescribed in WLS DEP 18.8-1 has no adverse impact on the post-fire safe shutdown capability. Therefore, the staff concludes that the proposed departure, relative to post-fire safe shutdown capability, is acceptable.

Resolution of Site-Specific RAIs

In addition to the review of the standard content, the staff also reviewed WLS site-specific content and issued five site-specific RAIs.

In its response to site-specific RAI 9.5.1-1 related to the applicant specifically identifying the engineer in charge of fire protection as responsible for the fire brigade organization, the applicant revised FSAR Section 13.1.2.1.2.9 to state that the engineer in charge of fire protection is responsible for the fire brigade organization. Based on the above, the staff finds that this meets the guidance of RG 1.189 and, therefore, is acceptable.

In its response to site-specific RAI 9.5.1-2 related to the organizational responsibility and lines of communication needed for a successful fire protection program the applicant revised FSAR sections 13.1.1.2.10 and 13.1.1.3.2.2.3 to reflect that the engineer in charge of fire protection and the functional manager of emergency preparedness coordinate and communicate with each other to fulfill their individual fire protection related responsibilities. Based on the above, the staff finds the description of the lines of communications is in accordance with RG 1.189 and, therefore, is acceptable.

In its response to site-specific RAI 9.5.1-3 related to the qualifications of personnel in charge of the fire brigade drills the applicant revised FSAR section 13.1.1.2.10 to state that fire protection trainers are qualified to perform classroom instruction or practical training. Based on the above, the staff finds that this meets the guidance of RG1.189 and, therefore, is acceptable.

In its response to site-specific RAI 9.5.1-4 related to the filtering and treatment of fire water supplies to prevent or control biofouling or microbiologically induced corrosion of the fire water system the applicant revised FSAR Section 9.5.1.2.3 to add WLS SUP 9.2-2 to clarify how fire protection system makeup water quality is monitored and maintained. Additionally, the applicant stated that administrative controls will ensure that the makeup water supply to the fire water storage tanks will be monitored and treated such that the appropriate standards are maintained to prevent or control microbiologically induced corrosion which meets the guidance of RG 1.189. Based on the above, the staff finds that this meets the guidance of RG1.189 and, therefore, is acceptable.

In its response to site-specific RAI 9.5.1-5 related to the qualifications of the engineer in charge of fire protection, the applicant revised FSAR Section 13.1.2.1.2.9 to state that the engineer in charge of fire protection is trained and experienced in nuclear safety or has available personnel who are trained and experienced in nuclear plant safety. Based on the above, the staff finds that this meets the guidance of RG1.189 and, therefore, is acceptable.

9.5.1.5 *Post Combined License Activities*

The license condition language in this section has been clarified from previously considered language. In a letter dated March 22, 2016 (ADAMS Accession No. ML16084A099), the applicant did not identify any concerns with the clarified license condition language. The changes do not affect the staff's above analysis of the conditions, and therefore, for the reasons discussed in the technical evaluation section above, the staff finds the following license conditions acceptable:

- License Condition (9-2) – The licensee shall implement the Fire Protection Program or applicable portions thereof as described in the milestones below:
    1. The fire protection measures in accordance with Regulatory Guide (RG) 1.189 for designated storage building areas (including adjacent fire areas that could affect the storage area) implemented before initial receipt of byproduct or special nuclear materials that are not fuel (excluding exempt quantities as described in 10 CFR 30.18);
    2. The fire protection measures in accordance with RG 1.189 for areas containing new fuel (including adjacent areas where a fire could affect the new fuel) implemented before receipt of fuel onsite;
    3. All fire protection program features implemented before initial fuel load;

- License Condition (9-3) – No later than 12 months after issuance of the COL, the licensee shall submit to the Director of the NRO a schedule that supports planning for and conduct of NRC inspections of the FP Program. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until the FP Program has been fully implemented.

9.5.1.6 *Conclusion*

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to the fire protection system, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the guidance in Section 9.5.1 of NUREG-0800 and RG 1.189. The staff based its conclusion on the following:

- WLS DEP 6.3-1, related to quantifying the duration that the passive residual heat removal system heat exchanger can maintain safe shutdown conditions, is reviewed and found acceptable by the staff in Section 21.1 of this SER.

- STD SUP 9.5-1, addressing compatibility of piping threads with equipment used by the offsite fire department is adequately addressed by the applicant and is resolved.

- STD COL 9.5-1, addressing the qualification and training requirements for the fire protection program at WLS is adequately addressed by the applicant and is resolved.

- STD COL 9.5-4, addressing the deviations from the applicable NFPA codes and standards and to those in the AP1000 DCD is also adequately addressed by the applicant and is resolved.

- STD COL 9.5-6, addressing the establishment of a process for identifying deviations between the as-built installation of fire barriers and their tested configurations is adequately addressed by the applicant and is resolved.

- STD COL 9.5-8, addressing establishment of procedures to minimize risk for fire areas breached during maintenance is adequately addressed by the applicant and is resolved.

- STD COL 9.5-3, addressing the site-specific implementation of the Fire Protection Program is adequately addressed by the applicant and is resolved.

- WLS COL 9.5-2, addressing the site-specific fire protection analysis information for the WLS yard areas and outlying buildings is adequately addressed by the applicant and is resolved.

- WLS DEP 18.8-1, addressing the relocation of the OSC relative to the post-fire safe shutdown capability, is adequately addressed by the applicant and is resolved.

- WLS SUP 9.2-4, addressing the makeup water, which is provided to the fire water storage tanks by RWS, is adequately addressed by the applicant and is resolved.

9.5.2 Communication System

9.5.2.1 *Introduction*

The communication system provides intra-plant communications and plant-to-offsite communications during normal, maintenance, transient, fire, and accident conditions, including loss of offsite power.

9.5.2.2 *Summary of Application*

Section 9.5 of the WLS COL FSAR, Revision 11, incorporates by reference Section 9.5 of the AP1000 DCD, Revision 19. Section 9.5 of the DCD includes Section 9.5.2.

In addition, in WLS COL FSAR Section 9.5.2, the applicant provided the following:

AP1000 COL Information Items

- WLS COL 9.5-9, involving offsite interfaces

The applicant provided additional information in WLS COL 9.5-9 to resolve COL Information Item 9.5-9 (COL Action Item 9.5.2-3).

- WLS COL 9.5-10, involving emergency offsite communications

The applicant provided additional information in WLS COL 9.5-10 to resolve COL Information Item 9.5-10 (COL Action Item 9.5.2-1).

- STD COL 9.5-11, involving security communications

The applicant provided additional information in STD COL 9.5-11 to resolve COL Information Item 9.5-11 (COL Action Item 9.5.2-2).

9.5.2.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the communications system are given in Section 9.5.2 of NUREG-0800.

The regulatory basis for WLS COL 9.5-9, addressing interfaces to offsite locations, is based on:

- Appendix E to 10 CFR Part 50, Part IV.E(9), “Emergency Planning and Preparedness for Production and Utilization Facilities”

The regulatory basis for WLS COL 9.5-10, addressing the emergency offsite communication system, including the crisis management radio system, is based on:

- 10 CFR 50.47(b)(8), “Emergency plans”

The regulatory basis for STD COL 9.5-11, addressing the description of the security communication system is based on:

- 10 CFR 73.45(g)(4)(i), “Performance capabilities for fixed site physical protection systems”
- 10 CFR 73.46(f), “Fixed site physical protection systems, subsystems, components, and procedures”
- 10 CFR 73.55(e), “Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage”
- 10 CFR 73.55(f) “Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors Against Radiological Sabotage”

9.5.2.4 *Technical Evaluation*

The NRC staff reviewed Section 9.5.2 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff’s review confirmed that the information in the application and incorporated by reference addresses the required information relating to the communications system. The results of the NRC staff’s evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, in WLS COL FSAR Section 9.5.2, the applicant provided the following:

<u>*AP1000 COL Information Items*</u>

- WLS COL 9.5-9 (COL Action Item 9.5.2-3) Involving Offsite Interfaces

The applicant provided additional information in WLS COL 9.5-9 to resolve COL Information Item 9.5-9. COL Information Item 9.5-9 states:

> Combined License applicants referencing the AP1000 certified design will address interfaces to required offsite locations; this will include addressing the recommendations of BL-80-15 ([DCD] Reference 21) regarding loss of the emergency notification system due to a loss of offsite power.

The commitment was also captured as COL Action Item 9.5.2-3 in Appendix F of NUREG-1793, which states:

> The COL applicant will address interfaces to offsite locations; this will include addressing the recommendations of NRC Bulletin (BL) 80-15 regarding loss of the emergency notification system as a result of loss of offsite power.

The staff reviewed the resolution to the Lee COL Item 9.5-9 involving offsite interfaces included under Section 9.5.2.2.3.1 and Section 9.5.2.5.1 of the WLS COL FSAR. Section 9.5.2.2.3.1 of the WLS COL FSAR states that the primary means of communication between the station and the NRC is the Emergency Telephone System (ETS). The ETS provides a reliable communication link to the NRC Operations Center. The ETS provides voice and data communication between the station and the NRC headquarters. Calls using the ETS phones are connected directly to Duke's long distance provider over Duke's private, fiber-optic network. Onsite systems supporting the ETS phones are provided with diverse alternate or backup power sources with automatic transfer capability to maintain continuity of communication in the event the normal power source is lost. This aspect of the design is based on the applicant's stated adherence to the guidance provided by NRC Bulletin 80-15 in the FSAR.

In the event of an emergency at the station, notification and activation of the state, local and corporate emergency response network is established. This network requires communication interfaces between the station and the following offsite agencies:

- North Carolina State Emergency Operations Center
- South Carolina Warning Point
- Cherokee County Warning Point
- Cleveland County Warning Point
- York County Warning Point
- Duke Energy Emergency Operating Facility (EOF)

The applicant stated that the primary means of communication between the station and these offsite agencies is the Selective Signal System using Duke telecommunication interfaces to dedicated private lines leased from local telephone companies. The design of the selective signaling system utilizes exiting corporate telecommunications equipment so as to avoid calls being routed through the local telephone company switch. Onsite systems supporting the selective signaling system are provided with sufficient alternate or backup power sources having automatic transfer capability to maintain continuity of communication in the event the normal power source is lost, based upon guidance provided by NRC Bulletin 80-15. The secondary means of communication between the station and various offsite local, state and corporate agencies is provided by commercial telephone lines. The station radio system provides another alternative means of communication between the station and offsite agencies. Communications between the station and offsite radiological monitoring teams is by the radio system as well. The site radio system is powered by non-essential AC sources with built-in battery backups. As an alternative to ground-based communications, in the event of a natural disaster the Lee Station also maintains a satellite phone system. This phone system is portable, self-contained, and intended for use with communications with the NRC.

Appendix E to 10 CFR Part 50, Part IV.E (9) requires at least one onsite and one offsite communications system; each system shall have a backup power source. In addition, NRC Bulletin 80-15 states that the applicant should provide backup power sources for the ENS in case of loss-of-offsite power. With the design of the ETS, Selective Signaling System and radio system the applicant provided adequate means for onsite and offsite communications. In addition, the applicant states that the guidance of NRC Bulletin 80-15 regarding the backup power supplies has been incorporated into the design of the primary and backup power supplies for both onsite and offsite communications, which the staff finds acceptable. The applicant demonstrated sufficient means for onsite and offsite communications, with adequate backup power sources, to meet the requirements of Appendix E to 10 CFR Part 50, Part IV.E(9). Therefore, the staff concludes that COL Action Item 9.5.2-3 has been adequately addressed.

9.5.2.4.2 WLS COL Item 9.5-10 (COL Action Item 9.5.2-1) Involving Emergency Offsite Communications

The applicant provided additional information in WLS COL Item 9.5-10 to resolve COL Information Item 9.5-10. COL Information Item 9.5-10 states:

> "The emergency offsite communication system, including the crisis management radio system, will be addressed by the Combined License applicant."

The commitment was also captured as COL Action Item 9.5.2-1 in Appendix F of the NRC staff's FSER for the AP1000 FSAR (NUREG-1793), which states:

> "The COL applicant will provide a description of the emergency offsite communication system, including the crisis management radio system."

The staff reviewed the resolution to the Lee COL Item 9.5-10 on the emergency offsite communications included under Sections 9.5.2.2.3.2.1, 9.5.2.2.3.2.2, and 9.5.2.5.2 of the WLS FSAR. The ETS is the primary voice and data communication from the main control rooms and Technical Support Center (TSC) to the NRC. The minimum communications links provided for the primary onsite communication center include connections to the NRC ENS, the NRC Health Physics Network (HPN), the emergency response data system (ERDS), and dedicated phone lines for use by NRC personnel for dialing onsite and offsite locations. These dedicated phone lines include:

- Reactor safety counterpart link (RSCL),
- Protective measure counterpart link (PMCL)
- Management counterpart link (MCL)
- Operational center link (OCL)

The dedicated telephones in the ETS use Duke Energy fiber-optic lines to public long distance lines. The secondary means of communication between the station and the NRC are commercial telephone company lines. Sufficient backup or alternate power sources are provided with automatic transfer capability. The primary means of communication between the station and offsite agencies listed in Section 9.5.2.4.1 of this safety evaluation for emergency communication is the Selective Signaling System using private lease lines. The design utilizes existing corporate telecommunications equipment to complete calls without having to go through a local telephone company switch. Sufficient backup or alternate power sources are provided with automatic transfer capability. The secondary means for communication to these offsite agencies are commercial telephone lines.

FSAR Section 9.5.2.2.3.2.2 states that the plant radio system is also provided either through the Duke radio network or the local radio network provided by each offsite emergency agency.

Communication between the station, offsite radiological teams and the EOF can is provided by the radio system. 10 CFR 50.47(b)(8) requires that adequate emergency facilities and equipment to support the emergency response be provided and maintained. The staff finds the offsite communications systems described above and in Section 9.5.2.4.1 of this evaluation are adequate in providing emergency communications equipment and facilities and thus meet the requirements of 10 CFR 50.47(b)(8). The staff finds that the backup radio system adequately serves as the crisis management radio system, and thus the staff concludes that the COL Action Item 9.5.2-1 has been adequately addressed.

- STD COL 9.5-11 (COL Action Item 9.5.2-2) Involving Security Communications

The applicant provided additional information in STD COL 9.5-11 to resolve COL Information Item 9.5-11. COL Information Item 9.5-11 states:

> Specific details for the security communication system are as discussed in separate security documents referred to in Section 13.6.

The commitment was also captured as COL Action Item 9.5.2-2 in Appendix F of NUREG-1793, which states:

> The COL applicant will provide a description of the security communication system.

The staff's review of STD COL 9.5-11 related to security communications is documented in Section 13.6 of this SER.

9.5.2.5 *Post Combined License Activities*

There are no post-COL activities related to this section.

9.5.2.6 *Conclusion*

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff’s review confirmed that the applicant addressed the required information relating to the communication system, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the guidelines given in Section 9.5.2 of NUREG-0800. The staff based its conclusion on the following:

- WLS COL 9.5-9 has been adequately addressed by the applicant in that the onsite and offsite communications interfaces meet the communications requirements of 10 CFR Part 50, Appendix E, Section IV.E(9). In addition, the staff finds the emergency diesel generator capable of providing backup power for the emergency notification system in case of loss of offsite power, and thus meets the guidance in NRC Bulletin 80-15.

- WLS COL 9.5-10 has been adequately addressed by the applicant in that the WLS emergency offsite communications system is capable of providing for notification of personnel and implementation of evacuation procedures in case of emergency and meets the requirements of 10 CFR 50.47(b)(8).

- STD COL 9.5-11, which involves security communications, is documented in Section 13.6 of this SER.

9.5.3 Plant Lighting System (Related to RG 1.206, Section C.III.1, Chapter 9, C.I.9.5.3, "Lighting Systems")

The plant lighting system provides normal, emergency, panel, and security lighting. The normal lighting provides normal illumination during plant operating, maintenance, and test conditions. The emergency lighting provides illumination in areas where emergency operations are performed upon loss of normal lighting. The panel and security lighting is designed to provide the minimum illumination required.

Section 9.5 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 9.5.3, "Plant Lighting System," of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

9.5.4 Diesel Generator Fuel Oil System (Related to RG 1.206, Section C.III.1, Chapter 9, C.I.9.5.4, "Diesel Generator Fuel Oil Storage and Transfer System)

9.5.4.1 *Introduction*

The standby diesel generator fuel oil system maintains the fuel oil system for the diesel engines that provide backup onsite power. This system includes all piping up to the connection to the engine interface, fuel oil storage tanks, fuel oil transfer pumps, day tanks, and the tank storage vaults.

9.5.4.2 *Summary of Application*

Section 9.5 of the WLS COL FSAR, Revision 11, incorporates by reference Section 9.5 of the AP1000 DCD, Revision 19. Section 9.5 of the AP1000 DCD includes Section 9.5.4.

In addition, in WLS COL FSAR Section 9.5.4.5.2, the applicant provided the following:

<u>*AP1000 COL Information Item*</u>

- STD COL 9.5-13

The applicant provided additional information in STD COL 9.5-13 to resolve fuel oil sampling and testing to protect against degradation.

9.5.4.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the diesel generator fuel oil system are given in Section 9.5.4 of NUREG-0800.

9.5.4.4 *Technical Evaluation*

The NRC staff reviewed Section 9.5.4 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the diesel generator fuel oil system. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside of the scope of the DC and use this review in evaluating subsequent COL applications. To ensure the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.
- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.
- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the BLN Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from Section 9.5.4.4 of the VEGP SER:

> *<u>AP1000 COL Information Item</u>*
>
> - *STD COL 9.5-13*
>
> *The applicant provided additional information in STD COL 9.5-13 to resolve COL Information Item 9.5-13. COL Information Item 9.5-13 states:*
>
> > *Address the diesel fuel specifications grade and the fuel properties consistent with manufacturers' recommendations and the measures to protect against fuel degradation by a program of fuel sampling and testing.*
>
> *The commitment was also captured as COL Action Item 9.5.9-2 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:*
>
> > *The COL applicant will develop site-specific factors in the fuel oil storage tank installation specification to reduce the effects of sun heat input into the stored fuel, as well as the diesel fuel specifications grade and fuel properties consistent with manufacturers' recommendations, and will develop a program of fuel sampling and testing to protect against fuel degradation.*
>
> *Revision 17 of the DCD addressed the requirement for limiting heat input by specifying a white epoxy-urethane coating system. Therefore, this information is no longer required from COL applicants.*
>
> *The COL information in Revision 0 of the applicant's FSAR added Section 9.5.4.5.2, "Fuel Oil Quality." The new section addressed fuel quality as follows:*
>
> > *High fuel oil quality is provided by specification of the required grade and properties of the fuel oil for procurement, by testing of samples of new fuel oil prior to addition into the tanks, and by monitoring the fuel oil for contamination and degradation with periodic testing of samples from the storage tanks in accordance with manufacturer's recommendations.*
> >
> > *The fuel oil storage tanks are inspected at least once per 92 days to check for and remove accumulated water.*
> >
> > *The fuel oil quality is verified by sampling and testing from the storage tanks at least once per 92 days. New fuel oil is tested prior to its addition to the storage tanks to verify that the sample meets the following minimum requirements:*
> >
> > - *Water and sediment content of less than or equal to 0.05 volume percent.*

- *Kinematic viscosity at 40°C of greater than or equal to 1.0 mm2/s (1.9 centistokes), but less than or equal to 4.1 mm2/s (4.1 centistokes).*

- *Specific gravity as specified by the manufacturer at 16/16°C (60/60°F), or an API [American Petroleum Institute] gravity at 16°C (60°F), within limits established in accordance with manufacturer's recommendations.*

- *Tested impurity level of less than 2 mg of insolubles per 100 ml. The analysis is completed within 7 days after obtaining the sample, but may be performed after the addition of new oil.*

As a result of the staff's review of BLN COL FSAR Section 9.5.4.5.2, the staff identified two questions that were submitted to the applicant in RAIs.

In RAI 9.5.4-1(a), the staff requested that the applicant identify the controls in place to ensure the fuel oil quality program is implemented according to BLN COL FSAR Section 9.5.4.5.2. In response, the applicant stated that implementation of the fuel oil program according to the FSAR is ensured by the Quality Assurance Program Description (QAPD) described in Chapter 17 and Part 11 of the COL application. The applicant stated QAPD Part III, Section 1, contains quality controls for non-safety-related SSCs that would require and verify implementation of the fuel oil program based on the FSAR description. The staff reviewed the information provided and concludes the proposed quality control requirements can ensure implementation of the fuel oil program in accordance with the BLN COL FSAR.

In RAI 9.5.4-1(b), the staff requested that the applicant provide quality requirements for the periodic testing of stored fuel oil. Section 9.5.4.5.2 of the BLN COL stated that diesel fuel oil from the storage tanks is sampled and tested, but no requirements were listed. The application listed quality requirements that appeared to apply only to new fuel oil. In its response, the applicant proposed the following revised BLN COL FSAR Section 9.5.4.5.2:

> *The diesel fuel oil testing program requires testing both new fuel oil and stored fuel oil. High fuel oil quality is provided by specifying the use of ASTM [American Society for Testing and Materials] Grade 2D fuel oil with a sulfur content as specified by the engine manufacturer.*
>
> *A fuel sample is analyzed prior to addition of ASTM Grade 2D fuel oil to the storage tanks. The sample moisture content and particulate or color is verified per ASTM 4176. In addition, kinetic [sic] viscosity is tested to be within the limits specified in Table 1 of ASTM D975. The remaining critical parameters per Table 1 of ASTM D975 are verified compliant within 7 days.*

> *Fuel oil quality is verified by sample every 92 days to meet ASTM Grade 2D fuel oil criteria. The addition of fuel stabilizers and other conditioners is based on sample results.*
>
> *The fuel oil storage tanks are inspected on a monthly basis for the presence of water. Any accumulated water is to be removed.*

The staff reviewed this revision and finds it acceptable because it addresses both the new and stored fuel oil and the requirements are the manufacturer's specifications and the same ASTM standards applied to safety-related diesel generators. The staff also confirmed that the revised fuel oil testing program was included as shown above in Revision 1 of the BLN COL FSAR.

Correction of Error in the Standard Content Evaluation Text

*The NRC staff identified an error in the text reproduced above from Section 9.5.4.4 of the BLN SER that requires correction. The BLN SER includes the following statement: "In addition, kinetic [sic] viscosity is tested to be within the limits specified in Table 1 of the ASTM D975." The world "kinetic" should read as "kinematic." The staff feels this was a typographical error on the applicant's part because Table 1 of ASTM D975, which is the appropriate reference, specifies "kinematic viscosity." Therefore, the staff concludes that STD COL 9.5-13 has been resolved pending incorporation of the proposed revision in the VEGP COL FSAR, which is being tracked as **Confirmatory Item 9.5-3**.*

Resolution of Standard Content Confirmatory Item 9.5-3

Confirmatory Item 9.5-3 is an applicant commitment to revise its FSAR Section 9.5.4.4 to correct a typographical error. The staff verified that the VEGP COL FSAR was appropriately revised. As a result, Confirmatory Item 9.5-3 is now closed.

9.5.4.5 *Post Combined License Activities*

There are no post-COL activities related to this section.

9.5.4.6 *Conclusion*

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to the standby diesel generator fuel oil system, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR revision 4 is acceptable and meets the guidelines given in Section 9.5.4 of NUREG-0800. The staff based its conclusion on the following:

- STD COL 9.5-13 has been adequately addressed by the applicant in that it ensures that the manufacturers' recommendations using industry standards are met and provides a fuel sampling and testing program to protect against fuel degradation.

9.5.5 Standby Diesel Generator Cooling Water System (Related to RG 1.206, Section C.III.1, Chapter 9, C.I.9.5.5, "Diesel Generator Cooling Water System")

Section 9.5.5 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 9.5.5, "Standby Diesel Generator Cooling Water System," of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

9.5.6 Standby Diesel Generator Starting Air System (Related to RG 1.206, Section C.III.1, Chapter 9, C.I.9.5.6, "Diesel Generator Starting System")

Section 9.5.6 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 9.5.6, "Standby Diesel Generator Starting Air System," of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

9.5.7 Standby Diesel Generator Lubrication System (Related to RG 1.206, Section C.III.1, Chapter 9, C.I.9.5.7, "Diesel Generator Lubrication System")

Section 9.5.7 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 9.5.7, "Standby Diesel Generator Lubrication System," of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

9.5.8 Standby Diesel Generator Combustion Air Intake and Exhaust System (Related to RG 1.206, Section C.III.1, Chapter 9, C.I.9.5.8, "Diesel Generator Combustion Air Intake and Exhaust System")

Section 9.5.8 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 9.5.8, "Standby Diesel Generator Combustion Air Intake and Exhaust System," of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

10.0 STEAM AND POWER CONVERSION

10.1 Summary Description

10.1.1 Introduction

The steam and power conversion (S&PC) system is designed to convert heat energy from the reactor coolant system via the two main steam generators (SGs) and to convert it to electrical power in the turbine-generator (T-G). The main condenser deaerates the condensate and transfers heat that is not used in the cycle to the circulating water system (CWS). The regenerative turbine cycle heats the feedwater, and the main feedwater system returns it to the SG. This section also addresses the materials selection, fabrication, and fracture toughness of the American Society of Mechanical Engineers (ASME) Code Section III, Class 2 and Class 3 pressure boundary components of the steam and feedwater systems and also discusses material issues identified through operating experience.

10.1.2 Summary of Application

Section 10.1 of the William States Lee III Nuclear Station (WLS) combined license (COL) Final Safety Analysis Report (FSAR), Revision 11, incorporates by reference Section 10.1 of the AP1000 Design Control Document (DCD), Revision 19.

In addition, in WLS COL FSAR Section 10.1.3, the applicant provided the following:

AP1000 COL Information Item

- Standard (STD) COL 10.1-1

The applicant provided additional information in STD COL 10.1-1 to address COL Information Item 10.1-1, providing information related to the monitoring of flow-accelerated corrosion (FAC).

License Condition

- Part 10, License Condition 6, Operational Program Readiness

The applicant proposed a license condition to provide a schedule to support the U.S. Nuclear Regulatory Commission's (NRC) inspection of operational programs including the FAC program.

10.1.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793, "Final Safety Evaluation Report [FSER] Related to Certification of the AP1000 Standard Design."

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the FAC program are given in Section 10.3.6 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)."

The applicable regulatory guidance for STD COL 10.1-1 is as follows:

- Generic Letter (GL) 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning"

The staff notes that request for additional information (RAI) numbering was based on NUREG-0800, Section 10.3.6. The evaluation is presented in this section because the applicant provided information in Section 10.1.3 of the WLS COL FSAR.

10.1.4 Technical Evaluation

The NRC staff reviewed Section 10.1 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the S&PC summary description. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this safety evaluation report (SER) provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the design certification (DC) and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (Vogtle Electric Generating Plant (VEGP), Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the Bellefonte Nuclear Plant (BLN), Units 3 and 4 COL application.

[1] See Section 1.2.2 for a discussion of the staff's review related to verification of the scope of information to be included in a COL application that references a design certification (DC).

The following portion of this technical evaluation section is reproduced from Section 10.1.4 of the VEGP SER:

<u>AP1000 COL Information Item</u>

- *STD COL 10.1-1*

The applicant also provided information (STD COL 10.1-1) in BLN COL FSAR Section 10.1.3.1 to address a COL information item as described in AP1000 DCD Section 10.1.3. BLN COL FSAR Section 10.1.3.1, "Erosion-Corrosion Monitoring," describes general attributes of the applicant's program for monitoring and managing degradation (e.g., thinning) of piping and components susceptible to FAC, sometimes called erosion-corrosion.

In AP1000 DCD Section 10.1.3, Westinghouse identified a COL information item on FAC monitoring. The COL information item identified the need for a COL applicant to address the preparation of a FAC monitoring program for carbon steel portions of the S&PC systems that contain water or wet steam in order to address the concerns identified in GL 89-08. Similarly, in the NRC staff's FSER (NUREG-1793), Section 10.3.2, the staff identified COL Action Item 10.3.2-1 for the COL applicant to develop a FAC monitoring program to address industry guidelines and the concerns identified in GL 89-08.

The staff reviewed the information provided by the applicant in Section 10.1.3.1 of the BLN COL FSAR (STD COL 10.1-1) addressing a monitoring program for FAC. The staff also reviewed additional information provided in letters dated June 27, 2008 (ML081830410) and May 26, 2009 (ML091480012). In the letters, the applicant provided additional information requested by the staff about implementation of the FAC program during the plant construction phase, pre-service thickness measurements, and the basis for determining minimum allowable thickness.

In RAI 10.3.6-1, the staff requested that the applicant discuss its implementation schedule for the detailed FAC program (i.e., the FAC program activities that will be conducted during the plant construction phase and the schedule for those activities). This information was not provided in the application and was needed by the staff to make its reasonable assurance finding that the FAC concerns discussed in GL 89-08 are adequately addressed.

In RAI 10.3.6-2, the staff asked the applicant to confirm that its program for addressing and monitoring FAC will include pre-service thickness measurements of as-built components considered susceptible to FAC, and that these measurements will use grid locations and measurement methods most likely to be used for inservice inspection (ISI) according to industry guidelines. In addition, the staff requested that the applicant describe how the pre-service testing requirement was documented in the COL application.

In RAI 10.3.6-3, the staff asked the applicant to identify the industry guidelines or established procedures for determining the minimum allowable wall thickness at which components must be repaired or replaced.

In the June 27, 2008, letter, the applicant responded that susceptibility of piping and components to FAC will be evaluated prior to fuel load as design and as-built information becomes available, and those categorized as high risk for FAC failure will be evaluated for baseline testing prior to startup. For other piping, nominal dimensions may be used until baseline wall thickness is measured, but the applicant did not state when this will occur.

The applicant also proposed revising FSAR Section 10.1.3.1 by deleting the following sentence and replacing it with a paragraph that identifies a specific industry guideline (Electric Power Research Institute (EPRI) NSAC-202L) that contains more details about the approach to FAC monitoring.

> *In addition, the FAC monitoring program considers the information of Generic Letter 89-08 and industry guidelines.*

This revision addressed the staff's concern about the basis for determining the minimum allowable thickness because it references the industry guidance (EPRI NSAC-202L) that addresses the concerns in GL 89-08. The response also addressed the staff's concern about pre-service thickness testing because it affirms the need for pre-service testing, and because the application will reference the guidance of NSAC-202L. The response confirmed that the EPRI CHECWORKS computer program will be used for wall thickness evaluations. Based on operating experience, the staff considers the EPRI guidance document and CHECWORKS program an effective approach to managing FAC. However, the staff also identified open items on this topic as discussed below. The open items are related to information that must be either clarified or added to the COL application.

The response to RAI 10.3.6-1 described how susceptibility to FAC will be evaluated as the design and as-built information becomes available, and high-risk (of FAC) components will be evaluated for baseline testing prior to startup. The staff had the following concerns:

a) The applicant stated that piping and/or components with a high risk of FAC failure will be "evaluated for baseline testing prior to startup." This statement suggests baseline testing may not be performed on high-risk components.

b) The reference to piping and/or components "deemed to have a high risk of failure due to FAC" led the staff to question the extent to which FAC prevention was included in the plant design. Given that the plant has not yet been constructed and a predictive model such as CHECWORKS can estimate FAC rates, it is the staff's understanding that materials susceptible to FAC can be avoided where FAC is a potential degradation mechanism.

c) The applicant did not add the FAC program implementation schedule and construction phase activities to the COL application.

The response to RAI 10.3.6-2 and the associated COL application revisions include the terms "Pass 1 analysis" and "Pass 2 analysis." Since these are terms defined in EPRI NSAC-202L in the context of the CHECWORKS analysis program, reference to CHECWORKS needs to be addressed in the application.

The response to RAI 10.3.6-3 refers to "Systems Not Modeled components." Based on the context of this statement, the staff understands that this statement refers to "Susceptible Not Modeled lines," as discussed in EPRI NSAC-202L.

The applicant submitted a supplemental RAI response dated May 26, 2009 (ML091480012). In the revised responses to the RAIs the applicant clarified that the plant is designed to prevent FAC, and no piping/components are expected to have a high risk of FAC failure, but the possibility of a high-risk piping/component cannot be ruled out until the as-built design is analyzed. The response also clarified that baseline testing would be performed on all high-risk piping/components, and it corrected the wording to reference "Susceptible-Not-Modeled" lines. In the response to RAI 10.3.6-2 the applicant also proposed the following revision to FSAR Section 10.1.3.1:

> *In addition, the FAC monitoring program considers the information of Generic Letter 89-08, EPRI NSAC-202L-R3, and industry operating experience. The program requires a grid layout for obtaining consistent pipe thickness measurements when using Ultrasonic Test Techniques. The FAC program obtains actual thickness measurements for highly susceptible FAC locations for new lines as defined in EPRI NSAC-202L-R3. At a minimum, a CHECWORKS type Pass 1 Analysis is used for low susceptible FAC locations and a CHECWORKS type Pass 2 Analysis for highly susceptible FAC locations will be considered. To determine wear of piping and components where operating conditions are inconsistent or unknown the guidance provided in EPRI NSAC-202L is used to determine wear rates.*

The revised response to RAIs 10.3.6-1, 10.3.6-2, and 10.3.6-3 therefore addressed all of the concerns identified above, with the exception of identifying the program implementation schedule in the application. This is ***Open Item 10.1-1****. The staff identifies the FSAR revisions proposed by the applicant in its May 26, 2009 letter as* **Confirmatory Item 10.1-1***. Pending resolution of the open item and confirmatory item, the staff finds the COL information item on the FAC program addresses the concerns expressed in GL 89-08.*

Resolution of Standard Content Open Item 10.1-1

In a letter dated July 16, 2009, the VEGP applicant addressed Open Item 10.1-1 by proposing to include the FAC program as part of License Condition 6, "Operational Program Readiness." Specifically, the applicant stated that in a future application revision License Condition 6 will include the requirement to submit a FAC program implementation schedule, including the construction phase activities. The proposed license condition is consistent with SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria." The staff verified that this change was incorporated into Revision 2 of the COL application. As a result, Open Item 10.1-1 is resolved.

Resolution of Standard Content Confirmatory Item 10.1-1

In a letter dated September 9, 2009, the BLN applicant revised the May 26, 2009, response to RAI 10.3.6-2 related to preservice inspection. The letter clarified that the CHECWORKS Pass 1 analysis (corrosion rates based on the plant model) would be performed for locations with both low and high FAC susceptibility. In addition, the response stated that the Pass 2 analysis (use of inspection data for model refinement, corrosion measurement, and trending) will be performed for high-susceptibility locations if warranted by the Pass 1 analysis. The original response stated that the Pass 2 analysis "will be considered" for high-susceptibility locations. The response includes the following revised wording in FSAR Section 10.1.3.1:

> *The FAC program obtains actual thickness measurements for highly susceptible FAC locations for new lines as defined in EPRI NSAC-202L-R3 (Reference 201). At a minimum, a CHECWORKS type Pass 1 analysis is used for low and highly susceptible FAC locations and a Pass 2 analysis is used for highly susceptible FAC locations when Pass 1 results warrant.*

The staff determined that this revised FSAR text is acceptable because it clarified how the plant predictive model is used to perform FAC analysis, and the approach conforms to the EPRI NSAC-202L guidelines. The VEGP applicant has endorsed the standard RAI responses, and has incorporated the associated changes into Revision 2 of the FSAR. The staff determined that the VEGP applicant has fully addressed all RAI responses, and as a result, Confirmatory Item 10.1-1 is now resolved.

10.1.5 Post Combined License Activities

For the reasons discussed in the technical evaluation section above, the staff finds the following license condition acceptable:

- License Condition (10-1) – Prior to initial fuel load, the licensee shall implement the flow accelerated corrosion (FAC) program including construction phase activities. No later than 12 months after issuance of the COL, the licensee shall submit to the Director of the Office of New Reactors (NRO) a schedule that supports planning for and conduct of NRC inspections of the FAC program implementation including construction phase activities. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until the FAC program has been fully implemented.

10.1.6 Conclusion

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to FAC, and there is no outstanding information to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff concludes that the information presented in the WLS COL FSAR is acceptable because it meets the acceptance criteria provided in Section 10.3.6 of NUREG-0800 and the guidance in GL 89-08. The staff based its conclusion on the following:

- STD COL 10.1-1, relating to the monitoring of the FAC program, is acceptable because it conforms to the acceptance criteria and guidelines provided under Section 10.3.6 of NUREG-0800 and GL 89-08.

10.2 <u>Turbine-Generator</u>

10.2.1 Introduction

The T-G includes the turbine generator system (TGS), associated equipment (including moisture separation), use of extraction steam for feedwater heating, and control functions. Details of TGS component construction materials are included in the AP1000 DCD. The T-G control and overspeed system is described in detail in the DCD; including redundancy and diversity of controls, types of control utilized, overspeed setpoints, and valve actions required for each set point. Because turbine rotors have large masses and rotate at relatively high speeds during normal reactor operation, failure of a rotor may cause excessive vibration of the turbine rotor assembly and result in the generation of high energy missiles. Measures taken by the applicant to ensure turbine rotor integrity and reduce the probability of turbine rotor failure are included in this section of the application.

10.2.2 Summary of Application

Section 10.2 of the WLS COL FSAR, Revision 11, incorporates by reference Section 10.2 of the AP1000 DCD, Revision 19. In addition, in WLS COL FSAR Section 10.2, the applicant provided the following:

Supplemental Information

- STD Supplement (SUP) 10.2-1

The applicant provided supplemental information in WLS COL FSAR Section 10.2.2, "System Description," which describes the probability of generating a turbine missile.

- STD SUP 10.2-2

In Revision 0 of the WLS COL FSAR, the applicant provided supplemental information regarding the main steam stop and control valves. This supplemental information was deleted in a later revision of the WLS COL FSAR; this is discussed in Section 10.2.4 (Technical Evaluation) of this SER.

- STD SUP 10.2-3

The applicant provided supplemental information in WLS COL FSAR Section 10.2.3.6, "Maintenance and Inspection Program Plan," which describes the ISI program for the turbine assembly.

- STD SUP 10.2-4

The applicant provided supplemental information in WLS COL FSAR Section 10.2.2, "System Description," which describes the turbine assembly preoperational and startup tests.

- STD SUP 10.2-5

The applicant provided supplemental information in WLS COL FSAR Section 10.2.3, "Turbine Rotor Integrity," which describes the turbine assembly operations and maintenance procedures.

AP1000 COL Information Item

- STD COL 10.2-1

The applicant provided additional information in STD COL 10.2-1, which states that a turbine maintenance and inspection program will be submitted to the NRC for review prior to initial fuel load. This addresses the COL information item in Section 10.2.6, "Combined License Information on Turbine Maintenance and Inspection," of the AP1000 DCD (COL Action Item 10.5-2).

License Condition

- License Condition 2, Item 10.2-1, relating to the turbine maintenance and inspection program

10.2.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for turbine rotor integrity are given in Sections 10.2 and 10.2.3 of NUREG-0800.

10.2.4 Technical Evaluation

The NRC staff reviewed Section 10.2 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the T-G. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the BLN Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from Section 10.2.4 of the VEGP SER:

<u>Supplemental Information</u>

- *STD SUP 10.2-1*

The applicant provided supplemental information as part of the BLN COL FSAR regarding the probability of generating a turbine missile. In FSAR Section 10.2.2, "System Description," the applicant stated that Section 3.5.1.3 addresses the probability of generation of a turbine missile for AP1000 plants in a side-by-side configuration. The staff's review of the acceptability of the probability of generating a turbine missile is documented in Section 3.5.1, "Missile Selection and Description," of this SER.

- *STD SUP 10.2-2*

In Revision 0 of the BLN COL FSAR, the applicant provided supplemental information regarding the frequency for exercising the main steam stop and control valves. However, the valve exercise frequency is specified in Revision 17 of the DCD, and therefore, this supplemental information is no longer necessary. In Revision 1 of BLN COL FSAR, this information is no longer provided.

- *STD SUP 10.2-3*

The applicant provided supplemental information as part of the BLN COL FSAR regarding the ISI program for the turbine assembly. The applicant added text to the end of Section 10.2.3.6 of the AP1000 DCD, Revision 17, to describe the breadth of the turbine assembly ISI program.

The NRC staff reviewed the standard supplemental information provided in STD SUP 10.2-3 regarding the text added to Section 10.2.3.6 related to the turbine assembly ISI program. The staff concludes that STD SUP 10.2-3 is acceptable because it is a statement of the scope of the turbine ISI program consistent with the acceptance criteria of Section 10.2.3 of NUREG-0800.

- *STD SUP 10.2-4*

The applicant provided supplemental information as part of the FSAR regarding the turbine assembly preoperational and startup tests. The NRC staff reviewed the standard supplemental information provided in STD SUP 10.2-4 regarding the text added to Section 10.2.2 related to the turbine assembly preoperational and startup testing. The staff determined that this additional information provides further clarity regarding the turbine system startup tests. This additional information does not affect the design aspects of the system or its regulatory basis.

- *STD SUP 10.2-5*

The applicant provided supplemental information as part of the BLN COL FSAR regarding turbine assembly operations and maintenance procedures. The *applicant added text to the end of Section 10.2.3 of the AP1000 DCD, Revision 17, to note that operations and maintenance procedures mitigate potential degradation mechanisms in the turbine rotor and buckets/blades. STD SUP 10.2-5 is a general statement about the purpose of operations and maintenance procedures and does not affect those procedures that are part of the staff's review of Section 10.2.3 of the DCD application.*

<u>AP1000 COL Information Item</u>

- *STD COL 10.2-1*

The applicant provided additional information (STD COL 10.2-1) in BLN COL FSAR Section 10.2.6, "Combined License Information on Turbine Maintenance and Inspection," to resolve a COL information item identified in AP1000 DCD, Section 10.2.6. STD COL 10.2-1 identifies the turbine maintenance and inspection program, plant-specific turbine rotor test data, and plant-specific calculated toughness curves as items that must be submitted by the COL holder to the NRC staff for review prior to fuel load.

The AP1000 COL information item identified in DCD Section 10.2.6 states:

> *The Combined License holder will submit to the NRC staff for review prior to fuel load and then implement a turbine maintenance and inspection program. The program will be consistent with the maintenance and inspection program plan activities and inspection intervals identified in Subsection 10.2.3.6. The Combined License holder will have available plant-specific turbine rotor test data and calculated toughness curves that support the material property assumptions in turbine rotor analysis after the fabrication of the turbine and prior to fuel load.*

BLN COL FSAR Section 10.2.6, "Combined License Information on Turbine Maintenance and Inspection," replaces Section 10.2.6 of the AP1000 DCD with the following:

> *A turbine maintenance and inspection program will be submitted to the NRC staff for review prior to fuel load. The program will be consistent with the maintenance and inspection program plan activities and inspection intervals identified in DCD Subsection 10.2.3.6. Plant-specific turbine rotor test data and calculated toughness curves that support the material property assumptions in the turbine rotor analysis will be available for review after fabrication of the turbine and prior to fuel load.*

The applicant proposed License Condition 2, Item 10.2-1 related to the above. The staff is currently reviewing Revision 17 of the DCD which contains the turbine maintenance and inspection program elements. License Condition 2 provides that the applicant will submit, prior to fuel load, its turbine maintenance *and inspection program for the as-built rotor, including its material properties. The staff finds this condition acceptable because the inspection program, updated with as-built information, will be submitted to verify consistency with the maintenance and inspection program plan activities and inspection intervals identified in Section 10.2.3.6 of the DCD.*

10.2.5 Post Combined License Activities

For the reasons discussed in the technical evaluation section above, the staff finds the following license condition acceptable:

- License Condition (10-2) – Prior to initial fuel load, the licensee shall implement a turbine maintenance and inspection program, which will be consistent with the maintenance and inspection program plan activities and inspection intervals identified in FSAR Section 10.2.3.6. No later than 12 months after issuance of the COL, the licensee shall submit to the Director of NRO a schedule that supports planning for and conduct of NRC inspections of the turbine maintenance and inspection program. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until the turbine maintenance and inspection program has been fully implemented.

10.2.6 Conclusion

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to the T-G, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the acceptance criteria of Section 10.2 of NUREG-0800. The staff based its conclusions on the following:

- STD SUP 10.2-1, related to the probability of generating a turbine missile, is reviewed by the staff in Section 3.5.1, "Missile Selection and Description," of this SER.

- STD SUP 10.2-2, related to frequency for exercising the main steam stop and control valves, was deleted in Revision 1 of the WLS COL FSAR.

- STD SUP 10.2-3, related to the ISI program for the turbine assembly, is acceptable to the staff because the description of the ISI program is consistent with Section 10.2.3 of NUREG-0800.

- STD SUP 10.2-4, relating to the turbine assembly preoperational and startup tests, is acceptable to the staff because the proposed valve testing is consistent with the guidance in Section 10.2 of NUREG-0800.

- STD SUP 10.2-5, relating to mitigation of potential degradation mechanisms for the turbine rotor and buckets/blades, is acceptable to the staff because it is a general statement about the purpose of operations and maintenance procedures and does not affect those procedures that are part of the staff's review of Section 10.2.3 of the DCD application.

- STD COL 10.2-1, relating to the turbine maintenance and inspection program, is acceptable to the staff because the applicant proposed a license condition that appropriately addresses this information item.

10.3 Main Steam Supply System

10.3.1 Introduction

The main steam supply system (MSSS) transports the steam generated by the nuclear steam supply system to the S&PC system and various safety-related and nonsafety-related auxiliaries. Portions of the MSSS may be used as part of the heat sink that removes heat from the reactor facility during certain operations. The MSSS for the pressurized-water reactor (PWR) plant extends from the connections to the secondary sides of the SGs up to and including the turbine stop valves.

10.3.2 Summary of Application

Section 10.3 of the WLS COL FSAR, Revision 11, incorporates by reference Section 10.3 of the AP1000 DCD, Revision 19.

In addition, in WLS COL FSAR Section 10.3, the applicant provided the following:

Supplemental Information

- STD SUP 10.3-1

The applicant provided supplemental information in WLS COL FSAR Section 10.3.2.2.1, "Main Steam Piping," which addresses operations and maintenance procedures.

- STD SUP 10.3-2

The applicant provided supplemental information in WLS COL FSAR Section 10.3.5.4, "Chemical Addition," related to secondary-side water chemistry.

- STD SUP 10.3-3

The applicant provided supplemental information in WLS COL FSAR Section 10.3.6.2, "Material Selection and Fabrication," which addresses intergranular stress corrosion cracking (IGSCC).

10.3.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the MSSS are given in Sections 10.3.1 and 10.3.6 of NUREG-0800.

The applicable regulatory requirements and guidance for STD SUP 10.3-1, STD SUP 10.3-2, and STD SUP 10.3-3 are as follows:

- General Design Criterion (GDC) 4, "Environmental and Dynamic Effects Design Bases"
- Regulatory Guide (RG) 1.37, Revision 1, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants"
- Branch Technical Position (BTP) 5-1, "Monitoring of Secondary Side Water Chemistry in PWR Steam Generators"

The regulatory basis for acceptance of the supplemental information on controls to prevent stress-corrosion cracking of stainless steels and nickel alloys is the quality assurance requirements in Appendix B, "Quality assurance criteria for nuclear power plants and fuel reprocessing plants," of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic licensing of production and utilization facilities," and the guidance in RG 1.37, as they relate to quality assurance requirements for the design, fabrication, and construction of safety-related structures, systems, and components (SSCs).

10.3.4 Technical Evaluation

The NRC staff reviewed Section 10.3 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the MSSS. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the BLN Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from Section 10.3.4 of the VEGP SER:

> *Supplemental Information*
>
> - *STD SUP 10.3-1*
>
> *The applicant provided additional information as part of the BLN COL FSAR regarding operations and maintenance procedures. The applicant added text to Section 10.3.2.2.1 of the AP1000 DCD, Revision 17, to address steam hammer and relief valve discharge reaction loads.*
>
> *The NRC staff reviewed the standard supplemental information provided in STD SUP 10.3-1 regarding the text added to Section 10.3.2.2.1 related to MSSS operations and maintenance procedures.*
>
> *During its review of Revision 0 of the BLN COL FSAR, the staff did not find any further details regarding these procedures. Therefore, the staff raised a concern regarding the adequacy of these procedures. Also, Section 10.3 of NUREG-0800, "MAIN STEAM SUPPLY SYSTEM," Item II, related to GDC 4, describes that the main steam system should adequately consider water (steam) hammer and relief valve discharge loads to assure that system safety functions can be performed and should assure that operating and maintenance procedures include adequate precautions to prevent water (steam) hammer and relief valve loads. In order to ensure the adequacy of the MSSS and its agreement with the NUREG-0800 criteria, the staff requested the key elements of the procedures for staff's review in RAI 10.3-1.*
>
> *In its response, dated July 21, 2008, concerning precluding or mitigating water hammer events, the applicant identified that good operating practice and operating experience including, but not limited to Institute of Nuclear Power Operations (INPO) significant event reports and significant operating event reports, NRC information notices and bulletins, and other industry operating experience information are programmatically integrated into the AP1000 Operations Procedure development. The applicant also stated that specific operating experience to preclude or mitigate water hammer is included in this*

population of operating experience. In addition, the applicant explained that the AP1000 has been designed to prevent or minimize steam and water hammer. The applicant stated that BLN COL FSAR Section 10.3.2.2.1 will be revised to include additional precautions, when appropriate, to minimize the potential for steam and water hammer.

With respect to the relief valve discharge loads, in its response, the applicant explained that Westinghouse addressed these loads for main steam safety valves in the AP1000 DCD, Section 10.3.2.2.2, "Main Steam Safety Valves," which BLN incorporated by reference with no departures and supplements. Further, the applicant stated that as described in NUREG-0927, Revision 1, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants," preventive measures for relief valve loading are addressed by design. Therefore, the applicant stated that the COL application Part 2, BLN COL FSAR Section 10.3.2.2.1 will be revised to remove the associated procedure precautions as related to the relief valve discharge reaction loading. In addition, Section 10.3.2.2.1 will be revised to state that operations and maintenance procedures include precautions, when appropriate, to minimize the potential for steam and water hammer. The applicant listed several precautionary items, such as: prevention of rapid valve motion, process for avoiding voids and flashing in water-filled lines and venting these lines, process for avoiding introduction of water into steam lines and proper warm-up and drainage of these lines, and effects of valve alignments on line conditions.

Based on its review, the staff finds the applicant's response acceptable because a detailed list of the procedural precautions (identified above) is provided and included as a proposed revision to COL application Part 2, BLN COL FSAR Section 10.3.2.2.1. The staff reviewed the precautions and compared them to the industry experience and staff guidance, and finds that they adequately address steam and water hammer. Therefore, the staff agrees that the deletion of the relief valve discharge reaction load occurrences from BLN COL FSAR Section 10.3.2.2.1 is acceptable, because its discussion was already identified in the AP1000 DCD Section 10.3.2.2.1. In BLN COL FSAR Section 10.3.2.2.1, Revision 1, the applicant revised STD SUP 10.3-1 as indicated above in its response to RAI 10.3-1. Therefore, the staff's concern in RAI 10.3-1 is resolved.

- *STD SUP 10.3-2*

The applicant provided additional information as part of the BLN COL FSAR regarding the secondary chemistry. In FSAR Section 10.3.5.4, "Chemical Addition," the applicant proposed adding the following at the end of DCD Subsection 10.3.5.4:

> *Alkaline chemistry supports maintaining iodine compounds in their nonvolatile form. When iodine is in its elemental form, it is volatile and free to react with organic compounds to create organic iodine compounds, which are not assumed to remain in solution. It is noted that no significant level of organic compounds is expected in*

> *the secondary system. The secondary water chemistry, thus, does not directly impact the radioactive iodine partition coefficients.*

The staff reviewed the secondary water chemistry under Section 10.4.6 of this SER and found it acceptable with respect to the EPRI PWR Secondary Water Chemistry Guidelines. As discussed in Section 10.4.6, the staff considers application of the guidance of the EPRI PWR Secondary Water Chemistry Guidelines, and a programmatic commitment to use these guidelines, to be an acceptable method for the applicant to ensure compliance with GDC 14 as it relates to ensuring the integrity of the reactor coolant boundary (specifically, as the secondary water chemistry program ensures the integrity of the SG tubing). As the applicant stated in STD SUP 10.3-2, the secondary water chemistry does not directly impact the iodine partition coefficients. In addition, radioactive iodine is not a consideration in the EPRI Secondary Water Chemistry Guidelines. The staff finds that STD SUP 10.3-2 is a statement of fact that does not affect the staff's review. The management of radioactive compounds, including iodine, is addressed by the staff in Chapter 11.

- *STD SUP 10.3-3*

The applicant provided additional information as part of the BLN COL FSAR regarding IGSCC. The applicant added text to the end of Section 10.3.6.2 "Material Selection and Fabrication" of the AP1000 DCD, Revision 17, to include providing the necessary controls to minimize the susceptibility of components made of stainless steel and nickel-based materials to IGSCC. The applicant proposed adding the following at the end of DCD Section 10.3.6.2:

> *Appropriate operations and maintenance procedures provide the necessary controls during operation to minimize the susceptibility of components made of stainless steel and nickel-based materials to IGSCC by controlling chemicals that are used on system components.*

The staff finds the supplemental information, addressing IGSCC concerns related to stainless steels and nickel-base alloys, acceptable because the AP1000 DCD meets the technical guidelines specified in RG 1.37. In addition, the staff notes that these materials are not proposed for use in the main steam and feedwater piping systems at BLN Units 3 and 4.

<u>*Correction of Error in the Standard Content Evaluation Text*</u>

The NRC staff identified an error in the text reproduced above from the BLN SER, Section 10.3.4, that requires correction. The BLN SER states that the staff reviewed the secondary water chemistry in Section 10.4.6 of the SER. Secondary water chemistry is actually reviewed in Section 10.4.7 of the SER.

10.3.5 Post Combined License Activities

There are no post-COL activities related to this section.

10.3.6 Conclusion

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to MSSS, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the requirements of Appendix A to 10 CFR Part 50, GDC 4, 10 CFR 52.79, "Contents of applications; technical information in final safety analysis report," and conforms to the guidance in Sections 10.3 and 10.3.6 of NUREG-0800, BTP 5-1, and RG 1.37. The staff based its conclusions on the following:

- STD SUP 10.3-1, relating to operations and maintenance procedures, is acceptable because the applicant provided sufficient information to satisfy GDC 4 as related to MSSS design considering the water (steam) hammer effects on the safety-related SSCs.

- STD SUP 10.3-2, relating to secondary chemistry, is a statement of fact that does not affect the staff's review.

- STD SUP 10.3-3, relating to IGSCC, is acceptable to the staff because the AP1000 DCD meets the technical guidelines specified in RG 1.37.

10.4 Other Features of Steam and Power Conversion System

10.4.1 Main Condensers

During normal operation, the main condenser receives, condenses and deaerates exhaust steam from the main turbine and the turbine bypass system whenever the turbine bypass system is operated. The main condenser is also a collection point for other steam cycle miscellaneous drains and vents.

Section 10.4 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 10.4.1 of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

10.4.2 Main Condenser Evacuation System

10.4.2.1 *Introduction*

Main condenser evacuation is performed by the condenser air removal system. The system removes noncondensable gases and air from the main condenser during plant startup, cooldown, and normal operation. This action is performed by liquid ring vacuum pumps.

10.4.2.2 *Summary of Application*

Section 10.4 of the WLS COL FSAR, Revision 11, incorporates by reference Section 10.4 of the AP1000 DCD, Revision 19. Section 10.4 of the DCD includes Section 10.4.2.2.

In addition, in WLS COL FSAR Section 10.4.2.2, the applicant provided the following:

<u>*Site-Specific Information Replacing Conceptual Design Information*</u>

- WLS CDI

The applicant provided additional information to replace conceptual design information (CDI) in WLS COL FSAR Section 10.4.2.2.1, "General Description," which describes the plant-specific cooling water source for the vacuum pump seal water heat exchangers.

The applicant also provided additional information to replace CDI in WLS COL FSAR Section 10.4.2.2.2, "Component Description," which describes the plant-specific tube side water flow in the seal water heat exchangers.

10.4.2.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

Additional regulatory basis is Appendix A to 10 CFR Part 50 and GDC 60, "Control of Releases of Radioactive Materials to the Environment."

Acceptance criteria associated with the relevant requirements of the Commission regulations for the main condenser evacuation system are given in Section 10.4.2 of NUREG-0800.

10.4.2.4 *Technical Evaluation*

The NRC staff reviewed Section 10.4.2 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the main condenser evacuation system. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the following WLS plant-specific design information that replaces the CDI identified in the AP1000 DCD:

Site-Specific Information Replacing Conceptual Design Information

- WLS CDI

The WLS plant-specific design information was annotated as "WLS CDI" in WLS COL FSAR Section 10.4.2.2. In this section, the applicant replaced bracketed (conceptual design) text in Sections 10.4.2.2.1, "General Description," and 10.4.2.2.2, "Component Description," of the AP1000 DCD to provide specific information regarding the sources of cooling water for the vacuum pump seal water heat exchangers.

The WLS CDI in WLS COL FSAR Section 10.4.2.2.1 is related to the CWS and raw water system (RWS) supplying cooling water for the main condenser vacuum pump seal water heat exchangers. The WLS CDI in FSAR Section 10.4.2.2.2 clarifies that the seal water flows through the shell side of the seal water heat exchanger and CWS water flows through the tube side. Based on its review, the staff concludes that this WLS plant-specific design information will have no adverse effects on the capability of the main condenser evacuation system, CWS, or RWS and associated equipment. Also, the staff concludes that adding this WLS plant-specific design information will not affect the functions of any safety-related equipment, components, or systems of the plant. The staff accepts these revisions as stated, because the information provided in this WLS CDI meets the acceptance criteria in Section 10.4.2 of NUREG-0800, and therefore, meets GDC 60 as it relates to the main condenser evacuation system design for the control of releases of radioactive materials to the environment.

10.4.2.5 *Post Combined License Activities*

There are no post-COL activities related to this section.

10.4.2.6 *Conclusion*

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to the main condenser evacuation system, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the acceptance criteria of Section 10.4.2 of NUREG-0800 and the requirements of GDC 60. The staff based its conclusions on the following:

- WLS CDI, relating to WLS COL FSAR Section 10.4.2.2.1, "General Description," concerning cooling water source for the vacuum pump seal water heat exchanger, is acceptable to the staff because it meets GDC 60 for the control of releases of radioactive materials to the environment.

- WLS CDI, relating to WLS COL FSAR Section 10.4.2.2.2, "Component Description," concerning the tube side water flow in the seal water heat exchangers, is acceptable to the staff because it meets GDC 60 for the control of releases of radioactive materials to the environment.

10.4.3 Gland Sealing System (Related to RG 1.206, Section C.III.1, Chapter 10, C.I.10.4.3, "Turbine Gland Sealing System")

The gland seal system prevents the escape of steam from the turbine shaft, turbine casing penetrations, and valve stems. The gland seal system also prevents air in-leakage through sub-atmospheric turbine glands. The system provides a source of sealing steam to the annulus space where the turbine and large steam valve shafts penetrate the turbine casings.

Section 10.4 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 10.4.3 of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

10.4.4 Turbine Bypass System

The turbine bypass system provides the capability to discharge main steam from the steam generators directly to the main condenser, which minimizes load transient effects on the nuclear steam supply system. The turbine bypass system is designed to discharge a certain percentage of rated main steam flow directly to the main condenser, bypassing the turbine. The system is also used to discharge main steam during reactor hot standby and cooldown operations.

Section 10.4 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 10.4.4 of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

10.4.5 Circulating Water System

10.4.5.1 *Introduction*

The CWS removes waste heat from the main condenser. This waste heat is subsequently transferred to the power cycle heat sink. The CWS provides a continuous supply of cooling water to the main condenser to remove the heat rejected by the turbine cycle and auxiliary systems.

10.4.5.2 *Summary of Application*

Section 10.4 of the WLS COL FSAR, Revision 11, incorporates by reference Section 10.4 of the AP1000 DCD, Revision 19. Section 10.4 of the DCD includes Section 10.4.5.

In addition, in WLS COL FSAR Section 10.4.5, the applicant provided the following:

AP1000 COL Information Item

- WLS COL 10.4-1

The applicant provided additional information related to the CWS design parameters in WLS COL 10.4-1 to resolve the COL information item in Section 10.4.12.1 of the AP1000 DCD (COL Action Item 10.5-3).

Site-Specific Information Replacing Conceptual Design Information

- WLS CDI

The applicant provided additional information to replace CDI in WLS COL FSAR Section 10.4.5, which describes the following various aspects of the site-specific CWS:

- Power generation design basis
- General description
- Component description
- System operation
- Tests and inspections
- Instrumentation applications

10.4.5.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the regulatory basis for acceptance of COL Information Item 10.4-1 (COL Action Item 10.5-3) is established in GDC 4, as it relates to design provisions provided to accommodate the effects of discharging water that may result from a failure of a component or piping in the CWS.

In accordance with Section 10.4.5 of NUREG-0800, the requirements of GDC 4 are met when the CWS design includes provisions to accommodate the effects of discharging water that may result from a failure of a component or piping in the CWS. Means should be provided to prevent or detect and control flooding of safety-related areas so that the intended safety function of a system or component will not be precluded due to leakage from the CWS. Malfunction or a failure of a component or piping of the CWS, including an expansion joint, should not have unacceptable adverse effects on the functional performance capabilities of safety-related systems or components.

10.4.5.4 *Technical Evaluation*

The NRC staff reviewed Section 10.4.5 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the CWS. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the WLS COL FSAR and the applicant's responses to the staff RAIs, and provides its evaluation as described below:

<u>*AP1000 COL Information Items*</u>

- WLS COL 10.4-1

In WLS COL FSAR Section 10.4.5, the applicant provided additional information in WLS COL 10.4-1 to resolve the COL information item in Section 10.4.12.1, "Circulating Water System," of the AP1000 DCD, which states:

> The Combined License applicant will address the final configuration of the plant circulating water system including piping design pressure, the cooling tower or other site-specific heat sink.
>
> As applicable, the Combined License applicant will address the acceptable Langelier or Stability Index range, the specific chemical selected for use in the CWS water chemistry control, pH adjuster, corrosion inhibiter, scale inhibiter, dispersant, algaecide and biocide applications reflecting potential variations in site water chemistry and in micro macro biological life forms. A biocide such as sodium hypochlorite is recommended. Toxic gases such as chlorine are not recommended. The impact of toxic gases on the main control room habitability is addressed in Section 6.4. The Combined License applicant will also be responsible for the design, routing, and disposition requirements associated with the main condenser waterbox drains.

This item was also captured as COL Action Item 10.5-3 in Appendix F of NUREG-1793:

> The COL applicant is responsible for the site-specific configuration of the plant circulating water system (including piping design pressure), the cooling tower, or other site-specific heat sink.

The applicant addressed the above COL information item of the AP1000 DCD in WLS COL FSAR Sections 10.4.5.2.1, "General Description"; 10.4.5.2.2, "Component Description"; and 10.4.5.5, "Instrumentation Applications"; by providing additional information concerning CWS heat sink capability, design parameters, cooling towers, waterbox drains, and CWS water chemistry control. The staff reviewed the applicant's information in these FSAR sections.

In WLS COL FSAR Section 10.4.5.2.1, the applicant described the WLS site-specific CWS. The CWS and the cooling towers provide a heat sink for waste heat exhausted from the main steam turbine. Also, to address COL Information Item 10.4-1 of the AP1000 DCD, the applicant provided WLS-specific design parameters in WLS COL FSAR Table 10.4-202, "Design Parameters for Major Circulating Water System Components." These design parameters in the FSAR Table 10.4-202 are compatible with those in the DCD Table 10.4.5-1, "Design Parameters for Major Circulating Water System Components." WLS FSAR Section 10.4.5.2.2, describes that the maximum pressure of the CWS, including piping, valves, condenser water boxes, and tube bundles, is 90 psig. According to the DCD Table 10.4.1-1, "Main Condenser Design," the water box pressure is also 90 psig. Since the WLS CWS design parameters, including the waterbox design pressure, are compatible with those of the DCD, the staff finds the design parameters and design pressure of the WLS CWS are acceptable.

With respect to maintaining the CWS water chemistry, in FSAR Section 10.4.5.2.2, "Component Description," the applicant provided information on the chemical treatment program for the CWS. The applicant stated that the design of the WLS chemical treatment program is based on experience gained from the operation of the Catawba Nuclear Station (CNS), which is also operated by the applicant. The applicant further stated that based on a similarity of the water chemistry produced by the two water sheds and the similarity in the construction of the cooling towers, CNS was used as a model for the design of the chemical treatment program for the CWS at WLS. Accordingly, as in the CNS, WLS would utilize oxidizing chemistry (e.g., sodium hypochloride, sodium bromide, etc.) for the control of bio-fouling and the growth of algae, sulphuric acid for pH adjustment, and a polyacrylate as a silt dispersant. The applicant stated no need for corrosion and scale inhibitors based on the materials of construction of the CWS and the constituency of the dissolved and suspended solids in the Broad River from where the WLS station would draw water. Also, in Section 10.4.5.2.2, the applicant stated that specific chemicals used within the system are determined by the site water conditions and are monitored by plant chemistry personnel. Additionally, in FSAR Section 10.4.5.5, the applicant stated that circulating water chemistry is controlled by cooling tower blowdown via regulating the blowdown valve, and chemical addition to an acceptable Stability Index range of approximately 6 to 7. The staff finds that the applicant satisfactorily addressed the site-specific chemicals selected for use in CWS water chemistry control as required by the DCD.

In Revision 5 of WLS FSAR Section 10.4.5.2.2, the applicant stated that the condenser water box drains allow the condenser to be drained to the turbine building sumps. According to AP1000 DCD Section 9.2.9.2.1, "General Description," these turbine building sumps are equipped with radiation monitors at the sump pump discharge piping, which trip the pump, and provide an alarm upon detection of radioactivity in the discharge water. The staff finds the routing of the WLS condenser water drains acceptable because they flow to the turbine building sump which is equipped to monitor the radioactivity in the discharge water.

The staff reviewed the information provided in the above WLS COL FSAR sections and finds that the applicant addressed the final configuration of the CWS as specified in the COL Information Item 10.4-1. The staff also finds that the design piping pressures of the WLS CWS are consistent with the design pressures of the conceptual (nonsite-specific) design of the AP1000 CWS, and are, therefore, acceptable.

The staff's evaluation of the CWS final configuration is addressed below under the CDI discussions.

Site-Specific Information Replacing Conceptual Design Information

- WLS CDI

The applicant provided WLS site-specific design information as part of the FSAR to replace the CDI in the AP1000 DCD regarding the CWS. The applicant replaced bracketed text throughout Section 10.4.5 of the AP1000 DCD to provide site-specific CWS power generation design basis information, general CWS description, component description, system operation, tests and inspections, and instrumentation applications. The staff reviewed the WLS CDIs provided throughout WLS COL FSAR Section 10.4.5, including the Revision 5 updates related to the CWS system, and the following provides the staff's evaluation of these CDIs in the application.

In WLS COL FSAR Sections 10.4.5.1, "Design Bases," and 10.4.5.2, "System Description," the applicant provided a description of its CWS system configuration. The CWS is a nonsafety-related system. The CWS supplies cooling water to remove heat from the main condensers, the turbine building closed cooling water system heat exchangers and the condenser vacuum pump seal water heat exchangers under varying conditions of power plant loading and design weather conditions.

In WLS COL FSAR Section 10.4.5.2.1, "General Description," the applicant provided site-specific design information in that the WLS CWS consists of four 33-1/3 percent capacity circulating water pumps, two mechanical draft cooling towers, and associated piping, valves, and instrumentation. Three pumps are normally operating with one pump on standby. In Section 10.4.5.2.2, "Component Description," the applicant states that each pump has a discharge motor operated butterfly valve and stop logs for suction isolation. This permits isolation of each pump for maintenance.

In WLS COL FSAR Section 10.4.5.2.2, "Component Description," the applicant provided WLS-specific design information regarding the CWS major components, such as circulating water pumps, cooling tower, cooling tower makeup and blowdown, and piping and valves, to address the configuration of the CWS. The applicant states that the two mechanical draft cooling towers are round counter-flow type cooling towers with an impingement-type drift eliminator system, and a bypass system. The applicant further states that each cooling tower has a diameter of approximately 360 feet and a height of 85 feet). Also, the cooling towers are designed to cool the circulating water to 88 °F with a hot water inlet temperature of 113 °F. These conceptual design temperatures are consistent with the DCD design parameters for major CWS components found in DCD Table 10.4.5-1, and therefore acceptable to the staff.

Regarding external flooding considerations, the staff could not find any further details regarding the location and proximity of the mechanical draft cooling towers with respect to the plant and safety-related equipment. Therefore, the staff raised a concern regarding the effects of the cooling tower failure on the nearby safety-related equipment and structures of the plant. To complete its review, the staff requested the applicant in WLS RAI 10.04.05-2, to provide clarification and/or additional information to ensure that failure of these towers will not affect the structures, systems and components (SSCs) that perform or support a safety function.

In response to WLS RAI 10.04.05-2, dated September 10, 2008, the applicant revised the FSAR Section 10.4.5.2.2, third paragraph under "Cooling Towers," to read as follows:

The cooling tower basins serve as storage for the circulating water inventory and allow bypassing of the cooling tower during cold weather operations. The cooling tower nearest to the Unit 1 safety-related structures, systems and components (SSCs) is located over 700 ft. west of the Unit 1 auxiliary building. The cooling tower nearest to the Unit 2 safety-related SSCs is located over 600 ft. east of the Unit 2 containment building.

In Revision 5 of WLS COL FSAR, the applicant further revised the this section to include a statement that the cooling tower basins are below grade such that a basin failure will not result in migration of water across the site. It further stated that the site is graded to direct surface water flow away from the nuclear islands and that a break in the cooling tower basin or the associated CWS piping will not have an adverse effect on safety-related SSCs resulting from external plant flooding. The grading of the site combined with the location and below-grade elevation of the cooling tower basins and the associated CWS piping will preclude adverse interactions with safety-related SSCs. The staff finds the applicant's response to RAI 10.04.05-2 acceptable, since the design provisions of the WLS CWS with respect to external flooding meet the requirements of GDC 4 criteria, as described in SRP Section 10.4.5. Therefore, the staff's concern regarding the external flooding due to failure of the cooling towers and its associated piping is resolved, and RAI 10.04.05-2 is closed.

Regarding internal flooding, in WLS COL FSAR Section 10.4.5.2.3, "System Operation," the applicant refers to text from the AP1000 DCD, stating: "The effects of flooding due to a circulating water system failure, such as the rupture of an expansion joint, will not result in detrimental effects on safety-related equipment since there is no safety-related equipment in the turbine building and the base slab of the turbine building is located at grade elevation. Water from a system rupture will run out of the building through a relief panel in the turbine building west wall before the level could rise high enough to cause damage. Site grading will carry the water away from safety-related buildings." The staff finds that a malfunction or a failure of a component or piping of the CWS, including an expansion joint, will not have unacceptable adverse effects on the functional performance capabilities of safety-related systems or components for the reasons noted above. Therefore, the GDC 4 requirements have been satisfied since the flooding that results from failure of the CWS does not adversely impact any safety-related SSCs.

Further, the staff finds that the CWS cooling tower makeup is provided by the RWS, described in WLS COL FSAR Section 9.2.11, "Raw Water System." Makeup to and blowdown from the CWS is controlled by the makeup and blowdown control valves. The evaluation of RWS capabilities is provided in Section 9.2.11 of this SER.

The underground portions of the CWS piping are constructed of prestressed concrete pressure piping. The remainder of the piping is carbon steel and is coated internally with a corrosion-resistant compound. As indicated earlier, the condenser water box drains allow the condenser to be drained to the turbine building sump. Motor-operated butterfly valves are provided in each of the circulating water lines at their inlet to allow the condenser to be drained to the cooling tower basin. Control valves provide regulation of cooling tower makeup. The circulating water system is designed to withstand the maximum operating discharge pressure of the circulating water pumps. The piping design pressure is 621 kPa (90 psig), which is in accordance with the DCD value, and therefore acceptable.

In WLS COL FSAR Section 10.4.5.2.3, "System Operation," the applicant stated that if the circulating water pumps, the cooling tower, or the circulating water piping malfunction and the condenser is not available to adequately support unit operation, cooldown of the reactor may be accomplished by using the power-operated atmospheric steam relief valves or safety valves rather than the turbine bypass system. The staff finds that this alternate cooldown method is acceptable, because the turbine bypass system will not function during accident conditions and the CWS is not required for safe shutdown following an accident. Further, the applicant stated that circulating water flow to the cooling towers can be diverted directly to the basins, bypassing the cooling towers' internals, by opening the bypass valves during plant startup or partial load or to maintain CWS temperatures above 40 °F (4.4 °C). The staff finds that these provisions of the site-specific CWS design meet the requirements of GDC 4, as described in NUREG-0800, Section 10.4.5.

In WLS COL FSAR Section 10.4.5.5, "Instrumentation Application," the applicant identifies the configuration and function of the CWS pressure, temperature and level instrumentation at the WLS site. Also, the motor operated valve at each pump discharge is interlocked with the pump, so that the pump trips if the discharge valve fails to reach the full-open position shortly after starting the pump.

Based on its review of the information provided by the applicant, the staff concludes that the site-specific design of the WLS CWS (WLS CDI) provided in the WLS COL FSAR sections above adequately addresses the information that was specified in the AP1000 DCD.

10.4.5.5 *Post Combined License Activities*

There are no post-COL activities related to this section.

10.4.5.6 *Conclusion*

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to the CWS, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, WLS CDI involving the CWS is adequately addressed by the applicant. The staff concludes that the relevant information presented in the WLS COL FSAR and the applicant's RAI responses are acceptable and meet the acceptance criteria of Section 10.4.5 of NUREG-0800 and the requirements of GDC 4. The staff based its conclusions on the following:

- WLS COL 10.4-1, relating to the final configuration of the circulating water, is acceptable to the staff because the applicant addressed the site-specific design, the chemicals and control and maintenance of the CWS chemistry, in order to be consistent with AP1000 DCD.

- WLS CDI, relating to various aspects of the CWS, is acceptable to the staff because failure of the site-specific CWS design does not adversely impact any safety-related SSCs.

10.4.6 Condensate Polishing System (Related to RG 1.206, Section C.III.1, Chapter 10, C.I.10.4.6, "Condensate Cleanup System")

The condensate polishing system can be used to remove corrosion products and ionic impurities from the condensate system during plant startup, hot standby, power operation with abnormal secondary cycle chemistry, safe shutdown, and cold shutdown operations.

Section 10.4 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 10.4.6 of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

10.4.7 Condensate and Feedwater System

10.4.7.1 *Introduction*

The condensate and feedwater system provides feedwater at the required temperature, pressure, and flow rate to the SGs. Condensate is pumped from the main condenser hot well by the condensate pumps, passes through the low-pressure feedwater heaters to the feedwater pumps, and then is pumped through the high-pressure feedwater heaters to the SGs.

10.4.7.2 *Summary of Application*

Section 10.4 of the WLS COL FSAR, Revision 11, incorporates by reference Section 10.4 of the AP1000 DCD, Revision 19. Section 10.4 of the DCD includes Section 10.4.7.

In addition, in WLS COL FSAR Section 10.4.7.2.1, the applicant provided the following:

AP1000 COL Information Item

- WLS COL 10.4-2

The applicant provided additional information in WLS COL 10.4-2 to address the COL information item in Section 10.4.12.2, "Condensate, Feedwater and Auxiliary Steam System Chemistry Control," of the AP1000 DCD (COL Action Item 10.5-4).

Supplemental Information

- STD SUP 10.4-1

The applicant provided supplemental information in WLS COL FSAR Section 10.4.7.2.1, “General Description,” which addresses operations and maintenance procedures.

- STD SUP 10.4-2

The applicant provided supplemental information, which states that the EPRI Secondary Water Chemistry Guidelines will be used for guidance on selection of pH control agents and pH optimization as described in Nuclear Energy Institute (NEI) 97-06, “Steam Generator Program Guidelines.”

10.4.7.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the regulatory basis for acceptance of the COL information item and STD SUP 10.4-2 is GDC 14-Reactor coolant pressure boundary, as it relates to ensuring the integrity of the reactor coolant pressure boundary (specifically as the secondary water chemistry program ensures the integrity of the SG tubing). The applicable acceptance criteria for meeting GDC 14 are found in NUREG-0800 Sections 10.4.6 and 5.4.2.1, including BTP 5-1. The regulatory basis for acceptance of STD SUP 10.4-1 is established in GDC 4, insofar as it requires that the dynamic effects associated with possible fluid flow instabilities (e.g., water hammers) during normal plant operation, as well as during upset or accident conditions be considered, and that SSCs important to safety be designed to accommodate the effects of, and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

GDC 4 can be complied with by meeting the relevant acceptance criteria specified in Section 10.4.7 of NUREG-0800, “Condensate and Feedwater System.” In regard to fluid instabilities, the requirements of GDC 4, as related to protecting SSCs against the dynamic effects associated with possible fluid flow instabilities (e.g., water hammers) during normal plant operation, as well as during upset or accident conditions can be met by: (1) meeting the guidance in BTP 10-2, “Design Guidelines for Avoiding Water Hammers in Steam Generators,” for reducing the potential for water hammers in SGs; and (2) meeting the guidance related to feedwater-control-induced water hammer. Guidance for water hammer prevention and mitigation is given in NUREG-0927, Revision 1, “Evaluation of Water Hammer Occurrences in Nuclear Power Plants.”

10.4.7.4 *Technical Evaluation*

The NRC staff reviewed Section 10.4.7 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff’s review confirmed that the information in the application and incorporated by reference addresses the required information relating to the condensate and feedwater system. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the BLN Units 3 and 4 COL application.

The staff reviewed the information in the WLS COL FSAR:

AP1000 COL Information Item

- WLS COL 10.4-2

In WLS COL FSAR Section 10.4.7.2.1, the applicant provided additional information in WLS COL 10.4-2 to address the COL information item in Section 10.4.12.2, "Condensate, Feedwater and Auxiliary Steam System Chemistry Control," of the AP1000 DCD, which states:

> The Combined License applicant will address the oxygen scavenging agent and pH adjuster selection for the turbine island chemical feed system.

The commitment was also captured as COL Action Item 10.5-4 in Appendix F of NUREG-1793:

> The COL applicant is responsible for chemistry control of the condensate, feedwater, and auxiliary steam system.

The WLS COL FSAR modified Section 10.4.7.2.1 of the AP1000 DCD, to state:

> The oxygen scavenger agents are hydrazine and carbohydrazide. The pH control agents are dimethylamine and methoxypropylamine.

The NRC staff reviewed the resolution to WLS COL 10.4-2 regarding the text added to Section 10.4.7.2.1, related to condensate, feedwater, and auxiliary steam system chemistry control.

The description of the secondary water chemistry control program is addressed in the AP1000 DCD, Section 10.3.5. Consistency with industry guidelines was addressed in the AP1000 DCD, Section 10.3.5.5, which stated that action taken when chemistry parameters are outside normal operating ranges will, in general, be consistent with action levels described in Reference 1 (“PWR Secondary Water Chemistry Guidelines,” EPRI technical report (TR) TR-102134-R5, March 2000). However, the AP1000 DCD does not specify the oxygen scavenger or pH control chemicals to be used. This is to be addressed by COL Information Item 10.4-2 of the AP1000 DCD.

Revision 6 of the EPRI Secondary Water Chemistry Guidelines (EPRI Guidelines), which is the latest published version of these guidelines, does not require a specific oxygen scavenging agent. However, the guidelines do note that hydrazine and carbohydrazide are the most commonly used oxygen scavenger for PWR secondary systems and are generally recognized as effective for this purpose. Therefore, the staff finds the identified oxygen scavenger agents are consistent with the EPRI guidelines.

For pH control, the EPRI secondary water chemistry guidelines do not require specific amines. Section 3.3.1 of the EPRI Guidelines recommends a plant-specific amine be selected based on a number of factors. Section 3.3.1 of the EPRI Guidelines lists several amines that have been used or are being used in PWR plants as pH control agents, including dimethylamine and methoxypropylamine. Section 3.3.1.2 of the EPRI Guidelines states that if implementing advanced amine treatment, a site-specific materials compatibility review will be necessary to ensure that components, particularly elastomers, are compatible with the amine. The EPRI Guidelines, in Table 5-4, “Recirculating Steam Generator Power Operation (≥30% Reactor Power) Feedwater Sample,” refer to several other EPRI reports for guidance for optimization of the pH in conjunction with the amine selected. The applicant did not explicitly describe how the selected amine was qualified, or how the pH will be optimized in conjunction with the selected amines.

Although the applicant did not explicitly describe how the selected amines were qualified, STD SUP 10.4-2 ensures that the qualification of the chosen oxygen scavenging and pH control chemicals will be consistent with the EPRI PWR Secondary Water Chemistry Guidelines. (See evaluation of STD SUP 10.4-2 below under evaluation of supplemental information).

The staff finds the pH control and oxygen scavenger chemical acceptable because the proposed chemicals will be qualified and the resulting pH optimized following the guidance of the EPRI PWR Secondary Water Chemistry Guidelines, which is referenced in NUREG-0800 as acceptable guidance to ensure that the secondary water chemistry program meets GDC 14. On the basis of the information provided by the applicant and the acceptance criteria in BTP 5-1, the staff concludes that the proposed secondary chemistry that uses hydrazine and carbohydrazide, and dimethylamine and methoxypropylamine is acceptable.

The following portion of this technical evaluation section is reproduced from Section 10.4.7.4 of the VEGP SER:

Supplemental Information

- *STD SUP 10.4-1*

The applicant provided supplemental information as part of the BLN COL FSAR regarding operations and maintenance procedures. The applicant added the following text to the end of Section 10.4.7.2.1 of the AP1000 DCD, Revision 17*:*

Operations and maintenance procedures include appropriate precautions to avoid steam/water hammer occurrences.

The NRC staff reviewed the standard supplemental information provided in STD SUP 10.4-1 regarding the text added to Section 10.4.7.2.1 related to operations and maintenance procedures.

In Section 10.4.7 of NUREG-0800, Acceptance Criteria 2, provides acceptable methods of compliance with the requirements in GDC 4, as it applies to fluid flow instabilities, (e.g., water hammer). Criteria 2B, "Meeting the guidance related to feedwater-control-induced water hammer," states that guidance for water hammer and mitigation is found in NUREG-0927. The supplemental information added to the BLN COL FSAR states that operations and maintenance procedures include appropriate precautions to avoid steam/water hammer occurrences; however, the supplemental information being proposed by the applicant did not identify what type of precautions included in the procedures minimize the potential for water hammer occurrences. In order to ensure that the procedures adequately address water hammer prevention and mitigation, the staff requested in RAI 10.4-7-1, in a letter dated June 3, 2008, that the applicant provide a more detailed statement concerning the use of operations and maintenance procedures, including information on what specific elements in the procedures (i.e., venting) will result in reduced potential of water hammer occurrences.

In its response, dated July 17, 2008, concerning reducing the potential for water hammer events, the applicant identified that they programmatically integrate into the AP1000 Operations Procedure development good operating practice and operating experience including, but not limited to, Institute of Nuclear Power Operations (INPO) significant event reports and significant operating event reports, NRC information notices and bulletins, and other industry operating experience information. Further, the applicant explained that specific operating experience to preclude or mitigate water hammer is included in this population of operating experience. In addition, the applicant explained that the AP1000 has been designed to prevent or minimize steam and water hammer. The applicant *agreed to revise the procedure elements in BLN COL FSAR Section 10.4.7.2.1, and described in STD SUP 10.4-1, to include additional precautions to minimize the potential for steam and water hammer.*

The revised STD SUP 10.4-1, in BLN COL FSAR Section 10.4.7.2.1 now reads as follows:

> *Operations and maintenance procedures include precautions, when appropriate, to minimize the potential for steam and water hammer, including:*
>
> - *Prevention of rapid valve motion.*
> - *Process for avoiding introduction of voids into water-filled lines and components.*
> - *Proper filling and venting of water-filled lines and components.*
> - *Process for avoiding introduction of steam or heated water that can flash into water-filled lines and components.*
> - *Cautions for introduction of water into steam-filled lines or components.*
> - *Proper warmup of steam-filled lines.*
> - *Proper drainage of steam-filled lines.*
> - *The effects of valve alignments on line conditions.*

Based on its review, the staff finds the applicant's response acceptable because a detailed list of the procedural precautions that would reduce or minimize the occurrence of water hammer was provided and included as a proposed revision to the COL application, Part 2, BLN COL FSAR Section 10.4.7.2.1. Further, the staff reviewed the precautions and compared them to the industry experience and staff guidance in accordance with Section 10.4.7 of NUREG-0800 and BTP 10-2. The staff finds that the applicant has adequately addressed the steam and water hammer. Therefore, the staff's concern described in RAI 10.4.7-1 is resolved.

- *STD SUP 10.4-2*

The applicant provided supplemental information explaining that the EPRI PWR Secondary Water Chemistry Guidelines will be used for guidance on selection of pH control agents and pH optimization as described in NEI 97-06.

EPRI documents provide detailed guidelines for both qualification of the selected pH control chemicals and the optimization of the secondary pH. While the staff does not review or accept the EPRI PWR Secondary Water Chemistry Guidelines through a safety evaluation, these guidelines are recognized as representing the industry consensus on best practices in water chemistry control and have been proven to be effective via many years of successful operating experience. As such, the staff finds the application of the guidance of the EPRI PWR Secondary Water Chemistry Guidelines, and a programmatic commitment to use these guidelines, to be an acceptable method for the applicant to ensure compliance with GDC 14. As discussed in a Federal Register (FR) notice, dated March 2, 2005, 70 FR 10298, the reference to NEI 97-06 and the associated water chemistry guidelines provide reasonable assurance that steam generator tube integrity will be maintained.

10.4.7.5 *Post Combined License Activities*

There are no post-COL activities related to this section.

10.4.7.6 *Conclusion*

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to the condensate and feedwater system, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the requirements of GDC 4 and GDC 14 and the guidance in Sections 10.4.6, 10.4.7, and 5.4.2.1 of NUREG-0800, NUREG-0927, BTP 5-1, and BTP 10-2. The staff based its conclusions on the following:

- WLS COL 10.4-2 and STD SUP 10.4-2, relating to the condensate, feedwater, and auxiliary system chemistry control program, are in accordance with EPRI PWR Secondary Water Chemistry Guidelines, which is referenced in NUREG-0800 Sections 10.4.6 and 5.4.2.1, including BTP 5-1 of NUREG-0800. Meeting these guidelines ensures that GDC 14 is met with respect to integrity of the reactor coolant pressure boundary, specifically as the secondary water chemistry program ensures the integrity of the SG tubing.

- STD SUP 10.4-1, relating to operations and maintenance, is acceptable to the staff because the applicant has provided a detailed list of the procedural precautions that are consistent with Section 10.4.7 of NUREG-0800 and the BTP 10-2 acceptance criteria.

10.4.8 Steam Generator Blowdown System (Related to RG 1.206, Section C.III.1, Chapter 10, C.I.10.4.8, “Steam Generator Blowdown System (PWR)”)

The SG blowdown system assists in maintaining acceptable secondary coolant water chemistry during normal operation and during anticipated operational occurrences, such as main condenser inleakage or primary to secondary SG tube leakage. It does this by processing water from each SG and removing impurities.

Section 10.4 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 10.4.8 of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff’s review confirmed that there is no outstanding issue related to this section. The results of the NRC staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

10.4.9 Startup Feedwater System

The startup feedwater system provides a supply of feedwater to the SGs during plant startup, hot standby and shutdown conditions, and during transients in the event of main feedwater system unavailability. The startup feedwater system is composed of components from the AP1000 main and startup feedwater system and SG system.

Section 10.4 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 10.4.9 of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff’s review confirmed that there is no outstanding issue related to this section. The results of the NRC staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

10.4.10 Auxiliary Steam System

The auxiliary steam system provides the steam required for plant use during startup, shutdown, and normal operation. Steam is supplied from either the auxiliary boiler or the main steam system.

Section 10.4 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 10.4.10 of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff’s review confirmed that there is no outstanding issue related to this section. The results of the NRC staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

10.4.11 Turbine Island Chemical Feed

The turbine island chemical feed system injects required chemicals into the condensate, feedwater, auxiliary steam, service water, and demineralized water treatment. Chemical feed system components are located in the turbine building.

Section 10.4 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 10.4.11 of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

10.4.12 Combined License Information

Section 10.4.12 of the WLS COL FSAR, Revision 11, incorporates by reference Section 10.4.12, "Combined License Information," of Revision 19 of the AP1000 DCD. The NRC staff reviewed Section 10.4.12 of the WLS COL FSAR and checked the referenced DCD to ensure the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1]

The applicant addressed COL Information Items 10.4-1, 10.4-2, and 10.4-3. These items are discussed and evaluated in Sections 10.4.5, 10.4.7, and 9.2.5 of this SER, respectively.

11 RADIOACTIVE WASTE MANAGEMENT

The radioactive waste management systems are designed to control, collect, handle, process, store, and dispose of liquid, gaseous, and solid wastes that may contain radioactive materials. The systems include the instrumentation used to monitor and control the release of radioactive effluents and wastes and are designed for normal operation (including refueling; purging; fuel handling and storage; radioactive material handling, processing, use, storage, and disposal; maintenance; routine operational surveillance; in-service inspection (ISI); and calibration), and anticipated operational occurrences (AOOs).

11.1 Source Terms

The radioactive source terms are used to identify the potential dose to members of the public and plant employees as a result of plant operation. This includes consideration of parameters used to determine the concentration of each radionuclide in the reactor coolant, fraction of fission product activity released to the reactor coolant, and concentrations of all non-fission product radionuclide in the reactor coolant. Gaseous and liquid waste sources are considered in the evaluation of effluent releases.

Section 11.1 of the William States Lee III Nuclear Station (WLS) combined license (COL) Final Safety Analysis Report (FSAR), Revision 11, incorporates by reference, Section 11.1, “Source Terms,” of the AP1000 Design Control Document (DCD), Revision 19. In addition, in the WLS COL FSAR, the applicant provided the following:

Departures

- WLS DEP 6.4-1

The applicant provided additional information in Section 11.1 of the WLS COL FSAR about WLS DEP 6.4-1 related to design changes affecting habitability of the main control room and changes to the calculated doses to control room operators. This information, as well as related WLS DEP 6.4-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.2 of this safety evaluation report (SER).

The U.S. Nuclear Regulatory Commission (NRC) staff (the staff) reviewed Section 11.1 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this section.[1] The NRC staff’s review confirmed that there is no outstanding issue related to this section. The results of the NRC staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design,” and its supplements.

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[1] See Section 1.2.2 of this SER for a discussion of the staff’s review related to verification of the scope of information to be included in a COL application that references a design certification (DC).

11.2 Liquid Waste Management Systems

11.2.1 Introduction

The liquid waste management system (LWMS) is designed to control, collect, process, handle, store, and dispose of liquid radioactive waste generated as the result of normal operation, including anticipated operational occurrences.

11.2.2 Summary of Application

Section 11.2 of the WLS COL FSAR, Revision 11, incorporates by reference Section 11.2 of the AP1000 DCD, Revision 19.

In addition, in WLS COL FSAR Section 11.2, the applicant provided the following:

AP1000 COL Information Items

- STD COL 11.2-1

The applicant provided additional information in Standard (STD) COL 11.2-1 to resolve COL Information Item 11.2-1 (COL Action Item 11.2-1). The additional information addresses the use of mobile or temporary equipment to process liquid effluents in WLS COL FSAR Section 11.2.1.2.5.2.

- STD COL 11.2-2

The applicant provided additional information in STD COL 11.2-2 regarding liquid radwaste cost-benefit analysis methodology.

- WLS COL 11.2-1 and WLS COL 13.5-1

The applicant provided additional information in WLS COL 11.2-1 and WLS COL 13.5-1 to ensure that the total inventory of radioactivity contained in waste processing equipment, skid-mounted systems, and in-process waste located in the Radwaste Building is limited in accordance with RG 1.143, Revision 2. This information is provided to resolve STD COL 11.2-1.

- WLS COL 11.2-2

The applicant provided additional information in WLS COL 11.2-2 to resolve COL Information Item 11.2-2 (COL Action Item 11.2-2). The additional information addresses the dilution factors used for dose calculations and the cost-benefit analysis of population doses in WLS COL FSAR Sections 11.2.3.3 and 11.2.3.5.

- WLS COL 2.4-5 and WLS COL 15.7-1

WLS COL FSAR Section 11.2 does not identify WLS COL 2.4-5 and WLS COL 15.7-1 as COL information items applicable to Section 11.2. However, WLS COL 2.4-5 and WLS COL 15.7-1 provide information regarding a postulated liquid waste tank failure, which is evaluated by the staff as part of liquid waste management. Therefore, WLS COL 2.4-5 and WLS COL 15.7-1 are evaluated in Section 11.2.4 of this SER. In WLS COL FSAR Section 2.4.13, the applicant performed the consequence analysis of a postulated liquid waste tank failure to address COL Information Items 2.4-5 and 15.7-1.

- WLS COL 11.5-3

The applicant provided additional information in WLS COL 11.5-3 to resolve COL Information Item 11.5-3 (COL Action Item 11.5-3). The additional information addresses compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," Section II.A in WLS COL FSAR Section 11.2.3.5.

Supplemental Information

- STD SUP 11.2-1

The applicant added in WLS COL FSAR Section 11.2.3.6 supplemental (SUP) information to address the quality assurance (QA) program to be applied to the LWMS.

- STD SUP 11.2-2

The applicant added supplemental information in WLS COL FSAR Section 11.2.3 to address the liquid effluent site interface parameter.

- WLS SUP 11.2-3

The applicant added supplemental information in WLS COL FSAR Section 11.2.1.2.4 regarding the exterior radwaste discharge piping.

License Condition

- Part 10, License Condition, "Radwaste Building Radioactivity Limits"

WLS COL application, Part 10, Section 13, "Radwaste Building Radioactivity Limits," states that prior to initial fuel load, the licensee shall develop, implement, and maintain procedural controls limiting radionuclide inventory in each of the Radwaste Building Monitor Tanks, and separately in each of up to three Radwaste Building mobile radwaste processing systems to below A2 quantities for radionuclides specified in Appendix A to 10 CFR Part 71 (Tables A-1 and A-3), as described in FSAR Section 13.5.2.2.5. The procedures shall also ensure that any additional equipment located in the RWB is limited to the A_2 quantities and that the total cumulative radioactive inventory contained in unpackaged wastes (including liquid waste, wet waste, solid waste, gaseous waste, activated or contaminated metals and components, and contaminated waste present at any time in the Radwaste Building) is limited so that an unmitigated release, occurring over a 2 hour time period, would not result in a dose of greater than 500 millirem at the protected area boundary or an unmitigated exposure, occurring over a 2 hour time period, would not result in a dose of greater than 5 rem to site personnel located 10 feet from the total cumulative radioactive inventory.

11.2.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the regulatory basis for acceptance of the supplementary information on the LWMS is established in:

- 10 CFR 20.1301(e), as it relates to compliance with 40 CFR 190, "Environmental Radiation Protection Standards for Nuclear Power Operations"
- 10 CFR 20.1302, "Compliance with Dose Limits for Individual Members of the Public"
- 10 CFR 20.1406, "Minimization of Contamination"
- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 60, "Control of Releases of Radioactive Materials to the Environment"
- 10 CFR Part 50, Appendix A, GDC 61, "Fuel Storage and Handling and Radioactivity Control"
- 10 CFR Part 50, Appendix I, Sections II.A and II.D as it relates to liquid effluent dose objectives and associated cost-benefit analysis
- 10 CFR 50.34a, "Design Objectives for Equipment to Control Releases of Radioactive Material in Effluents—Nuclear Power Reactors"
- 10 CFR 52.80(a), as it relates to those inspections, tests, analysis that the licensee shall perform, and the necessary acceptance criteria that are necessary to show the facility shall be constructed and operated in conformity with the COL
- Title 40 of the *Code of Federal Regulations* (40 CFR) Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations"

Guidance for accepting the supplementary information on the LWMS is in:

- The codes and standards listed in Table 1 of Regulatory Guide (RG) 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," Revision 2
- Regulatory Position C.1.1 of RG 1.143, Revision 2
- RG 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1
- RG 1.110, "Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors"

- RG 1.113, “Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I,” Revision 1

- RG 4.21, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning”

The acceptance criteria associated with the LWMS are given in Section 11.2 of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition),” and NUREG-0800, Section 2.4.13, Acceptance Criterion No. 5, including Branch Technical Position (BTP) 11-6, referencing the March 2007 Standard Review Plan (SRP).

11.2.4 Technical Evaluation

The staff reviewed Section 11.2 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The staff’s review confirmed that the information in the application and incorporated by reference addresses the required information relating to the LWMS. The results of the staff’s evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff’s review of this application included the following COL information and supplementary items:

- STD COL 11.2-1, Processing of Liquid Waste by Mobile Equipment

- STD COL 11.2-2, Liquid Radwaste Cost-Benefit Analysis Methodology

- WLS COL 11.2-1, Radwaste Building Source Term Inventories

- WLS COL 13.5-1, Radioactive Waste Management Procedures

- WLS COL 11.2-2, Cost-benefit Analysis of Population Doses

- WLS COL 2.4-5, Accidental Release of Liquid Effluents into Groundwater and Surface Water

- WLS COL 15.7-1, Consequences of Tank Failure

- WLS COL 11.5-3, Individual Dose Limits in 10 CFR Part 50, Appendix I

- STD SUP 11.2-1, Quality Assurance

- WLS SUP 11.2-2, Interface Requirements

- WLS SUP 11.2-3, Exterior Radwaste Discharge Piping

In addition to the above items, the staff reviewed the entire section against Section 11.2 of NUREG-0800 to determine if the information in WLS COL FSAR Section 11.2 met the regulatory requirements in the regulations stated above (SER Section 11.2.3) and the

NUREG-0800 acceptance criteria. The relevant NUREG-0800 acceptance criteria are as follows:

- The LWMS should have the capability to meet the dose design objectives and include provisions to treat liquid radioactive wastes such that the following is true:

    A. The calculated annual total quantity of all radioactive materials released from each reactor at the site to unrestricted areas will not result in an estimated annual dose or dose commitment from liquid effluents for any individual in an unrestricted area from all pathways of exposure in excess of 0.03 millisievert (mSv) (3 millirem (mrem)) to the total body or 0.1 mSv (10 mrem) to any organ. RG 1.109, RG 1.112, and RG 1.113 provide acceptable methods for performing this analysis.

    B. In addition to A above, the LWMS should include all items of reasonably demonstrated technology that, when added to the system sequentially and in order of diminishing cost-benefit return for a favorable cost-benefit ratio, can effect reductions in doses to the population reasonably expected to be within 80 kilometers (km) (50 miles (mi)) of the reactor. RG 1.110 provides an acceptable method for performing this analysis.

    C. The concentrations of radioactive materials in liquid effluents released to unrestricted areas should not exceed the concentration limits in Table 2, Column 2 of Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage" to 10 CFR Part 20, "Standards for Protection Against Radiation."

- The LWMS should be designed to meet the anticipated processing requirements of the plant. Adequate capacity should be provided to process liquid wastes during periods when major processing equipment may be down for maintenance (single failures) and during periods of excessive waste generation. Systems that have adequate capacity to process the anticipated wastes and that are capable of operating within the design objectives during normal operation, including anticipated operational occurrences, are acceptable. To meet these processing demands, interconnections between subsystems, redundant equipment, mobile equipment, and reserve storage capacity will be considered.

- System designs should describe features that will minimize, to the extent practicable, contamination of the facility and environment; facilitate eventual decommissioning; and minimize, to the extent practicable, the generation of radioactive waste, in accordance with the guidelines of RG 1.143, for liquids and liquid wastes produced during normal operation and anticipated operational occurrences, and the requirements of 10 CFR 20.1406. These system design features should be provided in the FSAR or the COL application to the extent that they are not addressed in a referenced certified design or design certification (DC) application.

- BTP 11-6, as it relates to the assessment of a potential release of radioactive liquids following the postulated failure of a tank and its components, located outside of containment, and impacts of the release of radioactive materials at the nearest potable water supply, located in an unrestricted area, for direct human consumption or indirectly through animals, crops, and food processing.

Section 1.2.3 of this SER provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (Vogtle Electric Generating Plant (VEGP) Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from requests for additional information (RAIs).
- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.
- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and finds the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) contains evaluation material from the SER for the Bellefonte Nuclear Plant (BLN), Units 3 and 4 COL application.

AP1000 COL Information Items

The following portion of this technical evaluation section is reproduced from Section 11.2.4 of the VEGP SER:

> *The following portion of this technical evaluation section is reproduced from Section 11.2.4 of the BLN SER:*
>
> - STD COL 11.2-1
>
> *The applicant provided additional information in STD COL 11.2-1 to resolve COL Information Item 11.2-1. COL Information Item 11.2-1 states:*
>
> > *The Combined License applicant will discuss how any mobile or temporary equipment used for storing or processing liquid radwaste conforms to Regulatory Guide 1.143. For example, this includes discussion of equipment containing radioactive liquid radwaste in the non-seismic Radwaste Building.*

The commitment was also captured in COL Action Item 11.2-1 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:

> *The COL applicant will provide information on how any mobile or temporary equipment used for storing or processing liquid radwaste conforms to RG 1.143.*

The applicant provided information in BLN COL FSAR Section 11.2.1.2.5.2 that addresses how any mobile or temporary equipment that will be used for storing or processing liquid radwaste conforms to RG 1.143. For example, this includes discussion of equipment containing radioactive liquid radwaste in the non-seismic Radwaste Building. The staff issued Request for Additional Information (RAI) 11.2-5 to clarify some of the language used in the COL concerning the extent of compliance with RG 1.143 for the temporary and mobile equipment. The applicant responded to this RAI by proposing a revision to the BLN COL FSAR text to clearly state that the applicable requirements in RG 1.143 pertain to mobile and temporary equipment.

The NRC staff reviewed the resolution of COL Information Item 11.2-1 related to the use of mobile or temporary equipment included under Section 11.2 of the BLN COL FSAR and found that the applicant's commitments for installing and operating mobile systems meets the acceptance criteria in Section 11.2 of

NUREG-0800 and RG 1.143. The NRC staff verified that Revision 1 of the BLN COL FSAR (STD COL 11.2-1) adequately incorporates the above. As a result, RAI 11.2-5 is closed.

- *STD COL 11.2-2*

The discussion of VEGP COL 11.2-2 addresses the site-specific cost-benefit analysis performed to address the requirements of 10 CFR Part 50, Appendix I, regarding population doses due to liquid effluents. The applicant provided additional information in STD COL 11.2-2 to resolve COL Information Item 11.2-2 with regard to the cost-benefit analysis methodology.

The NRC staff reviewed the resolution of COL Information Item 11.2-2 related to the cost-benefit analysis methodology described in VEGP FSAR Section 11.2.3.5.1 and concluded that the methodology used for the analysis was consistent with the guidance of RG 1.110 and was, therefore, acceptable.

- WLS COL 11.2-1 and WLS COL 13.5-1

For the staff's evaluations of WLS COL 11.2-1 and WLS COL 13.5-1, the staff applied the design centered review approach discussed in Section 1.2.3 of this SER. Under this approach, the staff performed a single review where multiple COL applicants submitted identical information. In this case, the reference COL is the Levy Nuclear Plant (LNP) Units 1 and 2, and the WLS COL is a subsequent COL.

The following portion of this technical evaluation section is reproduced from Section 11.2.4 of the LNP SER:

While BLN RAI 11.2-5 and COL FSAR Section 11.2.1.2.5.2 address mobile and temporary processing equipment, neither the response to BLN RAI 11.2-5 or information already contained in this FSAR section included a discussion of how the cumulative source term inventories of all relevant radioactive materials present in the Radwaste Building, including that in mobile or temporary equipment, conforms with the RG 1.143, Revision 2 dose acceptance criteria. Specifically, Regulatory Position C.5.1 of RG 1.143, Revision 2 states, "for a given structure housing radwaste processing systems or components, if the total design basis unmitigated radiological release (considering the maximum inventory) at the boundary of the unprotected area is greater than 500 millirem per year or the maximum unmitigated exposure to site personnel within the protected area is greater than 5 rem per year, the external structures are classified as RW-IIa." Since the AP1000 Radwaste Building is classified as RW-IIc (a classification less stringent than RW-IIa), the inventories of radioactive materials in this building should be managed and controlled in a way that will not result in these dose criteria being exceeded.

After reviewing the response to BLN RAI 11.2-5 and the FSAR information addressing COL information item 11.2-1, the staff issued RAI 11.02-4 requesting that the applicant provide information related to the types and quantities of radioactive material within the Radwaste Building and describing how the unmitigated dose criteria to a worker and members of the public will be met, given the guidance and acceptance criteria of RG 1.143, Revision 2.

In the response to RAI 11.02-4, dated February 11, 2013, the applicant indicated that there will be three primary types of radioactive waste within the Radwaste Building. The three types of waste are; 1) liquid waste stored within the three 15,000 gallon monitor tanks, 2) waste associated with liquid mobile waste processing systems which may be utilized within the Radwaste Building, and 3) solid wastes and wastes which have been packaged and are ready for shipment.

The applicant provided information explaining how operational programs and procedures will ensure that the RG 1.143, Revision 2 dose criteria are not exceeded from the monitor tanks and mobile equipment. In this context, waste that is packaged and ready for shipment is not within the scope of RG 1.143, Revision 2. In its response, the applicant assumed that monitor tanks and a mobile skid-mounted processing system located in the radwaste building have the same radionuclide distributions and inventories as the effluent holdup tank listed in FSAR Table 2.4.13-202, normalized to the 10 CFR Part 71, Appendix A, A_2 limit (with A_2 quantities being calculated using 10 CFR Part 71, Appendix A information). The total radioactivity in a mobile skid-mounted processing equipment was assumed to be analogous to the radioactivity that would be contained in a demineralizer used for the same functional purpose. Using conservative assumptions, the applicant calculated dose rates that were less than the unmitigated release and exposure acceptance criteria of RG 1.143, Revision 2. In addition, the applicant provided a proposed FSAR markup and license condition requiring that procedures be developed, prior to fuel load, limiting the amount of radioactive materials in each of the monitor tanks and in the mobile processing equipment to below the 10 CFR Part 71 A_2 quantities.

While this response partially resolved the staff's technical and regulatory concerns, the effluent holdup tank radioactive source term, provided in FSAR Table 2.4.13-202, used in developing the A_2 quantities for the monitor tanks and mobile equipment was based on a fuel failure rate of 0.125 percent. While this fuel failure rate assumption is acceptable for complying with SRP Section 11.2, BTP 11-6, for the purposes of RG 1.143 the design basis failed fuel fraction of 0.25 percent should have been used instead, consistent with the guidance provided in SRP Section 12.2. In addition, while RG 1.143, Revision 2 indicates that the total building inventory should be considered in accordance with Regulatory Position C.5.1, it was unclear if the applicant was considering the cumulative source term of all components typically used in a mobile processing skid and if the cumulative source term from up to three mobile skids were being considered to support waste processing operations. AP1000 DCD, FSAR Chapter 11, indicates that three mobile skids may be present at any one time in the Radwaste Building. Also, the staff was concerned that pre-processed or unpackaged waste may be present in the Radwaste Building, such as contaminated equipment or components or waste previously transferred from mobile equipment, and were potentially not being considered in the response and proposed FSAR markup and license condition. Finally, the staff determined that additional information should be provided in response to COL Information Items 11.2-1 and 11.4-1 since the responses to the COL items did not fully address how waste associated with mobile equipment or unpackaged waste would be controlled in complying with the safety classification assigned to the Radwaste Building. As a result, the staff closed RAI 11.02-4 and issued supplementary RAI 11.02-5 to resolve the above concerns and request additional information related to the response to COL Information Items 11.2-1 and 11.4-1 and conformance with RG 1.143, Revision 2, acceptance criteria.

In the initial response to RAI 11.02-5, dated April 26, 2013, the applicant revised the source term for an individual monitor tanks using the RCS source term and radionuclide concentrations described in FSAR Table 2.4.13-202 and DCD Table 11.1-2. This source term is based on the design basis defective fuel fraction of 0.25 percent. This source term was normalized to the 10 CFR Part 71, Appendix A, A_2 limit and is provided in Table 1 of the response. This source term was also used in calculating doses from each mobile waste processing skid, as each skid is also being limited to an inventory corresponding the 10 CFR Part 71, A_2 quantities. In addition, the applicant indicated that the source term assigned to each mobile skid was calculated assuming that the entire source term is contained in a demineralizer as a conservative approach in calculating doses. Using these source terms, the applicant recalculated the cumulative dose rate to a worker and member of the public from an unmitigated release. The applicant calculated a dose of 87 mrem to a member of the public at the protected area boundary using conservative assumptions. The dose to a worker was calculated to be 2,230 mrem at a distance of 10 feet from multiple radioactive sources in the building. However, the applicant did not provide the basis for the 10-foot distance in its analysis.

As a further commitment, the applicant updated FSAR Section 13.5.2.2.5 and proposed to revise operational procedures to include a provision requiring that spent filtration and adsorption media transferred from mobile radwaste *processing systems be transferred and packaged for offsite shipment prior to placing the mobile radwaste processing system back into service. This provision is necessary to ensure that the total cumulative inventory of unpackaged waste in the RWB is not exceeded. Finally, the applicant updated its response to COL items 11.2-1 and 11.4-1 (FSAR Sections 11.2.1.2.5.2 and 11.4.6) and the proposed license condition, with new information, providing additional detail as to how the quantity of radioactive materials in the Radwaste Building will be controlled in ensuring that RG 1.143, Revision 2 dose acceptance criteria are met. However, even with the new information, staff determined that the proposed revision to the FSAR and new license condition did not provided sufficient information to ensure conformance with RG 1.143, Revision 2. Specifically, the applicant did not provide sufficient technical justification for the 10 foot distance used to calculate the unmitigated dose to a worker, and the proposed FSAR language and license condition did not ensure that all forms of unpackaged radioactive material in the Radwaste Building would be controlled during the operation of the plant.*

Consequently, the staff requested that the applicant address these concerns, and the applicant provided an updated revision to the response on July 1, 2013. In this response, the proposed FSAR markups were revised to include additional provisions to ensure that the total cumulative inventory of all unpackaged radioactive materials in the Radwaste Building would be limited to the unmitigated release and exposure criteria specified in RG 1.143, Revision 2. In addition, the applicant justified the assumed 10-foot distance in calculating the unmitigated dose to workers. The applicant explained that operator work stations and low dose rate waiting areas are typically no closer than 10 feet from the major sources of radioactivity located in the Radwaste Building. While the applicant provided a revised license condition in their response, the staff suggested specific revisions to the license condition to ensure that operational procedures limit all unpackaged waste in the Radwaste Building to the RG 1.143, Revision 2 dose acceptance criteria..

On August 23, 2013, the applicant provided a revised response to RAI 11.05-2 modifying the proposed license condition wording in LNP COL application, Part 10, License Conditions and ITAAC, and in Section 13, "Radwaste Building Radioactivity Limits" of the LNP FSAR, to ensure that operational procedures limit all unpackaged waste in the Radwaste Building to the RG 1.143, Revision 2 dose acceptance criteria, as suggested by the staff. In addition, the applicant proposed revised FSAR language in the response, but the proposed FSAR language was not entirely consistent with the proposed license condition. Finally, in a September 12, 2013, response (ML13259A147), the applicant proposed to revise the FSAR wording to make it consistent with the proposed license condition. The proposed FSAR wording and license condition ensure that the cumulative inventory of all unpackaged waste will be controlled in accordance with RG 1.143, Revision 2.

In summary, the applicant provided additional information in FSAR Sections 11.2.1.2.5.2, 11.4.6, and 13.5.2.2.5 which fully address COL Information Items 11.2-1 and 11.4-1 (a parallel discussion related to the resolution of COL Information Item 11.4-1 is provided in SER Section 11.4.4, below). Specifically, *the applicant committed to the implementation of operational procedures that will ensure that the quantity of radioactive materials associated with each of the three monitoring tanks, in each of up to three mobile processing systems, and in any additional equipment located in the Radwaste Building, containing unpackaged waste, are limited to less than the 10 CFR Part 71, A_2 quantities. In addition, the applicant's procedures ensure that the total cumulative inventory of all unpackaged waste in the Radwaste Building (including the waste in the monitoring tanks, mobile processing systems, and any additional equipment, as well as any other unpackaged waste in the Radwaste Building) is limited consistent with the RG 1.143, Revision 2 dose acceptance criteria, given the safety classification RW-IIc assigned to the Radwaste Building. Finally, the revised license condition and FSAR language ensure that the applicant's procedures will conform with RG 1.143, Revision 2. Therefore, the September 12, 2013, response to RAI 11.02-5, including the proposed license condition, is acceptable. In addition, the response fully and adequately addresses COL Information Items 11.2-1 and 11.4-1. The staff confirmed that FSAR Sections 11.2.1.2.5.2, 11.4.6, and 13.5.2.2.5 were updated in accordance with the language in the September 12, 2013 letter.*

- WLS COL 11.2-2

The applicant provided additional information in WLS COL 11.2-2 to resolve COL Information Item 11.2-2, which states:

> The analysis performed to determine offsite dose due to liquid effluents is based upon the AP1000 generic site parameters included in Chapter 1 and Tables 11.2-5 and 11.2-6. The Combined License [COL] applicant will provide a site specific cost-benefit analysis to address the requirements of 10 CFR 50, Appendix I, regarding population doses due to liquid effluents.

The commitment was also captured as COL Action Item 11.2-2 in Appendix F of NUREG-1793, which states:

> The applicant will provide a site-specific cost-benefit analysis to demonstrate compliance with 10 CFR Part 50, Appendix I, regarding population doses due to liquid effluents.

The staff reviewed the resolution of COL Information Item 11.2-2 related to the cost-benefit analysis included under Section 11.2.3.5.2 of the WLS COL FSAR and issued RAI 13, Question 11.02-1. This RAI stated that the applicant needed to provide a detailed and plant-specific cost-benefit analysis. The applicant provided this analysis in a December 11, 2008, response to the RAI.

The results of the applicant's analysis showed that the lowest-cost option for liquid radwaste treatment system augments is a 20 gallons per minute (gpm) cartridge filter at $11,140 per year, which yields a threshold value of 11.14 person-rem total body or thyroid dose from liquid effluents. For AP1000 sites with population dose estimates less than 11.14 person-rem total body or thyroid dose from liquid effluents, no further cost-benefit analysis is needed to demonstrate compliance with 10 CFR Part 50, Appendix I, Section II.D. The total body (0.296 person-rem) and thyroid (0.393 person-rem) population doses provided by the applicant in WLS COL FSAR Table 11.2-204 are a small fraction of the threshold dose of

11.14 person-rem. Thus, the applicant concluded that the LWMS meets the as low as reasonably achievable (ALARA) requirements and requires no augments.

The staff performed an independent assessment using the population doses calculated by the staff (see following section) and the guidance in RG 1.110 and came to the same conclusion. As a result, the staff considers RAI 13, Question 11.02-1 closed and COL Information Item 11.2-2 resolved.

- WLS COL 2.4-5 and WLS COL 15.7-1

The applicant provided additional information in WLS COL 2.4-5 and WLS COL 15.7-1 to resolve COL Information Items 2.4-5 and 15.7-1.

COL Information Item 2.4-5 states:

> Combined License applicants referencing the AP1000 certified design will address site-specific information on the ability of the ground and surface water to disperse, dilute, or concentrate accidental releases of liquid effluents. Effects of these releases on existing and known future use of surface water resources will also be addressed.

The commitment was also captured as COL Action Item 2.4.1-1 in Appendix F of NUREG-1793, which states:

> The COL applicant will provide site specific information on the ability of the ground and surface water to disperse, dilute, or concentrate accidental releases of liquid effluents. The COL applicant will also address the effects of such releases on existing and known future use of surface water resources.

COL Information Item 15.7-1 states:

> Combined License applicants referencing the AP1000 certified design will perform an analysis of the consequences of potential release of radioactivity to the environment due to a liquid tank failure as outlined in Subsection 15.7.3.

The commitment was also captured as COL Action Item 15.3.8-1 in Appendix F of NUREG-1793, which states:

> The COL applicant will perform a site-specific analysis of the consequences of a potential release of radioactivity to the environment as a result of a liquid tank failure.

Section 2.4.13 of the WLS COL FSAR addresses accidental release of liquid effluents into ground and surface water. The applicant postulated a release of the contents of the effluent hold-up tank (or hold-up tank). BTP 11-6 provides guidance in assessing potential release of radioactive liquids at the nearest potable water supply located in an unrestricted area. BTP 11-6 further states the evaluation of the release should consider the use of water for direct human consumption or indirectly through animals (livestock watering), crops (agricultural irrigation), and food processing (water as an ingredient).

Evaluations performed by the applicant determined that the hold-up tanks have the greatest potential radionuclide inventory of all waste effluent system tanks. Spent resin storage tanks were considered by the applicant, but were excluded because most of the activity is bound to the spent resins and have minimal free water in them. Tanks inside the containment building were not considered because the containment building is a Seismic Category I structure. Other tanks were considered such as the monitor tanks, the hold-up tanks, and the chemical waste tank. The hold-up tanks were found to have the highest potential radioactivity concentration and highest volume. Based on groundwater flow directions shown on WLS COL FSAR Figure 2.4.12-204, Sheet 8, Unit 2 was analyzed because its tanks are nearer the points of exposure, which are Hold-Up Pond A and the Broad River. The contents spilled from the tank were assumed to enter the groundwater instantaneously. The source term developed by the applicant is as follows:

- Tritium source term concentration is 1.0 microcuries per gram taken from AP1000 DCD Table 11.1-8
- Corrosion product source terms Cr-51, Mn-54, Mn-56, Fe-55, Fe-59, Co-58, and Co-60 taken from AP1000 DCD Table 11.1-2
- Other radionuclide source terms taken from AP1000 DCD Table 11.1-2, multiplied by 0.12/0.25 to adjust the radionuclide concentrations to the required 0.12 percent failed fuel fraction outlined in BTP 11-6

The applicable regulatory acceptance criteria for a liquid waste tank failure is that the postulated failure would not result in radionuclide concentrations in excess of 10 CFR Part 20, Appendix B, Table 2, Column 2 values (effluent concentration limit ((ECL)) values) at the nearest source of potable water, where the ECL radionuclide concentrations correspond to a calculated dose of 50 mrem per year from the drinking water pathway. The applicant provided an analysis for compliance with 10 CFR Part 20, in Section 2.4.13. Compliance was demonstrated by calculating concentration/ECL ratios for all the radionuclides expected to enter the Broad River. The nearest potable surface water supply was in the Broad River. All the ratios for the location were determined to be less than one. In addition, the applicant demonstrated compliance with the requirement that the sum of the individual ratios of nuclide concentration to its ECL must be less than unity. The result of this calculation was that the sum of the ratios was 0.1 for the Broad River. The calculation is conservative in that no credit is taken for dilution of radionuclides in the Broad River caused by water flow and that the radionuclides are assumed to remain in the Broad River near groundwater discharge point for a period of 1 year.

The applicant's initial application did not include an analysis of pathways of exposure other than drinking water. The staff concluded that the analysis should also have discussed other applicable pathways, such as fish and crop irrigation. These pathways of exposure may concentrate radionuclide levels, leading to potentially higher dose contributions. In a December 3, 2008, response to RAI 34, Question 02.04.13-2, the applicant evaluated potential doses from the fish and irrigated foods pathways for the liquid tank failure analysis. Using RESRAD-OFFSITE dose methodology, the applicant calculated hypothetical doses to members of the public of 0.071 mrem/year from fish consumption and 0.244 mrem/year from irrigated crops consumption. The staff reviewed the applicant's analysis and performed an independent evaluation of the fish and irrigated crop pathways. These evaluations are presented below.

The staff applied the dose calculational methodology of RG 1.109, using the applicant's conservatively evaluated maximum concentration of radionuclides in the Broad River assuming no additional dilution in the river. Using this methodology, the staff calculated hypothetical doses of 0.14 mrem/year for fish consumption and 0.043 mrem/year for irrigated crops ingestion. These doses are sufficiently consistent with those calculated by the applicant to constitute independent confirmation, with differences attributable to modeling assumptions.

As the above analyses for fish and irrigated crop ingestion show, doses resulting from the failure of a waste hold-up tank would be a small fraction of the established regulatory limit. The sum of the fish consumption and irrigated vegetable pathways with the drinking water pathway yields a hypothetical maximum individual dose of approximately 5 mrem assuming a full year exposure time. This total is a small fraction of the 50 mrem/year dose criterion for the liquid tank failure analysis.

Based on the above evaluations and the applicant's analysis in the WLS COL FSAR, the staff finds potential doses to members of the public resulting from an accidental release of liquid effluents meets Acceptance Criterion No. 5 in NUREG-0800 and the referenced BTP 11-6; therefore, the staff considers RAI 34, Question 02.04.13-2 closed and COL Information Items 2.4-5 and 15.7-1 resolved. A complete evaluation at the point of the dose receptor is presented in SER Section 2.4.13, "Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters."

- WLS COL 11.5-3

The applicant provided additional information in WLS COL 11.5-3 to resolve the COL responsibilities as set forth in Section 11.5.7 of the AP1000 DCD, which states:

> The Combined License applicant is responsible for addressing the 10 CFR Part 50, Appendix I, Sections II.A and II.D guidelines for maximally exposed offsite individual doses and population doses via liquid and gaseous effluents.

The commitment was also captured as COL Action Item 11.5-3 in Appendix F of NUREG-1793, which states:

> The COL applicant is responsible for addressing the guidelines of Appendix I to 10 CFR Part 50, as they relate to maximally exposed offsite individual doses and population doses attributable to liquid and gaseous effluents.

In WLS COL FSAR Section 11.2.3.5, the applicant discussed the methods used to assure that individual and estimated population doses are maintained ALARA in accordance with 10 CFR Part 50, Appendix I (this information is also applicable to WLS COL FSAR Sections 11.3.3.4 and 11.4).

The staff reviewed the applicant's response to WLS COL 11.5-3 related to compliance with 10 CFR Part 50, Appendix I, Sections II.A and II.D and issued RAI 13, Questions 11.02-3 and 11.02-4. In RAI 13, Question 11.02-3, the staff requested that the applicant provide the details of the individual and population dose analysis. In RAI 13, Question 11.02-4, the staff questioned the applicant's assumption concerning the elimination of the irrigation exposure pathway.

In a December 11, 2008, response to RAI, Questions 11.02-3 and 11.02-4, the applicant provided a description of the required model assumptions and input parameters needed to run LADTAP II computer codes and justification for excluding the irrigation exposure pathway to calculate doses.

Using radiological exposure models based on RG 1.109 and the LADTAP II computer program (NUREG/CR-4013, "LADTAP II - Technical Reference and User Guide," April 1986), the applicant calculated the estimated doses to a hypothetical maximally exposed individual (MEI) of the public and to the population within 80 km (50 mi) from the postulated liquid effluents discharged.

WLS COL FSAR Tables 11.2-201 and 11.2-202 include liquid pathway parameters used as input to the dose calculation, including discharge flow rate, site-specific dilution factors, transit-times to receptors, consumption factors for fish and water, and recreational usage data for the Broad River. The analysis assumed a completely mixed impoundment model to calculate dilution of the radioactive effluent by the Broad River. Given the proximity of the discharge structure to the Ninety-Nine Islands Hydroelectric Dam, which impounds the "run-of-the-river" Ninety-Nine Islands Reservoir, the diffusion characteristics of the discharge piping and the river flow velocity, the staff viewed the mixing model as a discharge into the river fully mixing with the river flow. The downstream exposure pathways would then effectively see a dilution of 189.4 (ratio of flow rate through the dam divided by liquid effluent discharge flow rate). The result for both individual and population doses were the same as in the applicant's analysis. WLS COL FSAR Tables 11.2-203 and 11.2-204 list the liquid pathway doses to the MEI and surrounding population, respectively.

The applicant calculated a maximum individual annual dose (per unit) to the adult total body of 6.09E-04 mSv (0.0609 mrem) and a maximum annual individual dose (per unit) to the child liver of 7.75E-04 mSv (0.0775 mrem). The applicant compared the MEI doses with the 10 CFR Part 50, Appendix I, Section II.A criteria and showed the doses to be well below the limits of 3 mrem to the total body and 10 mrem to any organ.

The calculated annual population doses listed in WLS COL FSAR Table 11.2-204 are 2.96E-03 person-Sv (0.296 person-rem) to the total body (per unit), and 3.93E-03 person-Sv (0.393 person-rem) to the thyroid (per unit). The applicant uses the population doses in the cost-benefit analysis previously described in this report.

In the response to RAI 13, Question 11.02-3, the applicant explained the derivation of values used for population water use, sport fish harvest, commercial fish harvest, and recreational time spent on the river. The staff reviewed the derivation of these values and found them to be reasonable upper bound estimates. Consequently, the staff used the applicant's values in its independent dose estimation.

In the response to RAI 13, Question 11.02-4, the applicant stated that consumption of most of an individual's annual intake of vegetables from a vegetable garden irrigated with public water was not regarded as either a pathway that fell within a reasonable deviation from the average for the population, or a pathway unique to the WLS site that was likely to contribute a dose increment equal to or greater than 10 percent of the total from all pathways considered in RG 1.109. Therefore, individual use of public water for garden irrigation was not considered in the determination of doses to the public from routine release of liquid reactor effluents from WLS Units 1 and 2. The applicant then provided a conservative dose analysis that concluded that the calculated individual dose associated with the irrigated, individual garden pathway contributes just 2 percent of the total body dose due to all liquid effluent pathways. The population dose did not include crop irrigation since it was not found to occur in the vicinity of the WLS site. Since the calculated dose does not have the potential to contribute 10 percent or more to individual or population doses, the applicant did not consider this pathway to be significant given the guidance of RG 1.109. The staff evaluated this response and concurred that this pathway is not significant. Therefore, the staff finds that the doses to the MEI and population associated with consuming vegetables watered by public drinking water are not included in the dose analyses.

The staff performed an independent assessment using the LADTAP II computer code and compared results to the applicant's and the Appendix I criteria. The modeling assumptions used by the staff for the MEI and population dose calculations, as shown in Table 11.2-1 of this SER, were consistent with the applicant's. Modeling parameter values, as shown in Table 11.2-2 of this SER, were also consistent with the applicant's. The results of the staff's calculations were consistent with those of the applicant.

Table 11.2-3 of this SER compares the resulting dose estimates between the applicant's analysis and the 10 CFR Part 50, Appendix I criteria. This table shows that all doses are below the Appendix I criteria. The staff concludes that the applicant has provided a bounding assessment demonstrating its capability to comply with the regulatory requirements in 10 CFR Part 20 and 10 CFR Part 50, Appendix I and, therefore, considers COL Information Item 11.5-3 resolved.

Based on the above evaluation, RAIs 11.2-3 and 11.2-4 are closed.

Supplemental Information

The following portion of this technical evaluation section is reproduced from Section 11.2.4 of the VEGP SER:

> *The following portion of this technical evaluation section is reproduced from Section 11.2.4 of the BLN SER:*
>
> - *STD SUP 11.2-1*
>
> *The applicant provided supplemental information in BLN COL FSAR Section 11.2.3.6, "Quality Assurance," addressing the quality assurance program to be applied to the liquid waste system and stated that the program complies with the guidance presented in RG 1.143.*
>
> *The NRC staff reviewed this supplemental quality assurance information included in BLN COL FSAR Section 11.2.3.6 and finds that this supplemental statement commits the applicant to the regulatory positions in RG 1.143 related to quality assurance and is acceptable.*

- WLS SUP 11.2-2

The applicant stated that the only liquid effluent site interface parameter outside the Westinghouse scope is the release point to the Broad River. The staff finds this statement correct because the release point to the environment of liquid radioactive effluent is site-specific and to the Broad River.

- WLS SUP 11.2-3

The applicant provided supplemental information in WLS SUP 11.2-3 related to the exterior radwaste discharge piping. The information stated that the exterior radwaste discharge piping is enclosed within a guard pipe and monitored for leakage and that liquid radwaste effluent will be discharged to the Broad River with plant discharge.

This item is related to 10 CFR 20.1406 and is addressed in Section 12.3 of this SER.

License Condition

The applicant proposed the following license condition:

> Prior to initial fuel load, the licensee shall develop, implement, and maintain procedural controls limiting radionuclide inventory in each of the Radwaste Building Monitor Tanks, and separately in each of up to three (3) Radwaste Building mobile radwaste processing systems to below A2 quantities for radionuclides specified in Appendix A to 10 CFR Part 71 (Tables A-1 and A-3), as described in FSAR Section 13.5.2.2.5. The procedures shall also ensure that any additional equipment located in the RWB is limited to the A_2 quantities and that the total cumulative radioactive inventory contained in unpackaged wastes (including liquid waste, wet waste, solid waste, gaseous waste, activated or contaminated metals and components, and contaminated waste present at any time in the Radwaste Building) is limited so that an unmitigated release, occurring over a 2-hour time period, would not result in a dose of greater than 500 millirem at the protected area boundary or an unmitigated exposure, occurring over a 2 hour time period, would not result in a dose of greater than 5 rem to site personnel located 10 feet from the total cumulative radioactive inventory.

The evaluation of this license condition is discussed above in the evaluation of WLS COL 11.2-1 and WLS COL 13.5-1.

Demonstrating Compliance with 10 CFR 20.1301(e)

Pursuant to 10 CFR 20.1301(e), the NRC-licensed facilities must comply with the U.S. Environmental Protection Agency's (EPA) generally applicable environmental radiation standards of 40 CFR Part 190 for facilities that are part of the fuel cycle. The EPA annual dose limits are 0.25 mSv (25 mrem) to the whole body, 0.75 mSv (75 mrem) to the thyroid, and 0.25 mSv (25 mrem) to any other organ. Meeting the requirements of 10 CFR 20.1301(e) requires the consideration of all potential sources of external radiation and radioactivity, including liquid and gaseous effluents and external radiation exposures from buildings, storage tanks, radioactive waste storage areas, and N-16 skyshine from boiling-water reactor (BWR) turbine buildings. The EPA standards apply to the entire site or facility, whether it has single or multiple units.

The staff's review of the WLS COL FSAR revealed that the applicant did not provide any information demonstrating compliance with 10 CFR 20.1301(e). Therefore, in RAI 13, Question 11.02-2, the staff requested that the applicant demonstrate compliance with the EPA standard.

The applicant provided the demonstration by summing the annual individual liquid and gaseous effluent doses for WLS Units 1 and 2. In a December 11, 2008, response to RAI 13, Question 11.02-2, the applicant listed the results in WLS COL FSAR Table 11.2-206. Table 11.2-4 of this SER lists these dose summations and compares them to the dose requirements in 40 CFR Part 190. The expected doses are below the EPA limits, therefore, the staff finds that the requirement of 10 CFR 20.1301(e) is met. The staff verified that Table 11.2-206 has been incorporated in the WLS COL FSAR. Accordingly, the staff considers RAI 13, Question 11.02-2 resolved.

Demonstrating Compliance with 10 CFR 20.1302

Pursuant to 10 CFR 20.1302, the annual average concentration of radioactive material released in liquid effluents at the boundary of the unrestricted area must not exceed the values specified in 10 CFR Part 20, Appendix B, Table 2. The applicant demonstrated compliance with this requirement by referencing the AP1000 DCD. AP1000 DCD Section 11.2.3.4 shows that even at the Technical Specification limit for percent failed fuel defects, the nominal blowdown flow provides sufficient dilution to ensure that the expected effluent release concentrations will be less than those specified in 10 CFR Part 20, Appendix B, Table 2 including the provisions of Appendix B, Table 2, footnote 4 for radionuclide mixtures.

In NUREG-1793, the staff evaluated and accepted the conclusions of Section 11.2.3.4 of the AP1000 DCD. Based on this acceptance, the staff concludes that the applicant complies with 10 CFR 20.1302.

Demonstrating Compliance with 10 CFR 20.1406

Pursuant to 10 CFR 20.1406, the applicant must provide a description of how facility design and procedures for operation will minimize, to the extent practicable, contamination of the facility and the environment; facilitate eventual decommissioning; and minimize, to the extent practicable, the generation of radioactive waste. The staff finds that the applicant demonstrated compliance with this requirement by incorporating by reference the design descriptions provided in the AP1000 DCD and providing the description of operating programs in WLS COL FSAR Sections 12.3 and 12.5. The staff's evaluation and conclusion pertaining to compliance with 10 CFR 20.1406 are included in Section 12.3 of this SER.

Demonstrating Compliance with 10 CFR Part 50, Appendix I

Pursuant to 10 CFR Part 50, Appendix I, Sections II.A and II.D, the applicant is responsible for addressing the requirements for dose objectives in controlling doses to a hypothetical maximally exposed member of the public and populations living near the proposed nuclear power plant. The requirements define dose objectives for liquid effluents, and require a cost-benefit analysis in justifying installed processing and treatment equipment of the LWMS, including any augmentation to the design in complying with 10 CFR Part 50, Appendix I. The staff finds that the applicant has demonstrated compliance with 10 CFR Part 50, Appendix I, Section II.A and II.D requirements by performing the required cost-benefit analysis through WLS COL 11.2-2, and performed the require dose compliance through WLS COL 11.5-3. The staff independently verified the results of the cost-benefit analysis and compliance with the dose objectives and finds that the applicant is in compliance with 10 CFR Part 50, Appendix I, Sections II.A and II.D. See Tables 11.2-1, 11.2-2, and 11.2-3 of this SER for a list of parameters and the dose comparison to the limits specified in the discussion above.

Table 11.2-1 Comparison of Important Modeling Assumptions

Pathways and Parameters	Application	NRC Staff's Analysis
Drinking water pathway for maximally exposed individual (MEI) and population	Yes	Yes
Fish ingestion pathway for MEI and population	Yes	Yes
Recreational use of river for MEI and population	Yes	Yes
Irrigation pathway for the MEI	No	No
Surface Water Dilution Model	Completely mixed impoundment	Fully mixed with river flow

Table 11.2-2 Modeling Parameter Values*

Parameter	Value	Basis
Annual radionuclide release (Ci/yr)	Multiple values	AP1000 DCD Table 11.2-7
Effluent discharge rate (cfs)	13.4	WLS COL FSAR Table 11.2-202
Annual average river flow for the MEI doses (cfs)	2,538	WLS COL FSAR Table 11.2-201
Dilution factors	1	WLS COL FSAR Table 11.2-202
Transit time (hr)	14.2 hr for drinking water 0 for other	WLS COL FSAR Table 11.2-202
Reconcentration model**	Complete mixing	WLS COL FSAR Table 11.2-201

* The staff used LADTAP II default values for parameters not listed in the table

** The staff assumed full mixing with the river rather than an impoundment model

Table 11.2-3 Comparison of Maximum Individual Doses (mrem/yr

Organ/Body	Application*	10 CFR Part 50, Appendix I, Section II.A
Liver	7.75E-02	10
Total Body	6.09E-02	3
Thyroid	5.32E-02	10

Table 11.2-4 Comparison of Maximum Individual Doses to 40 CFR Part 190 (mrem/yr)

Organ/Body	Application*	40 CFR Part 190
Total Body	3.74	25
Thyroid	20.0	75
Other Organ (Child Bone)	9.05	25

* Taken from WLS COL FSAR Table 11.3-206

11.2.5 Post Combined License Activities

For the reasons discussed in the technical evaluation section above, the staff finds the following license condition acceptable:

- License Condition (11-1) – Before initial fuel load, the licensee shall develop, implement, and maintain procedural controls limiting radionuclide inventory in each of the Radwaste Building Monitor Tanks, and separately in each of up to three (3) Radwaste Building mobile radwaste processing systems to below A2 quantities for radionuclides specified in Appendix A to 10 CFR Part 71 (Tables A-1 and A-3), as described in FSAR Subsection 13.5.2.2.5. The procedures shall also ensure that any additional equipment located in the RWB is limited to below A2 quantities and that the total cumulative radioactive inventory contained in unpackaged wastes (including liquid waste, wet waste, solid waste, gaseous waste, activated or contaminated metals and components, and contaminated waste present at any time in the Radwaste Building) is limited so that an unmitigated release, occurring over a two hour time period, would not result in a dose of greater than 500 millirem at the protected area boundary or an unmitigated exposure, occurring over a two hour time period, would not result in a dose of greater than 5 rem to site personnel located 10 feet from the total cumulative radioactive inventory.

11.2.6 Conclusion

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to the LWMS, and that there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff evaluated the additional COL information (STD COL 11.2-1, STD COL 11.2-2, WLS COL 11.2-1, WLS COL 13.5-1, WLS COL 11.2-2, WLS COL 11.5-3, and STD SUP 11.2-1, WLS COL 2.4-5, WLS COL 15.7-1, WLS SUP 11.2-2, WLS SUP 11.2-3) in the application against the relevant NRC regulations, acceptance criteria defined in NUREG-0800, Section 11.2, and other NRC regulatory guides. The applicant has satisfactorily addressed all RAIs related to Section 11.2.

The staff verified that the applicant had provided sufficient information and that the review and calculations support the conclusions that follow. The staff concludes that the LWMS (as a permanently installed system or in combination with mobile systems) includes the equipment necessary to control releases of radioactive materials in liquid effluents in accordance with GDC 60 and 61 of Appendix A to 10 CFR Part 50 and the requirements of 10 CFR 50.34a. Therefore, the staff concludes that the design of the LWMS is acceptable and meets the requirements of 10 CFR 20.1301(e), 10 CFR 20.1302, 10 CFR 20.1406, 10 CFR 50.34a, GDC 60 and 61, and Appendix I to 10 CFR Part 50.

11.3 Gaseous Waste Management System

11.3.1 Introduction

The gaseous waste management system (GWMS) is designed to control, collect, process, handle, store, and dispose of gaseous radioactive waste generated as the result of normal operation, including anticipated operational occurrences.

11.3.2 Summary of Application

Section 11.3 of the WLS COL FSAR, Revision 11, incorporates by reference Section 11.3 of the AP1000 DCD, Revision 19.

In addition, in WLS COL FSAR Section 11.3, the applicant provided the following:

<u>AP1000 COL Information Items</u>

- STD COL 11.3-1

The applicant provided additional information in STD COL 11.3-1 to resolve COL Information Item 11.3-1 (COL Action Item 11.3-1) regarding gaseous radwaste cost-benefit analysis methodology.

- WLS COL 11.3-1

The applicant provided additional information in WLS COL 11.3-1 to resolve COL Information Item 11.3-1 (COL Action Item 11.3-1). The additional information addresses the estimated doses to the public from the gaseous waste system and the associated cost-benefit analysis in WLS COL FSAR Section 11.3.3.4.

- WLS COL 11.5-3

The applicant provided additional information in WLS COL 11.5-3 to resolve COL Information Item 11.5-3 (COL Action Item 11.5-3). The additional information addresses compliance with 10 CFR Part 50, Appendix I, Sections II.B and II.C related to operation of the gaseous waste system in WLS COL FSAR Section 11.3.3.4.

<u>Supplemental Information</u>

- STD SUP 11.3-1

The applicant added supplemental information in WLS COL FSAR Section 11.3.3.6 to address the QA program to be applied to the GWMS.

- STD SUP 11.3-2

The applicant added supplemental information in WLS COL FSAR Section 11.3.3 to address the gaseous effluent site interface parameter.

11.3.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the regulatory basis for acceptance of the supplementary information on the GWMS is established in:

- 10 CFR 20.1301(e), as it relates to compliance with 40 CFR Part 190
- 10 CFR 20.1302, “Compliance with dose limits for individual members of the public”
- 10 CFR 20.1406, “Minimization of contamination”
- 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” GDC 3, “Fire protection”

- 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” GDC 60, “Control of releases of radioactive materials to the environment”
- 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” GDC 61, “Fuel storage and handling and radioactivity control”
- 10 CFR Part 50, Appendix I, Sections II.B, II.C and II.D as it relates to gaseous effluent dose objectives and associated cost-benefit analysis
- 10 CFR 50.34a, “Design objectives for equipment to control releases of radioactive material in effluents – nuclear power reactors”
- 10 CFR 52.80(a), as it relates to those inspections, tests, and analyses that the licensee shall perform, and the acceptance criteria that are necessary to show the facility shall be constructed and operated in conformity with the COL.

Guidance for meeting these requirements is in the following:

- Regulatory Position C.2 of RG 1.143, Revision 2
- RG 1.109, Revision 1
- RG 1.110
- RG 1.111, “Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Nuclear Power Reactors,” Revision 1
- RG 4.21

The acceptance criteria associated with the GWMS are given in Section 11.3 of NUREG-0800, including BTP 11-5.

11.3.4 Technical Evaluation

The staff reviewed Section 11.3 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The staff’s review confirmed that the information in the application and incorporated by reference addresses the required information relating to the GWMS. The results of the staff’s evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff’s review of this application included the following COL information and supplementary items:

- STD COL 11.3-1, Gaseous Radwaste Cost-Benefit Analysis Methodology
- WLS COL 11.3-1, Cost-Benefit Analysis of Population Doses
- WLS COL 11.5-3, 10 CFR Part 50, Appendix I, Sections II.B and II.C

- STD SUP 11.3-1, Supplemental Information on Quality Assurance

- STD SUP 11.3-2, Supplemental Information on Gaseous Effluent Site Interface Parameters

In addition to the above items, the staff reviewed the entire section against Section 11.3 of NUREG-0800 to determine if the information in WLS COL FSAR Section 11.3 met the regulatory requirements in the regulations stated above (Section 11.3.3 of this SER) and NUREG-0800 acceptance criteria. The relevant NUREG-0800 acceptance criteria are as follows:

- The GWMS should have the capability to meet the dose design objectives and should include provisions to treat gaseous radioactive wastes, such that the following is true:

    A. The calculated annual total quantity of all radioactive materials released from each reactor to the atmosphere will not result in an estimated annual external dose from gaseous effluents to any individual in unrestricted areas in excess of 0.05 mSv (5 mrem) to the total body or 0.15 mSv (15 mrem) to the skin. RGs 1.109 and 1.111 provide acceptable methods for performing this analysis.

    B. The calculated annual total quantity of radioactive materials released from each reactor to the atmosphere will not result in an estimated annual air dose from gaseous effluents at any location near ground level which could be occupied by individuals in unrestricted areas in excess of 0.01 centiGray (cGy) (10 millirads (mrad)) for gamma radiation or 0.02 cGy (20 mrad) for beta radiation. RG 1.109 and RG 1.111 provide acceptable methods for performing this analysis.

    C. The calculated annual total quantity of radioiodines, carbon-14, tritium, and all radioactive materials in particulate form released from each reactor at the site in effluents to the atmosphere will not result in an estimated annual dose or dose commitment from such releases for any individual in an unrestricted area from all pathways of exposure in excess of 0.15 mSv (15 mrem) to any organ. RG 1.109 and RG 1.111 provide acceptable methods for performing this analysis.

    D. In addition to A, B, and C, above, the GWMS should include all items of reasonably demonstrated technology that, when added to the system sequentially and in order of diminishing cost-benefit return, for a favorable cost-benefit ratio, can effect reductions in dose to the population reasonably expected to be within 80 km (50 mi) of the reactor. RG 1.110 provides an acceptable method for performing this analysis.

    E. The concentrations of radioactive materials in gaseous effluents released to an unrestricted area should not exceed the limits specified in 10 CFR Part 20, Appendix B, Table 2, Column 1.

    F. The regulatory position in RG 1.143 is met, as it relates to the definition of the boundary of the GWMS, beginning at the interface from plant systems to the point of controlled discharges to the environment as defined in the Offsite Dose Calculation Manual (ODCM), or at the point of storage in holdup tanks or decay beds for gaseous wastes produced during normal operation and anticipated operational occurrences.

- System designs should describe features that will minimize, to the extent practicable, contamination of the facility and environment; facilitate eventual decommissioning; and minimize, to the extent practicable, the generation of radioactive waste in accordance with RG 1.143, for gaseous wastes produced during normal operation and anticipated operational occurrences, and the requirements of 10 CFR 20.1406 or the DC application, update in the SAR, or the COL application to the extent not addressed in a referenced certified design.

- BTP 11-5, as it relates to potential releases of radioactive materials (noble gases) as a result of postulated leakage or failure of a waste gas storage tank or off-gas charcoal delay bed.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff completed its review and finds the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) contains evaluation material from the SER for the BLN Units 3 and 4 COL application.

AP1000 COL Information Items

The following portion of this technical evaluation section is reproduced from Section 11.2.4 of the VEGP SER:

- *STD COL 11.3-1*

The discussion of VEGP COL 11.3-1 addresses the site-specific cost-benefit analysis performed to address the requirements of 10 CFR Part 50, Appendix I, regarding population doses due to gaseous effluents. The applicant provided additional information in STD COL 11.3-1 to resolve COL Information Item 11.3-1 with regard to the cost-benefit analysis methodology.

The NRC staff reviewed the resolution of COL Information Item 11.3-1 related to the cost-benefit analysis methodology described in VEGP FSAR Section 11.3.3.4 *and concluded that the methodology used for the analysis was consistent with the guidance of RG 1.110 and was, therefore, acceptable.*

- WLS COL 11.3-1

The applicant provided additional information in WLS COL 11.3-1 to resolve COL Information Item 11.3-1, which states:

> The analysis performed to determine offsite dose due to gaseous effluents is based upon the AP1000 generic site parameters included in Chapter 1 and DCD Tables 11.3-1, 11.3-2 and 11.3-4. The Combined License applicant will provide a site specific cost-benefit analysis to demonstrate compliance with 10 CFR 50, Appendix I, regarding population doses due to gaseous effluents.

The commitment was also captured as COL Action Item 11.5-3 in Appendix F of NUREG-1793, which states:

> The COL applicant will provide a site-specific cost-benefit analysis to demonstrate compliance with 10 CFR 50, Appendix I, regarding population doses due to gaseous effluents.

The staff reviewed the resolution of COL Information Item 11.3-1 related to the cost-benefit analysis included under Sections 11.3.3.4.2 and 11.3.5.1 of the WLS COL FSAR and issued RAI 14, Question 11.03-1 because the Nuclear Energy Institute (NEI) Template 07-11, "Generic FSAR Template Guidance for Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors," cited by Duke Energy Carolinas, LLC had been withdrawn by NEI from further consideration. In RAI 14, Question 11.03-1, the staff requested that the applicant provide a detailed and plant-specific cost-benefit analysis.

In response to RAI 14, Question 11.03-1, the applicant performed a site-specific analysis to determine whether the offsite dose due to gaseous effluents is bounded by the AP1000 site parameters included in Chapter 1 and Tables 11.3-1, 11.3-2 and 11.3-4 of the AP1000 DCD. The applicant discussed the site-specific cost-benefit analysis in WLS COL FSAR Section 11.3.3.4 to address the requirements of 10 CFR Part 50, Appendix I, Section II.D, regarding population doses due to gaseous effluents. The dose and dose rate to man was calculated using the GASPAR II computer code, which is based on the methodology presented in RG 1.109. On December 20, 2012, the applicant submitted updates to calculations for the cost-benefit analysis in which the staff evaluated the results in conjunction with the results provided before. These updates are due to the change in location of the nuclear island for applicant's site where the relocation of Lee Unit 1 is 15.24 m (50 feet) east and 20.12 m (66 feet) south and Unit 2 was moved 20.12 m (66 feet) south from the original placement. This movement changed the distances to receptor locations of various MEI locations for the gaseous effluents and updated meteorological data caused changes to the cost-benefit analysis. The applicant's analysis showed that the lowest-cost option for gaseous radwaste treatment system augments is the steam generator flash tank vents to the main condenser at $6,320 per year. The population doses, 6.32 person-rem total body per reactor and 9.80 person-rem thyroid per reactor, are given in the WLS COL FSAR Table 11.3-204. Assuming 100 percent efficiency of this augment, the resulting cost per person-rem is determined by dividing the cost of the augment by the population dose, or $1,264 per person-rem total body ($6,320/5.00 person-rem). The cost per person-rem for thyroid, $9,800 (or 9.80 person-rem), exceed the cost value for total body, $6,800 (or 6.80 person-rem), so only augments for which the total annual costs is below $9,800 require further analysis by the applicant.

The applicant's further analysis of the population thyroid dose examined a number of potential gaseous radwaste treatment system augments based on their estimated 9.80 person-rem/year thyroid dose (and, therefore, those augments with a "Total Annual Cost" of less than $9,800). In order of decreasing total annual cost (TAC), the applicant evaluated:

- Pressurized-water reactor (PWR) air ejector charcoal/high efficiency particulate air (HEPA) filtration unit with a TAC of $9,140, which would have to remove at least 9.14 of the 9.80 person-rem (thyroid) to be cost-beneficial. The applicant stated that based on the system design, no radio-iodine is released through the condenser air removal (off-gas) system design; therefore, this augment does not affect the radio-iodine discharged by the plant which accounts for 4.85 person-rem in the thyroid population dose. Since it would be impossible to achieve the necessary dose reduction, this augment is not cost-beneficial because it exceeds the cost-benefit ratio of Section II.D of 10 CFR Part 50, Appendix I.

- Three ton charcoal absorber with a TAC of $8,770, which would have to remove at least 8.77 of the 9.80 person-rem (thyroid) to be cost-beneficial. It is assumed that this augment would be appended to the gaseous radwaste system where it would increase the delay time of noble gases exiting the existing activated carbon delay beds. The applicant stated that no radio-iodine is released through the gaseous radwaste system; therefore, this augment does not affect the radio-iodine discharged by the plant, which accounts for 4.85 person-rem in the thyroid population dose. Since it would be impossible to achieve the necessary dose reduction, this augment is not cost-beneficial because it exceeds the cost-benefit ratio of 10 CFR Part 50, Appendix I, Section II D.

- Main condenser vacuum pump charcoal/HEPA filtration systems with a TAC of $7,690, which would have to remove at least 7.69 of the 9.80 person-rem (thyroid) to be cost-beneficial. The applicant stated that based on the system design, no radio-iodine is released through the condenser air removal system; therefore, this augment does not affect the radio-iodine discharged by the plant which accounts for 4.85 person-rem in the thyroid population dose. Since it would be impossible to achieve the necessary dose reduction, this augment is not cost-beneficial because it exceeds the cost-benefit ratio of 10 CFR Part 50, Appendix I, Section II D.

- 1,000 cubic feet per minute (cfm) charcoal/HEPA filtration systems with a TAC of $7,580, which would have to remove at least 7.58 of the 9.80 person-rem (thyroid) to be cost-beneficial. The applicant stated that even assuming that this rather small capacity augment could be placed in the ventilation system at some point that would eliminate all radio-iodine and particulate releases, it would not be effective in reducing the noble gas releases, the carbon-14 release, or the airborne tritium release, all of which account for 4.67 person-rem in the thyroid population dose. Since it would be impossible to achieve the necessary dose reduction, this augment is not cost-beneficial because it exceeds the cost-benefit ratio of 10 CFR Part 50, Appendix I, Section II D.

- 600 ft^3 gas decay tank with a TAC of $7,460, which would have to remove at least 7.46 of the 9.80 person-rem (thyroid) to be cost-beneficial. This augment would be part of a conventional high pressure waste gas holding system. The applicant stated that based on the system design, no radio-iodine is released through this system; therefore, this augment does not affect the radio-iodine discharged by the plant, which accounts for 4.85 person-rem in the thyroid population dose. Since it would be impossible to achieve the necessary dose reduction, this augment is not cost-beneficial because it exceeds the cost-benefit ratio of 10 CFR Part 50, Appendix I, Section II D.

- Steam generator flash tank vent to main condenser with a TAC of $6,320, which would have to remove at least 6.32 of the 9.80 person-rem (thyroid) to be cost-beneficial. Addition of this augment presumes that the design already includes a steam generator flash tank; the augment being evaluated is the installation of vent piping and instrumentation from the tank to the main condenser. However, the system design does not include a steam generator flash tank; therefore, the TAC of $6,320 is underestimated. Additionally, the AP1000 design includes steam generator blowdown heat exchangers that provide cooling of the blowdown fluid and prevent flashing prior to the blowdown flow entering the main condenser. Therefore, this augment would not provide any additional dose reduction, and this augment is not cost-beneficial because it exceeds the cost-benefit ratio of 10 CFR Part 50, Appendix I, Section II D.

Based on the above evaluation, the applicant concluded that none of the radwaste augments are cost-beneficial in reducing the annual thyroid dose from gaseous effluents for WLS. A summary of the population dose break down by source can be found in Table 11.3-1 of this SER and a comparison of the application's and the staff's analysis for derived population dose results is found in Table 11.3-3 of this SER.

The staff reviewed this evaluation, including the evaluation due to the movement of the nuclear island, and concurred with its results. Thus, the staff concluded that the GWMS meets ALARA requirements and requires no augments. Therefore, the staff considers COL Information Item 11.3-1 resolved. Accordingly, the staff considers RAI 14, Question 11.03-1 resolved.

- WLS COL 11.5-3

The applicant provided additional information in WLS COL 11.5-3 to resolve COL Information Item 11.5-3, which states:

> The Combined License applicant is responsible for addressing the 10 CFR 50, Appendix I guidelines for maximally exposed offsite individual doses and population doses via liquid and gaseous effluents.

The commitment was also captured in COL Action Item 11.5-3 in Appendix F of NUREG-1793, which states:

> The COL applicant is responsible for addressing the guidelines of Appendix I to 10 CFR Part 50, as they relate to maximally exposed offsite individual doses and population doses attributable to liquid and gaseous effluents.

The staff reviewed the resolution of COL Information Item 11.5-3 related to the compliance with 10 CFR Part 50, Appendix I as presented in Section 11.3.3.4 of the WLS COL FSAR. In RAI 14 and RAI 109, Questions 11.03-2 and 11.03-4, respectively, and in RAI 110, Question 02.03.05-6(b), the staff requested that the applicant provide the details of the individual and population dose analyses. RAI 109, Question 11.03-4 and RAI 110, Question 02.03.05-6(b) were asked as part of the updated calculations that were provided to the staff as part of the applicant's updated submittal, which was submitted on December 20, 2012. As discussed before, the updates are due to the change in location of the nuclear island for the applicant's site. This movement changed the distances to receptor locations of various MEI locations for the gaseous effluents and caused changes to the calculated MEI doses while also having the gaseous effluent results being adjusted due to updated meteorological data.

In a May 2, 2013, response to RAI 14, Question 11.03-2, the applicant provided an evaluation of the impacts from gaseous effluent releases by considering the probable pathways to individuals and populations near the proposed new units. The applicant estimated the total-body and organ dose to the MEI from the gaseous effluent release pathways, and also calculated a collective total body and organ dose for the population within 80 km (50 mi) of the WLS site. The estimates of the maximum doses to the public are based on the AP1000 reactor's normal operational effluent releases, as discussed in the AP1000 DCD. The applicant evaluated the impact of these doses by comparing them to applicable regulatory limits.

If built, the postulated two new units at the WLS site would release gaseous effluents into the atmosphere. The applicant calculated doses for several airborne pathways, including direct exposure to a radioactive plume, direct exposure to radioactivity deposited on the ground, inhalation of airborne radioactivity and ingestion of contaminated agricultural products including, vegetables, milk, and meat. The applicant assumed that the MEI consumes both cow and goat's milk, while the population consumes only cow's milk.

In the response to RAI 14, Question 11.03-2, the applicant provided a description of all required model assumptions and input parameters needed to run the GASPAR II computer code. Using radiological exposure models based on RG 1.109, Revision 1 and the GASPAR II computer program (NUREG/CR-4653, "GASPAR II - Technical Reference and User Guide," March 1987), the applicant calculated the estimated doses to a hypothetical MEI of the public and to the population within 80 km (50 mi) from the postulated gaseous effluents discharged.

In an August 8, 2013, response to RAI 109, Question 11.03-4, the applicant provided clarification on dose results and the methodology used to meet compliance by making changes to the application. The applicant first made WLS COL FSAR Table 11.3-205 consistent with WLS COL FSAR Table 11.3-202 and added a clarifying footnote to Table 11.3-205 to clarify the pathways and adopted parameters in deriving the dose results. Clarifying footnotes were also added to WLS COL FSAR Tables 11.3-206 and 11.3-207 to show consistency and clarity with the NRC methodology described in RG 1.109 and revised dose results. The staff evaluated the response to RAI 109, Question 11.03-4 and agrees with the changes. The staff has confirmed the changes as a result of this RAI have been made and the staff also confirms through confirmatory calculations that the dose calculations the applicant has updated to the tables described in this response are consistent with the staff's request. As a result, the staff considers RAI 109, Question 11.03-4 resolved.

WLS COL FSAR Tables 2.3.5-287, 2.3.5-290, 2.3.5-291, and 2.3-292 include all the atmospheric dispersion and deposition factors used by the applicant to calculate individual and population doses. WLS COL FSAR Table 11.3-201 includes gaseous pathway parameters used as input to the dose calculation, including population data, and site-specific agricultural usage information. The applicant provided justifications for these parameter values in the response to RAI 14, Question 11.03-2. WLS COL FSAR Tables 11.3-202 and 11.3-204 list the gaseous pathway doses to the MEI and surrounding population, respectively. In RAI 110, Question 02.03.05-6(b), the staff requested that the applicant clarify the use of the Exclusion

Area Boundary (EAB) and the Site Boundary as the limiting boundary in the dose calculations. The applicant had previously used the EAB as the limiting boundary for MEI doses. The staff concluded that the applicant would need to create a new set of Site Boundary atmospheric dispersion values in order to adequately meet the dose objectives presented in 10 CFR Part 50, Appendix I. Specifically, referring to the section that states applicants need to provide sufficient information to show that the calculated annual dose at any location near ground level that could be occupied by individuals in unrestricted areas is below the limits specified in 10 CFR Part 50, Appendix I.

In a September 30, 2013, response to RAI 110, Question 02.03.05-6(b), the applicant provided updates to WLS COL FSAR Table 2.3-289 to add the Site Boundary Location for Units 1 and 2. This update then reflected changes to the dose calculations in WLS COL FSAR Section 11.3. Changes to WLS COL FSAR Section 11.3 are reflected in both the text and WLS COL FSAR Tables 11.2-206, 11.3-202, 11.3-203, 11.3-205, 11.3-206, and 11.3-207. The change to site boundary as the limiting boundary causes changes to the plume, ground and inhalation pathway doses, where the previously mentioned tables are all affected by this change and are reflected in the discussion of the dose results below. The analysis for staff's long-term atmospheric dispersion factors is found in WLS COL FSAR Chapter 2.3.5, "Long Term Diffusion Estimates."

The applicant calculated the gaseous pathway doses to the MEI. The results show for the worst-case location outside the exclusion boundary a gamma annual air dose of 0.00773 milliGray (mGy) or 0.773 mrad, a beta annual air dose of 0.0325 mGy or 3.25 mrad; a total annual body dose of 0.00732 mSv or .732 mrem and an annual skin dose of 0.0490 mSv or 4.90 mrem. Table 11.3-205 of the WLS COL FSAR was added, which lists the maximum annual organ dose (thyroid) of 0.0921 mSv or 9.21 mrem for the infant.

The calculated annual population doses listed in WLS COL FSAR Table 11.3-204 are 0.0500 person-Sv (5.00 person-rem) to the total body, and 0.0980 person-Sv (9.80 person-rem) to the thyroid. The applicant used the population doses in the cost-benefit analysis described in the WLS COL FSAR and evaluated in this SER.

The staff performed an independent assessment using the GASPAR II computer code and compared its results to the applicant's and the 10 CFR Part 50, Appendix I criteria. The staff notes that the modeling assumptions used and parameter values used were consistent with the applicant's.

In the response to RAI 14, Question 11.03-2, the applicant explained the derivation of values used for agricultural and usage parameters including the total production of vegetables, milk, and meat in the 8 km (5mi) area around the site. The staff evaluated and verified the derivation of these values and found them to be reasonable upper bound estimates. Consequently, the staff used the applicant's agricultural and usage values listed in WLS COL FSAR Table 11.3-201 for the dose estimation.

The staff evaluated and agreed with the approach taken by the applicant to calculate maximum annual individual doses from gaseous effluents. Using this same approach, the staff verified the individual doses in the WLS COL FSAR by independently running the GASPAR II computer code with the applicant's parameter values. Table 11.3-2 in this SER compares the resulting dose estimates from the applicant's analyses with the 10 CFR Part 50, Appendix I criteria. All doses are below the 10 CFR Part 50, Appendix I, Section II.B and II.C criteria.

The staff evaluated and agreed with the approach taken by the applicant to calculate population doses from gaseous effluents. Using this same approach, the staff evaluated the population doses in the WLS COL FSAR by independently running the GASPAR II computer code with the applicant's parameter values. The applicant then used these doses in a cost-benefit analysis for augments to the GWMS. Table 11.3-3 in this SER summarizes the results of the applicant's and staff's analysis of population doses. The staff has reviewed the application and confirms the changes made by the applicant in the FSAR. The staff considers RAI 14, Question 11.03-2 resolved. The staff considers RAI 110, Question 02.03.05-6(b) resolved.

The staff concluded that the information provided by the applicant for WLS COL 11.5-3 is acceptable. The staff finds that the applicant provided a bounding assessment demonstrating its capability to comply with the individual dose criteria in 10 CFR Part 20 and 10 CFR Part 50, Appendix I. In addition, the staff finds the applicant's calculation of the population dose to be appropriate for use in assessing the cost-benefit requirements in 10 CFR Part 50, Appendix I. Therefore, the staff considers COL Information Item 11.5-3 resolved.

Supplemental Information

The following portion of this technical evaluation section is reproduced from Section 11.3.4 of the VEGP SER:

> *The following portion of this technical evaluation section is reproduced from Section 11.3.4 of the BLN SER:*
>
> - *STD SUP 11.3-1*
>
> *The applicant provided supplemental information in BLN COL FSAR Section 11.3.3.6, "Quality Assurance," addressing the quality assurance program to be applied to the gaseous waste system and stated that the program complies with the guidance presented in RG 1.143.*
>
> *The NRC staff reviewed this supplemental quality assurance information included in BLN COL FSAR Section 11.3.3.6 and finds that this supplemental statement commits the applicant to the regulatory positions in RG 1.143 related to quality assurance and is acceptable.*

The following portion of this technical evaluation section is reproduced from Section 11.3.4 of the VEGP SER:

> - *STD SUP 11.3-2*
>
> *The applicant provided additional information in VEGP COL FSAR Section 11.3.3 to address gaseous effluent site interface parameters. The applicant stated that there are no gaseous effluent site interface parameters outside the Westinghouse scope. The staff finds this statement true because all gaseous effluent release points are through the main gas vent and the turbine building exhaust and are part of the certified design.*

<u>Postulated Radioactive Release Due to a Waste Gas Leak or Failure</u>

NUREG-0800, Section 11.3, acceptance criteria and BTP 11-5 require the staff to evaluate the results of a postulated radioactive release resulting from a leakage or failure of a waste gas storage tank or offgas charcoal delay bed. The waste gas system is part of the radioactive GWMS and information on the system is considered as part of the design information required by 10 CFR 50.34a.

The AP1000 DCD and NUREG-1793 addressed the results of this analysis. In response to RAI SRP11.3-CHPB-02 covering AP1000 DCD, Revision 17, Westinghouse detailed the results of this analysis for inclusion in the next revision of the DCD. The staff found this analysis acceptable and that it encompassed the site-specific parameters for the VEGP site. Once the staff confirms the inclusion of the failure analysis in a future revision of the AP1000 DCD and the incorporation by reference of that DCD revision by the VEGP applicant, the staff will consider this item closed for the VEGP COL FSAR. This is considered ***Confirmatory Item 11.3-1****.*

<u>Resolution of Standard Content Confirmatory Item 11.3-1</u>

Confirmatory Item 11.3-1 is a commitment by the applicant to incorporate changes, by reference, proposed by Westinghouse to Section 11.3.3.4 of the AP1000 DCD to include the results of the postulated radioactive release resulting from a leakage or failure of a waste gas storage tank or offgas charcoal delay bed. The staff verified that the applicant has incorporated the AP1000 DCD Revision 18 that includes the above changes. As a result, Confirmatory Item 11.3-1 is now closed.

<u>Demonstrating Compliance with 10 CFR 20.1301(e)</u>

The staff discusses compliance with 10 CFR 20.1301(e) in Section 11.2.4 of this SER.

<u>Demonstrating Compliance with 10 CFR 20.1302</u>

The annual average concentration of radioactive material released in gaseous effluents at the boundary of the unrestricted area must not exceed the values specified in Table 2 of Appendix B to 10 CFR Part 20. The applicant demonstrated compliance with this requirement by referencing the AP1000 DCD. Section 11.3.3.5 of the DCD shows that even at the Technical Specification limit for percent failed fuel defects, the site provides sufficient atmospheric dilution to ensure that the expected effluent release concentrations will be less than those specified in Table 2 of Appendix B to 10 CFR Part 20.

In NUREG-1793, the staff evaluated and accepted the conclusions of Section 11.3.3.5 of the DCD. Based on this acceptance, the staff concludes that the applicant complies with 10 CFR 20.1302.

Demonstrating Compliance with 10 CFR 20.1406

The staff discusses compliance with 10 CFR 20.1406 in Section 11.2.4 of this SER.

In addition, the staff confirmed that the limiting site boundary χ/Q, 1.5E-5 is bounded by the AP1000 DCD limiting χ/Q, 2.0E-5 to support the demonstration of compliance with 10 CFR 20.1302 and 10 CFR Part 20 Appendix B, Table 2, Footnote 4.

Demonstrating Compliance with 10 CFR Part 50, Appendix I

Pursuant to 10 CFR Part 50, Appendix I, Sections II.B, II.C, and II.D, the applicant is responsible for addressing the requirements for dose objectives in controlling doses to a hypothetical maximally exposed member of the public and populations living near the proposed nuclear power plant. The requirements define dose objectives for gaseous effluents, and require a cost-benefit analysis in justifying installed processing and treatment equipment of the GWMS, including any augmentation to the design in complying with 10 CFR Part 50, Appendix I. The staff notes that the applicant has demonstrated compliance with 10 CFR Part 50, Appendix I, Sections II.B, II.C, and II.D requirements by performing the required cost-benefit analysis through WLS COL 11.3-1, and performed the require dose compliance through WLS COL 11.5-3. The staff independently verified the results of the cost-benefit analysis and compliance with the dose objectives and finds that the applicant is in compliance with 10 CFR Part 50, Appendix I, Sections II.B, II.C, and II.D.

Table 11.3-1 Population Doses Breakdown by Source

Source	Total Body (person-rem)	% of Total	Thyroid (person-rem)	% of Total
Noble Gases	1.45E+00	29%	1.45E+00	14.8%
Iodine	1.00E-2	0.02%	4.85E+00	49.5%
Particulates	3.16E-01	6.32%	2.74E-01	2.8%
C-14	2.45E+00	49.0%	2.45E+00	25.0%
H-3	7.7E-01	15.4%	7.70E-01	7.9%
Total	5.00E+00	100%	9.80E+00	100%

Table 11.3-2 Comparison of Maximum Annual Individual Doses

Description	Application	10 CFR Part 50, Appendix I, Sections II.B and II.C
Noble Gases		
• Gamma Dose (mrad)	1.25*	10
• Beta Dose (mrad)	7.32*	20
• Total Body (mrem)	0.732*	5
• Skin (mrem)	4.90*	15
Radioiodines and Particulates		
• Maximum Organ (mrem)	9.21**	15

* Taken from WLS COL FSAR Table 11.3-205
** Dose for the infant thyroid

Table 11.3-3 Comparison of Population Doses (person rem/yr)

Organ/Body	Application*	NRC Staff's Analysis
Total Body	5.00	5.00
Thyroid	9.8	9.8

* Taken from WLS COL FSAR Table 11.3-204

11.3.5 Post Combined License Activities

There are no post COL activities related to this section.

11.3.6 Conclusion

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to the GWMS. There is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff evaluated the additional COL information (STD COL 11.3-1, WLS COL 11.3-1, WLS COL 11.5-3, STD SUP 11.3-1 and STD SUP 11.3-2) in the application against the relevant regulations, acceptance criteria defined in NUREG-0800, Section 11.3, and other NRC regulatory guides. The staff finds that the applicant has satisfactorily addressed all RAIs related to NUREG-0800, Section 11.3 including those related to the relocation of the nuclear island.

In other areas of the evaluation of the GWMS, the staff verified that the applicant had provided sufficient information and that the review and calculations support the conclusion that the GWMS includes the equipment necessary to control releases of radioactive materials in gaseous effluents in accordance with GDC 3, GDC 60, and GDC 61 of 10 CFR Part 50, Appendix A and the requirements of 10 CFR 50.34a. The staff finds that the applicant meets the requirements in GDC 3 by conforming to the guidance in BTP 11-5. The staff finds that the applicant meets the requirements in GDC 60 and GDC 61 by demonstrating compliance with 10 CFR Part 50, Appendix I. The staff also concludes that the design of the GWMS meets the requirements of 10 CFR 20.1301(e), 10 CFR 20.1302, 10 CFR 20.1406, 10 CFR 50.34a, GDC 3, 60 and 61, and 10 CFR Part 50, Appendix I.

11.4 Solid Waste Management (Related to RG 1.206, Section C.III.1, Chapter 11, C.I.11.4, "Solid Waste Management System")

11.4.1 Introduction

The solid waste management system (SWMS) is designed to collect and accumulate spent ion exchange resins and deep-bed filtration media, spent filter cartridges, dry active wastes, and mixed wastes generated from normal plant operation, including anticipated operational occurrences. Processing and packaging of wastes are by mobile systems and the packaged waste is stored in the auxiliary and radwaste buildings until it is shipped offsite to a licensed disposal facility.

11.4.2 Summary of Application

Section 11.4 of the WLS COL FSAR, Revision 11, incorporates by reference Section 11.4 of the AP1000 DCD, Revision 19.

In addition, in WLS COL FSAR Section 11.4, the applicant provided the following:

AP1000 COL Information Items

- STD COL 11.4-1

The applicant added supplemental information in WLS COL FSAR Section 11.4.5 to address how the solid radwaste system complies with the guidance in RG 1.143. STD SUP 11.4-1 also addresses the processes to be followed to ship waste that complies with 10 CFR 61.55, "Waste classification," and 10 CFR 61.56, "Waste characteristics," in WLS COL FSAR Section 11.4.6.1.

License Condition

- Part 10, License Condition 3, Operational Program Implementation

WLS COL FSAR Section 13.4, Table 13.4-201, "Operational Programs Required by NRC Regulations," identifies one entry under Item 9, the Process Control Program (PCP), as a program required to be implemented by a milestone. In accordance with License Condition 3, this program is to be implemented prior to initial fuel load.

- Part 10, License Condition 6, Operational Program Readiness

The applicant proposed a license condition to provide a schedule to support NRC inspection of operational programs including the PCP.

11.4.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the regulatory basis for acceptance of the supplemental information on the SWMS is established in the requirements and guidelines of several codes and standards. These include the following:

- 10 CFR Part 20, "Standards for Protection against Radiation"
- 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities"
- 10 CFR 52.79, "Contents of Applications; Technical Information in Final Safety Analysis Report"
- 10 CFR Part 71, "Packaging and Transportation of Radioactive Material"
- 49 CFR Part 173, "Shippers—General Requirements for Shipments and Packagings"
- State regulations and disposal site waste form requirements for burial at a low level waste disposal site that is licensed in accordance with 10 CFR Part 61 or equivalent State regulations
- RG 1.1.43, Revision 2, Table 1 and Regulatory Positions C.3.2 and C.3.3

The acceptance criteria associated with the SWMS are given in NUREG-0800, Section 11.4, including BTP 11-3.

11.4.4 Technical Evaluation

The staff reviewed Section 11.4 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the SWMS. The results of the staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff's review of this application included the following COL information item and supplemental information:

- STD COL 11.4-1, Solid Waste Management System Process Control Program

- STD SUP 11.4-1, Quality Assurance

In addition to the above items, the staff reviewed the entire section against NUREG-0800, Section 11.4, to determine if the information in WLS COL FSAR Section 11.4 met the regulatory requirements in the regulations stated above (Section 11.4.3 of this SER) and NUREG-0800 acceptance criteria. The relevant NUREG-0800 acceptance criteria are as follows:

- All effluent releases (gaseous and liquid) associated with the operation (normal and anticipated operational occurrences) of the SWMS will comply with 10 CFR Part 20 and RG 1.143, as they relate to the definition of the boundary of the SWMS beginning at the interface from plant systems, including multiunit stations, to the points of controlled liquid and gaseous effluent discharges to the environment or designated onsite storage locations, as defined in the PCP and ODCM.

- Operational Programs. For COL reviews, the description of the operational program and proposed implementation milestone for the PCP aspect of the Process and Effluent Monitoring and Sampling Program are reviewed in accordance with 10 CFR 20.1301, 10 CFR 20.1302, 10 CFR 50.34a, 10 CFR 50.36a, and 10 CFR Part 50, Appendix I, Sections II and IV. Its implementation is required by a license condition detailed in FSAR Section 13.4.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and finds the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) contains evaluation material from the SER for the BLN Units 3 and 4 COL application.

Although the staff concluded that the evaluation performed for the standard content is directly applicable to the WLS COL application, there is a difference in how the WLS applicant addressed STD COL 11.4-1 and how the VEGP applicant addressed this review item. This difference is evaluated by the staff below, following the standard content material for STD COL 11.4-1.

The following portion of this technical evaluation section is reproduced from Section 11.4.4 of the VEGP SER:

> *<u>AP1000 COL Information Items</u>*
>
> *The following portion of this technical evaluation section is reproduced from Section 11.4.4 of the BLN SER:*
>
> - *STD COL 11.4-1*
>
> *The applicant provided additional information in STD COL 11.4-1 to resolve COL Information Item 11.4-1. COL Information Item 11.4-1 states:*
>
>> *The Combined License applicant will develop a process control program in compliance with 10 CFR Sections 61.55 and 61.56 for wet solid wastes and 10 CFR Part 71 and DOT [Department of Transportation] regulations for both wet and dry solid wastes. Process control programs will also be provided by vendors providing mobile or portable processing or storage systems. It will be the plant operator's responsibility to assure that the vendors have appropriate process control programs for the scope of work being contracted at any particular time. The process control program will identify the operating procedures for storing or processing wet solid wastes. The mobile systems process control program will include a discussion of conformance to Regulatory Guide 1.143, Generic Letter GL-80-009, and Generic Letter GL-81-039 and, information of equipment containing wet solid wastes in the non-seismic Radwaste Building. In the event additional onsite storage facilities are a part of Combined License plans, this program will include a discussion of conformance to Generic Letter GL-81-038.*
>
> *The commitment was also captured as COL Action Item 11.4-1 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:*
>
>> *The COL applicant will develop a process control program for both wet and dry solid wastes.*
>
> *In BLN COL FSAR Section 11.4.6, the applicant addressed this COL information item. The applicant adopted NEI [Nuclear Energy Institute] 07-10*[A]*, "FSAR Template Guidance for Process Control Program (PCP) Description." The PCP describes the administrative and operational controls used for the solidification of liquid or wet solid waste and the dewatering of wet solid waste. It provides the necessary controls such that the final disposal waste product meets applicable federal regulations (10 CFR Parts 20, 50, 61, 71 and 49 CFR Part 173), state regulations, and disposal site waste form requirements for burial at a low level waste disposal site licensed in accordance with 10 CFR Part 61. Waste processing equipment and services may be provided by the plant or by third-party vendors. In a letter dated January 8, 2009, (ML082910077), the NRC accepted NEI 07-10*[A]*, Revision 3. Specifically, the NRC staff indicated that for COL applications NEI 07-10*[A]*, Revision 3, provides an acceptable template for*

assuring that the administrative and operational controls for waste processing, processing parameters, and surveillance requirements within the scope of the PCP will meet the requirements of 10 CFR 52.79. In a letter dated April 23, 2009 (ML091170073), the applicant proposed to revise BLN FSAR Section 11.4 to incorporate the approved NEI 07-10[A] *Revision 3. Since the BLN COL FSAR Section 11.4 has not adopted the approved version of the NEI Template, this is **Confirmatory Item 11.4-1.** Each process used meets the applicable requirements of the PCP. BLN COL FSAR Table 13.4-201 provides milestones for PCP implementation and is acceptable.*

*In STD COL 11.4-1, the applicant states that "no additional onsite radwaste storage is required beyond that described in the DCD." The applicant should explain why this statement is included or should remove it. In section 11.4 of NUREG-1793, the staff stated that if a need for onsite storage of low-level waste has been identified beyond that provided in AP1000 Standard Design because of unavailability of offsite storage, the applicant should submit the details of any proposed onsite storage facility to the NRC. The applicant needs to provide any arrangements for offsite storage for low-level waste or to submit plans for onsite storage. This is identified as **Open Item 11.4-1**.*

The following portion of this technical evaluation section is reproduced from Section 11.4.4 of the VEGP SER:

Resolution of Standard Content Confirmatory Item 11.4-1

To address Confirmatory Item 11.4-1 in the BLN SER with open items, the applicant updated VEGP FSAR Section 11.4.6 to indicate adoption of the NRC-approved version of NEI 07-10A. VEGP adoption of this template effectively resolves Confirmatory Item 11.4-1.

Resolution of Standard Content Open Item 11.4-1

To address Open Item 11.4-1 in the BLN SER with open items, the applicant updated VEGP FSAR Section 11.4 with information supporting the statement that no additional onsite radwaste storage was required beyond that described in the DCD. This additional information is in VEGP COL 11.4-1 and VEGP SUP 11.4-1 and is evaluated below.

Evaluation of Site-specific Information for STD COL 11.4-1

Regarding the Resolution of Standard Content Open Item 11.4-1, the staff does not consider the open item relevant to the WLS COL application because the applicant has available offsite disposal of all types of low-level radioactive waste through its membership in the Atlantic Compact. Therefore, an update of the WLS COL FSAR is not necessary to resolve this item.

The following portion of this technical evaluation section is reproduced from Section 11.4.4 of the VEGP SER:

<u>Supplemental Information</u>

The following portion of this technical evaluation section is reproduced from Section 11.4.4 of the BLN SER:

- *STD SUP 11.4-1*

The applicant provided supplemental information in Section 11.4.5 of the BLN COL FSAR to describe the QA program applicable to design, construction, installation and testing provisions of the solid radwaste system. This QA program is established by procedures and complies with the guidance presented in RG 1.143.

In BLN FSAR Section 11.4.6, the applicant also added a description of procedures relating to waste shipments, waste stream processing, verifying waste as non-radioactive, periodic system maintenance, personnel training, and document revision, clearing with third party vendors. The staff reviewed the descriptions and found them to be comprehensive and acceptable.

The NRC staff reviewed the supplemental information provided in STD SUP 11.4-1 related to the QA program for the solid radwaste system included under Section 11.4.4 of the BLN COL FSAR and finds that this supplemental statement commits the applicant to the regulatory positions in RG 1.143 related to quality assurance.

<u>License Conditions</u>

- *Part 10, License Condition 3, Operational Program Implementation*

VEGP COL FSAR Section 11.4.6 describes the process control program. VEGP COL FSAR Table 13.4-201 provides the milestone (prior to initial fuel load) for implementation of the process control program and is acceptable as described in the staff's SER related to NEI 07-10.

- *Part 10, License Condition 6, Operational Program Readiness*

The applicant proposed a license condition to provide a schedule to support NRC inspection of operational programs including the process control program. The proposed license condition is consistent with the policy established in SECY-05-0197 and is acceptable.

<u>Compliance with 10 CFR Part 50 Appendix I Design Criteria</u>

The design of the SWMS described in the AP1000 DCD has no release points directly to the environment. Compliance with Appendix I ALARA criteria is strictly based on the releases from the LWMS and GWMS and not the SWMS.

11.4.5 Post Combined License Activities

For the reasons discussed in the technical evaluation section above, the staff finds the following two license conditions acceptable:

- License Condition (11-2) – Before initial fuel load, the licensee shall implement an operational program for process and effluent monitoring and sampling. The program shall include the subprogram and documents for a Process Control Program.

- License Condition (11-3) – No later than 12 months after issuance of the COL, the licensee shall submit to the Director of the Office of New Reactors (NRO) a schedule that supports planning for and conduct of NRC inspections of the operational program for process and effluent monitoring and sampling (including process control program). The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until the operational program for process and effluent monitoring and sampling (including process control program) has been fully implemented.

11.4.6 Conclusion

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to the SWMS and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff evaluated the additional COL information (STD COL 11.4-1 and STD SUP 11.4-1) in the application against the relevant NRC regulations, acceptance criteria in NUREG-0800, Section 11.4, and other NRC regulatory guides.

Based on the evaluation above, the staff finds that the applicant's means for handling radioactive solid waste during normal operations, including AOOs are consistent with GDC 60. In accordance with 10 CFR 52.79(a)(3), the staff also finds that the applicant has provided sufficient information regarding the kinds and quantities of radioactive materials expected to be produced in the operation of the facility and the means for controlling and limiting radioactive effluents and exposures within the limits set forth in 10 CFR Part 20. The staff verified that the applicant has provided sufficient information and that the review supports the conclusion that the design and operation of the SWMS is acceptable and meets the requirements of GDC 61 of 10 CFR Part 50, Appendix A; 10 CFR 50.34a, 10 CFR 20.1301(e), 10 CFR 20.1406, and 10 CFR Part 50, Appendix I, and 10 CFR Parts 61 and 71.

11.5 Radiation Monitoring (Related to RG 1.206, Section C.III.1, Chapter 11, C.I.11.5, "Process and Effluent Radiological Monitoring and Sampling Systems")

11.5.1 Introduction

The radiation monitoring systems are used to monitor liquid and gaseous process streams and effluents from the LWMS, GWMS, and SWMS. The radiation monitoring systems include subsystems used to collect process and effluent samples during normal operation and AOO's, and under post-accident conditions.

11.5.2 Summary of Application

Section 11.5 of the WLS COL FSAR, Revision 11, incorporates by reference Section 11.5 of the AP1000 DCD, Revision 19.

In addition, in WLS COL FSAR Section 11.5, the applicant provided the following:

Departure

- WLS DEP 6.4-1

The applicant provided additional information in Section 11.5 of the WLS COL FSAR about WLS DEP 6.4-1 related to design changes affecting habitability of the main control room and changes to the calculated doses to control room operators. This information, as well as related WLS DEP 6.4-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.2 of this SER.

AP1000 COL Information Items

- STD COL 11.5-1

The applicant provided additional information in STD COL 11.5-1 to resolve COL Information Item 11.5-1 (COL Action Item 11.5-1). The information addresses the ODCM.

- STD COL 11.5-2

The applicant provided additional information in STD COL 11.5-2 to resolve COL Information Item 11.5-2 (COL Action Item 11.5-2). The information provides programmatic aspects of the effluent monitoring and sampling program.

- WLS COL 11.5-2

The applicant provided additional information in WLS COL 11.5-2 to add language to WLS COL FSAR Section 11.5.3 addressing extension of the existing Duke Energy program for QA of radioactive effluent and environmental monitoring to apply to WLS Units 1 and 2.

- WLS COL 11.5-3

The applicant provided additional information in WLS COL 11.5-3 to resolve COL Information Item 11.5-3 (COL Action Item 11.5-3). The information relates to the 10 CFR Part 50, Appendix I guidelines.

License Conditions

- Part 10, License Condition 3, Operational Program Implementation, Item G.3

WLS COL FSAR Section 13.4, Table 13.4-201, “Operational Programs Required by NRC Regulations,” identifies three entries under Item 9, “Process and Effluent Monitoring and Sampling Program,” as follows: (1) Radiological Effluent Technical Specifications/Standard Radiological Effluent Controls; (2) ODCM; and (3) Radiological Environmental Monitoring program, as programs identified in FSAR Section 11.5 required to be implemented by a milestone. In accordance with License Condition 3, Item G.3, these programs are to be implemented prior to initial fuel load.

- Part 10, License Condition 6, Operational Program Readiness

The applicant proposed a license condition to provide a schedule to support NRC inspection of operational programs including the Radiological Effluent Technical Specifications/Standard Radiological Effluent Controls; the ODCM; and the Radiological Environmental Monitoring program.

11.5.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the regulatory basis for acceptance of the supplementary information on radiation monitoring addressed in COL Information Items 11.5-1, 11.5-2, and 11.5-3 is established in the requirements and guidelines of the following:

- 10 CFR Part 50, Appendix A, GDC 64, “Monitoring Radioactivity Releases”
- 10 CFR Part 20, “Standards for Protection against Radiation Material”
- 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities
- 10 CFR Part 52,, “Licenses, Certifications, and Approvals for Nuclear Power Plants”
- 10 CFR Part 61, “Licensing Requirements for Land Disposal of Radioactive Waste”
- 10 CFR Part 71, “Packaging and Transportation of Radioactive Material”
- American National Standards Institute/Health Physics Society (ANSI/HPS) N13.1, “Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities”
- ANSI N42.18, “Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents”
- RG 1.21, “Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste,” Revision 2
- RG 4.15, “Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination) – Effluent Streams and the Environment,” Revision 2

The applicable acceptance criteria associated with the radiation monitoring system are given in NUREG-0800, Section 11.5.

11.5.4 Technical Evaluation

The staff reviewed Section 11.5 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic[1]. The staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the radiation monitoring system. The results of the staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the following information in the WLS COL FSAR:

AP1000 COL Information Items

- STD COL 11.5-1, "Plant Offsite Dose Calculation Manual (ODCM)"
- STD COL 11.5-2, Programmatic Aspects of the Effluent Monitoring and Sampling Program
- WLS COL 11.5-2 adds language to WLS COL FSAR Section 11.5.3 addressing extension of the existing Duke Energy program for QA of radioactive effluent and environmental monitoring to apply to WLS Units 1 and 2.
- WLS COL 11.5-3, 10 CFR Part 50, Appendix I Guidelines

In addition to the above items, the staff reviewed the entire section against NUREG-0800, Section 11.5, to determine if the information in WLS COL FSAR Section 11.5 met the regulatory requirements in the regulations stated above (SER Section 11.5.3) and NUREG-0800 acceptance criteria. The relevant NUREG-0800 acceptance criteria are as follows:

- Provisions should be made to ensure representative sampling from radioactive process streams and tank contents. Recirculation pumps for liquid waste tanks (collection or sample test tanks) should be capable of recirculating at a rate of not less than two tank volumes in 8 hours. For gaseous and liquid process stream samples, provisions should be made for purging sampling lines and for reducing the plate-out of radioactive materials in sample lines. Provisions for gaseous sampling from ducts and stacks should be consistent with ANSI/HPS N13.1-1999.
- For COL reviews, the description of the operational program and proposed implementation milestone for the radiological effluent technical specification/standard radiological effluent control, ODCM and Radiological Environmental Monitoring Program aspects of the Process and Effluent Monitoring and Sampling Program are reviewed in accordance with 10 CFR 20.1301, 10 CFR 20.1302, 10 CFR 50.34a, 10 CFR 50.36a, and 10 CFR Part 50, Appendix I, Sections II and IV. Its implementation is required by a license condition.

Section 1.2.3 of this SER provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP

Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff completed its review and finds the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) contains evaluation material from the SER for the BLN Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from Section 11.5.4 of the VEGP SER:

> *<u>AP1000 COL Information Items</u>*
>
> *The following portion of this technical evaluation section is reproduced from Section 11.5.4 of the BLN SER:*
>
> - *STD COL 11.5-1*
>
> *The applicant provided additional information in STD COL 11.5-1 to resolve COL Information Item 11.5-1. COL Information Item 11.5-1 states:*
>
> > *The Combined License applicant will develop an offsite dose calculation manual that contains the methodology and parameters used for calculation of offsite doses resulting from gaseous and liquid effluents. The Combined License applicant will address operational setpoints for the radiation monitors and address programs for monitoring and controlling the release of radioactive material to the environment, which eliminates the potential for unmonitored and uncontrolled release. The offsite dose calculation manual will include planned discharge flow rates.*
>
> *This commitment was also captured as COL Action Item 11.5-1 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:*
>
> > *The COL applicant will develop an offsite dose calculation manual that contains the methodology and parameters used to calculate offsite doses resulting from gaseous and liquid effluents.*
>
> *In BLN COL FSAR Section 11.5.7, the applicant adopts NEI 07-09*[A]*, "FSAR Template Guidance for Offsite Dose Calculation Manual (ODCM) Program*

Description." The ODCM program description contains: (1) the methodology and parameters used for calculating doses resulting from liquid and gaseous effluents; (2) operational setpoints, including planned discharge rates, for radiation monitors and monitoring programs; and (3) the limitations on operation of the radwaste systems, including functional capability of monitoring instruments, concentrations of effluents, sampling, analysis, 10 CFR Part 50, Appendix I dose and dose commitments and reporting. In a letter dated January 27, 2009 (ML083530745), the NRC accepted NEI 07-09, Revision 4. Specifically, the NRC indicated that for COL applications, NEI 07-09[A]*, Revision 4 provides an acceptable template assuring that the ODCM program meets applicable NRC regulations and guidance. In a letter dated April 23, 2009 (ML091170073), the applicant proposed to revise BLN COL FSAR Section 11.5 to incorporate the approved NEI 07-09*[A]*, Revision 4. Since the BLN COL FSAR Section 11.5 has not adopted the approved version of the NEI Template, this is* ***Confirmatory Item 11.5-1****. BLN COL FSAR Table 13.4-201 provides milestones for ODCM implementation. This section also addresses Plant Interface Item 11.4, "requirements for offsite sampling and monitoring of effluent concentrations." The staff finds the applicant's consideration of Plant Interface Item 11.4 to be acceptable based on a review of the ODCM program (NEI 07-09*[A]*). The NRC staff reviewed the resolution of STD COL 11.5-1 related to the ODCM included under Section 11.5.7 of the BLN COL FSAR and considers it adequately addressed in NEI 07-09*[A].

The following portion of this technical evaluation section is reproduced from Section 11.5.4 of the VEGP SER:

<u>Resolution of Standard Content Confirmatory Item 11.5-1</u>

To address Confirmatory Item 11.5-1, the applicant updated the VEGP FSAR Section 11.5.7 to indicate adoption of the NRC-approved version of NEI 07-09A. VEGP adoption of this template effectively resolves Confirmatory Item 11.5-1.

The following portion of this technical evaluation section is reproduced from Section 11.5.4 of the BLN SER:

- *STD COL 11.5-2*

The applicant provided additional information in STD COL 11.5-2 to resolve COL Information Item 11.5-2 (COL Action Item 11.5-2). COL Information Item 11.5-2 states:

> *The Combined License applicant is responsible for the site-specific and program aspects of the process and effluent monitoring and sampling in accordance with ANSI N13.1 and RGs 1.21 and 4.15.*

The commitment was also captured as COL Action Item 11.5-2 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:

> *The COL applicant is responsible for ensuring that the process and effluent monitoring and sampling program at its site conforms to the guidelines of ANSI N13.1-1969, RG 1.21, and RG 4.15.*

In BLN COL FSAR Sections 11.5.1.2, 11.5.2.4, 11.5.4, 11.5.4.1, 11.5.4.2 and 11.5.6.5, the applicant described the programmatic aspects of the effluent monitoring and sampling program. In addition, the applicant provided in BLN COL 11.5-2 specific language regarding the applicant's extension of the existing TVA program for quality assurance of radiological effluent and environmental monitoring which is based on RG 4.15, Revision 1, instead of the most current Revision 2. To maintain consistency, the applicant proposes to apply the same program to BLN Units 3 and 4.

The NRC staff reviewed the resolution of BLN COL 11.5-2 related to the effluent monitoring and sampling program included under Sections 11.5.1.2, 11.5.2.4, 11.5.3, 11.5.4, 11.5.4.1, 11.5.4.2 and 11.5.6.5 of the BLN COL FSAR and considers it adequately addressed in NEI 07-09[A]*.*

- WLS COL 11.5-2

In WLS COL 11.5-2, the applicant extended the existing Duke Energy QA program, including RG 4.15, Revision 1R1, for effluent and environmental monitoring to Units 1 and 2. By using the current program, which is based on RG 4.15, Revision 1 instead of Revision 2, the applicant will also avoid confusion and the potential for error because the program for the existing and planned units will share the same equipment and personnel. Therefore, the staff finds the use of RG 4.15, Revision 1 acceptable and considers COL Information Item 11.5-2 resolved.

- WLS COL 11.5-3

The applicant provided additional information in WLS COL 11.5-3 to resolve COL Information Item 11.5-3, which states:

> The Combined License applicant is responsible for addressing the 10 CFR 50, Appendix I guidelines for maximally exposed offsite individual doses and population doses via liquid and gaseous effluents.

The commitment was also captured as COL Action Item 11.5-3 in Appendix F of NUREG-1793, which states:

> The COL applicant is responsible for addressing the guidelines of Appendix I to 10 CFR Part 50, as they relate to maximally exposed offsite individual doses and population doses attributable to liquid and gaseous effluents.

The applicant addressed this COL item by adding information to WLS COL FSAR Sections 11.2.3.5 and 11.3.3.4 for liquid and gaseous effluents, respectively.

The staff reviewed the resolution of WLS COL 11.5-3 related to compliance with 10 CFR Part 50, Appendix I, as discussed in Sections 11.2.4 and 11.3.4 of this report, and considers it adequately addressed.

The following portion of this technical evaluation section is reproduced from Section 11.5.4 of the VEGP SER:

> *The following portion of this technical evaluation section is reproduced from Section 11.5.4 of the BLN SER:*
>
> *<u>Section 11.5.4.2, Representative Sampling</u>*
>
> *In this section, the applicant describes how it will take representative samples for analysis. Based on the staff's review, the staff issued RAIs 11.5-1 and 11.5-2. RAI 11.5-1 requested clarification about the use of ANSI/HPS N13.1-1999. RAI 11.5-2 requested more information concerning how the applicant ensures representative liquid effluent and environmental sampling.*
>
> *In response to RAI 11.5-1, the applicant revised its commitment to use the 1999 standard. Because the applicant made no changes to the certified design, it removed the commitment to use ANSI/HPS N13.1-1999, and committed to ANSI N13.1-1969 to be consistent with the AP1000 certified design. ANSI withdrew the 1969 standard and replaced it with ANSI/HPS N13.1-1999 because the approach taken in the 1969 standard did not provide assurance that the sample in the effluent vent would be representative. The 1999 standard differs significantly from the earlier version in that it is now performance based. NUREG-0800 Section 11.5 (2007) uses the 1999 standard as acceptance criteria. The staff is pursuing this issue through the DC because it deals with the design of the sampling systems for radioactive gas streams.* [While AP1000 DCD FSAR Rev. 19, Tier 2, Section 11.5 (p.11.5-1 and 11.5-18) still refers to ANSI N13.1-1969, the DCD has incorporated some of the provisions of the ANSI/HPS N13.1-1999 standard. Specifically, AP1000 DCD FSAR Tier 2, Section 11.5.2.3.3 (p.11.5-10 and 11.5-11) summarize key aspects of the ANSI/HPS N13.1-1999 standard. The staff found this approach acceptable.]
>
> *The applicant provided a response to RAI 11.5-2 and the staff finds the response acceptable. The response provided a more detailed description of how the applicant will assure that liquid samples will be representative. The applicant committed to follow the recommendations in ANSI N42.18 and RG 1.21. In addition, the applicant provided more operational descriptions for composite sampling. The NRC staff verified that Revision 1 of the BLN COL FSAR adequately addressed the above. As a result, RAI 11.5-2 is closed.*

The following portion of this technical evaluation section is reproduced from Section 11.5.4 of the VEGP SER:

> *<u>License Condition</u>*
>
> - *Part 10, License Condition 3, Operational Program Implementation, Item G.3*
>
> *VEGP COL FSAR Section 11.5.3 describes effluent monitoring and sampling and Section 11.5.7 describes the offsite dose calculation manual. License Condition 3, Item G.3 requires the licensee to implement the "Process and Effluent Monitoring and Sampling" program prior to initial fuel load. VEGP COL*

FSAR Section 13.4, Table 13.4-201, "Operational Programs Required by NRC Regulations," identifies three entries under Item 9, "Process and Effluent Monitoring and Sampling Program," as follows: (1) Radiological Effluent Technical Specifications/Standard Radiological Effluent Controls, (2) Offsite Dose Calculation Manual; and (3) Radiological Environmental Monitoring program, as programs identified in FSAR Section 11.5 required to be implemented by a milestone. The ODCM includes the Radiological Effluent Technical Specifications/Standard Radiological Effluent Controls and the Radiological Environmental Monitoring program. In accordance with License Condition 3, Item G.3, these programs are to be implemented prior to initial fuel load. VEGP COL FSAR Table 13.4-201 provides the milestones (prior to initial fuel load) for implementation of these elements of the Process and Effluent Monitoring and Sampling Program and is acceptable as described in the staff's SER related to NEI 07-09.

- *Part 10, License Condition 6, Operational Program Readiness*

The applicant proposed a license condition to provide a schedule to support NRC inspection of operational programs including the ODCM, effluent technical specifications, and the radiological environmental monitoring program. The proposed license condition is consistent with the policy established in SECY-05-0197 and is acceptable.

11.5.5 Post Combined License Activities

For the reasons discussed in the technical evaluation section above, the staff finds the following two license conditions acceptable:

- License Condition (11-4) – Before initial fuel load, the licensee shall implement an operational program for process and effluent monitoring and sampling. The program shall include the following subprograms and documents:

    a. Radiological Effluent Technical Specifications/Standard Radiological Effluent Controls

    b. Offsite Dose Calculation Manual

    c. Radiological Environmental Monitoring Program

- License Condition (11-5) – No later than 12 months after issuance of the COL, the licensee shall submit to the Director of NRO a schedule that supports planning for and conduct of NRC inspections of the operational program for process and effluent monitoring and sampling (including Radiological Effluent Technical Specifications/Standard Radiological Effluent Controls, Offsite Dose Calculation Manual, and Radiological Environmental Monitoring Program). The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until the above operational program has been fully implemented.

11.5.6 Conclusion

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to the radiation monitoring system, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff evaluated the additional COL information (STD COL 11.5-1, STD COL 11.5-2, WLS COL 11.5-2, and WLS COL 11.5-3) in the application against the relevant NRC regulations, acceptance criteria defined in NUREG-0800, Section 11.5, and other NRC regulatory guides. The staff concludes that the applicant has satisfactorily addressed all RAIs related to Section 11.5.

WLS DEP 6.4-1, related to design changes affecting habitability of the main control room and changes to the calculated doses to control room operators, is reviewed and found acceptable by the staff in Section 21.2 of this SER.

The staff verified that the applicant has provided sufficient information and that the review supports the conclusion that the process and effluent radiological monitoring and sampling systems are sufficient to comply with applicable portions of GDC 64 of 10 CFR Part 50, Appendix A; applicable requirements of 10 CFR Parts 20, 50 and 52; ANSI/HPS N13.1-1999, ANSI N42.18, RGs 1.21 and 4.15, and applicable acceptance criteria in NUREG-0800, Section 11.5.

12 RADIATION PROTECTION

This chapter provides information on radiation protection methods and estimated occupational radiation exposures (OREs) of operating and construction personnel during normal operations (including refueling; purging; fuel handling and storage; radioactive material handling, processing, use, storage, and disposal; maintenance; routine operational surveillance; in-service inspection (ISI); and calibration), and anticipated operational occurrences (AOOs). Specifically, this chapter provides information on facility and equipment design, planning and procedures programs, and techniques and practices employed by the applicant to meet the radiation protection standards set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, “Standards for protection against radiation,” and to be consistent with the guidance given in the appropriate regulatory guides (RGs), where the practices set forth in such guides are used to implement the U.S. Nuclear Regulatory Commission (NRC) regulations.

12.1 Assuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable

12.1.1 Introduction

Section 12.1 addresses policy and design considerations to ensure that the ORE to personnel will be kept as low as is reasonably achievable (ALARA). The ALARA program is addressed in this section and in Appendix 12AA of the William States Lee III Nuclear Station (WLS) combined license (COL) Final Safety Analysis Report (FSAR).

12.1.2 Summary of Application

WLS COL FSAR, Revision 11, Section 12.1 incorporates by reference AP1000 Design Control Document (DCD), Revision 19, Section 12.1.

In addition, in WLS COL FSAR Section 12.1, the applicant provided the following:

AP1000 COL Information Items

- STD COL 12.1-1

The applicant provided additional information in Standard (STD) COL 12.1-1 to resolve COL Information Item 12.1-1 (COL Action Item 12.2.1-1) that addresses ALARA and operational policies and conforms to regulatory guides. The applicant incorporated information complying with Nuclear Energy Institute (NEI) 07-08A, “Generic FSAR Template Guidance for Ensuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA),” into WLS COL FSAR Section 12.1 and NEI 07-03A, “Generic FSAR Template Guidance for Radiation Protection Program Description,” in Appendix 12AA. The applicant also provided the site specific information in their FSAR, in addition to that required by NEI 07-08A, Section 12.1.2, specifying that the applicant’s quality assurance criteria are described in Part III of the Quality Assurance Program Description discussed in WLS COL FSAR Section 17.

Supplemental Information

- STD Supplemental (SUP) 12.1-1

The applicant provided information addressing equipment layout to be added at the end of AP1000 DCD Section 12.1.2.4.

12.1.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," and its supplements. The regulatory requirements and acceptance criteria associated with the relevant requirements of NRC regulations for the ALARA program are given in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," Section 12.1.

The applicable regulatory requirements and guidance for STD COL 12.1-1 and STD SUP 12.1-1 are as follows:

- 10 CFR Part 20, "Standards for Protection against Radiation"
- 10 CFR 20.1101, "Radiation Protection Programs"
- 10 CFR 19.12, "Instructions to Workers"
- RG 1.8, "Qualification and Training of Personnel for Nuclear Power Plants," Revision 3
- RG 1.33, "Quality Assurance Program Requirements (Operation)," Revision 2
- RG 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Revision 4
- RG 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as is Reasonably Achievable," Revision 3
- RG 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as is Reasonably Achievable," Revision 1-R
- NUREG-1736, "Consolidated Guidance: 10 CFR Part 20 – Standards for Protection Against Radiation"

12.1.4 Technical Evaluation

The NRC staff (the staff) reviewed WLS COL FSAR Section 12.1 and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the WLS COL application and incorporated by reference addresses the required information relating to ensuring that the ORE to personnel will be kept ALARA. The results of the staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the design certification (DC) and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application, Vogtle Electric Generating Plant (VEGP) Units 3 and 4, were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from requests for additional information (RAIs).

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed and that those changes were actualized in the WLS COL FSAR.

- The staff verified that the site-specific differences, if any, did not adversely affect any previous relevant evaluation or conclusion.

The staff completed its review and concluded that the evaluation performed for the standard content is directly applicable to the WLS COL application. This standard content material is identified in this report by use of italicized, double-indented formatting. Section 1.2.3 of this report provides an explanation of why the standard content material from the Vogtle (VEGP) reference COL application SER contains evaluation material from the Bellefonte Nuclear Plant (BLN), Units 3 and 4 COL application SER.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 12.1.4:

> *The following portion of this technical evaluation section is reproduced from Section 12.1.4 of the BLN SER.*

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[1] See Section 1.2.2 of this report for a discussion of the staff's review related to verification of the scope of information to be included in a COL application that references a design certification (DC).

AP1000 COL Information Item

- *STD COL 12.1-1*

The applicant provided additional information in STD COL 12.1-1, related to ALARA and Operational Policies, to resolve COL Information Item 12.1-1. COL Information Item 12.1-1 states:

> *Operational considerations of ALARA, as well as operational policies and continued compliance with 10 CFR 20 and RGs 1.8, 8.8, and 8.10, will be addressed by the Combined Operating License applicant. In addition, the Combined Operating License applicant will address operational considerations of the Standard Review Plan to the level of detail provided in RG 1.70. RGs that will be addressed include: 8.2, 8.7, 8.9, 8.13, 8.15, 8.20, 8.25, 8.26, 8.27, 8.28, 8.29, 8.34, 8.35, 8.36, and 8.38.*

The commitment was also captured as COL Action Item 12.2.1-1 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:

> *The COL applicant will review all plant procedures and modification plans that involve personnel radiation exposure to ensure that the ALARA policy is applied. In addition, a COL applicant referencing the AP1000 certified design will address operational ALARA concerns and will submit an operational ALARA policy which conforms to the requirements of 10 CFR Part 20 and the recommendations of Revision 2 to RG 1.8, RG 8.8, and Revision 1-R to RG 8.10.*

In response to COL Action Item 12.2.1-1 in the BLN COL FSAR (Revision 1) as STD COL 12.1-1:

> *This section incorporates by reference [Nuclear Energy Institute] NEI 07-08, "Generic FSAR Template Guidance for Ensuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA)," Revision 2, which is currently under review by the NRC staff. See Table 1.6-201. ALARA practices are developed in a phased milestone approach as part of the procedures necessary to support the Radiation Protection Program. Table 13.4-201 describes the major milestones for ALARA procedures development and implementation.*

STD COL 12.1-1 includes a commitment to the use of a "Generic FSAR Template Guidance for Ensuring That Occupational Radiation Exposures Are as Low as Is Reasonably Achievable (ALARA)," as an operational program document, based on draft NEI Template 07-08, Revision 2. The NEI template presents the functional elements of an ALARA program, which, if met, would demonstrate compliance with 10 CFR 20.1101 and 10 CFR 19.12. Accordingly, BLN FSAR Section 12.1, STD COL 12.1-1 needs to be updated as to its commitment to the final NEI ALARA template if it is accepted by the NRC staff.

Therefore, the staff cannot find the applicant's reference to the NEI 07-08 template to be acceptable until the staff completes its review of this template as a method to meet the regulatory requirements of an ALARA program, and the BLN FSAR is updated to reference the final version of this template. This is identified as Open Item 12.1-1.

The NRC staff review finds that BLN FSAR Section 12.1 and Appendix 12AA describe programs and procedures that ensure ORE will be ALARA in accordance with the training requirements in 10 CFR 19.12 and the ALARA provisions of 10 CFR 20.1101(b). The ALARA policy will be described, displayed, and implemented in accordance with the provisions of RG 8.8 (Regulatory Position C.1) and RG 8.10 (Regulatory Position C.1) and NUREG-1736, as it relates to maintaining doses ALARA.

According to BLN FSAR Appendix 12AA, NEI 07-03, NEI 07-08, and Chapter 13, "Conduct of Operations," specific individual(s) will be designated and assigned responsibility and authority for implementing ALARA policy at the BLN site. The Functional Manager in charge of Radiation Protection and the Radiation Protection staff periodically will review, update, and modify as appropriate, plant design features and changes, as well as all operating and maintenance features, using exposure data and experience gained from operating nuclear power plants to ensure that occupational exposures will be kept ALARA in accordance with RG 8.8 guidance.

Using the guidance of Section 12.1 of NUREG-0800, the staff finds BLN FSAR Section 12.1 and Appendix 12AA are in accordance with the ALARA provisions of 10 CFR 20.1101(b) and RG 8.8 (Regulatory Position C.2) and will include incorporation of measures for reducing the need for time spent in radiological areas; measures to control access to radiological areas; measures to reduce the production, distribution, and retention of activated corrosion products throughout the primary system; measures for assuring that ORE during decommissioning will be ALARA; reviews of design modifications by competent radiation protection personnel; instructions to engineers regarding ALARA design; experience from operating plants and past designs; and continuing facility design reviews.

Using the guidance of Section 12.1 of NUREG-0800, the staff finds that BLN COL FSAR Section 12.1 and Appendix 12AA describe an acceptable program to develop plans and procedures in accordance with RGs 1.33, 1.8, 8.8, and 8.10 that can incorporate the experiences obtained from facility operation into facility and equipment design and operations planning and that will implement specific exposure control techniques.

Initially, it was not clear to the NRC staff when the appropriate ALARA program and planning procedures would be implemented as described in the proposed License Conditions (Part 10 of the BLN, Units 3 and 4 COL application). Therefore, the staff issued request for additional information (RAI) 12.1-1. In a letter dated September 22, 2008, the applicant stated that ALARA focused procedures are developed in conjunction with the Radiation Protection Program (RPP) and thus will follow the RPP milestones for implementation found in FSAR

Table 13.4-201. The applicant stated that FSAR Section 12.1, STD COL 12.1-1 text will be updated as to its commitment to the final ALARA program implementation. The NRC staff finds the RAI response acceptable because it clearly identified that ALARA practices will be in place at the same time as the RPP. The NRC staff verified that Revision 1 of the BLN COL FSAR adequately incorporates the above. As a result, RAI 12.1-1 is closed. For a discussion related to the proposed license condition related to the RPP, which includes ALARA practices, refer to SER Section 12.5.5.

In accordance with 10 CFR 20.1101(b), the staff finds that overall facility operations, as well as the RPP as described in BLN COL FSAR Section 12.5, Appendix 12AA, and NEI 07-03 will integrate the procedures necessary to ensure that radiation doses are ALARA, including work scheduling, work planning, design modifications, and radiological considerations. Operating and maintenance personnel will follow specific plans and procedures to ensure that goals related to keeping exposures ALARA are achieved in the operation of the plant. Engineering controls for the protection of personnel will be optimized. Operations involving high person-sievert (person-rem) exposures will be carefully preplanned and carried out by personnel who are well trained in radiation protection and using proper equipment. During maintenance activities, in radiological areas, personnel will be monitored for exposure to radiation and contamination. Their radiation exposures will be reviewed and used to make changes in future job procedures and techniques.

The BLN FSAR states that COL information item, STD COL 12.1-1 is addressed in NEI 07-08, and Appendix 12AA of the BLN COL FSAR, which references NEI 07-03. The staff has reviewed the current version of NEI 07-03 and NEI 07-08 with respect to compliance with RG 1.8. The NEI 07-03 template states that the Radiation Protection Manager, Radiation Protection Technicians, and Radiation Protection Supervisory and Technical Staff will be trained and qualified in accordance with the guidance of RG 1.8. In a letter dated March 18, 2009 (ML090510379), the NRC accepted NEI 07-03, Revision 7. Specifically, the NRC staff indicated that for COL applications, NEI 07-03, Revision 7 provides an acceptable template for assuring that the RPP meets the applicable NRC regulations and guidance. Since the BLN COL FSAR has not yet adopted the approved version of the NEI template, this is identified as Confirmatory Item 12.1-1. At present, the NRC has not accepted NEI-07-08 as an acceptable template to be used by the COL applicants. As a result, this is identified as ***Open Item 12.1-1.***

Supplemental Information

- *STD SUP 12.1-1*

The applicant added the following text to the end of Section 12.1.2.3, "Facility Layout General Design Considerations for ALARA," of the DCD included in the DC amendment:

A video record of the equipment layout in areas where radiation fields are expected to be high following operations may be used to assist in ALARA planning and to facilitate decommissioning.

The NRC staff acknowledges STD SUP 12.1-1 as a statement of fact not requiring NRC review.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 12.1.4:

Resolution of Standard Content Open Item 12.1-1 and Confirmatory Item 12.1-1

The NRC staff compared the VEGP and BLN COL applications and found them to be essentially identical, with two exceptions: first, the application material under STD COL 12.1-1 in Section 12.1 of the VEGP application references NEI 07-08A and the application material under STD COL 12.1-1 in Section 12.1 of the BLN application references NEI 07-08, Revision 2; and second, the VEGP FSAR Appendix 12AA references NEI 07-03A and the BLN FSAR Appendix 12AA references Revision 3 of NEI 07-03. Regarding these exceptions, the differing material associated with STD COL 12.1-1 in the VEGP FSAR is associated with adopting NEI 07-08A and NEI 07-03A, which are evaluated below as part of resolving Open Item 12.1-1 and Confirmatory Item 12.1-1.

In a letter from NEI to NRC dated October 29, 2009, NEI submitted NEI 07-08A to the NRC, which is the version of NEI 07-08 that has been accepted by the NRC. Accordingly, Open Item 12.1-1 is resolved for VEGP.

Confirmatory Item 12.1-1 is resolved for VEGP because the applicant has adopted the approved version of NEI 07-03, i.e., NEI 07-03A, (see paragraph below).

In Revision 2 of the VEGP COL FSAR, the applicant modified parts of FSAR Chapter 12, Appendix 12.AA that relate to STD COL 12.1-1. Specifically, in the FSAR, Revision 2, NEI 07-03A, is referenced. Accordingly, because NEI 07-03A is the approved version of NEI 07-03, the above conclusions regarding Confirmatory Item 12.1-1 are not affected by the changes to Revision 2 of the FSAR. One other change is the modification of a reference at the end of Appendix 12AA where the reference to RG 1.97 is changed from Revision 4 to Revision 3. The staff found the change acceptable, since Revision 3 provides for a more comprehensive version of the RG and also provides for portable radiation monitoring equipment. Revision 4 of RG 1.97 indicates that partial implementation is not recommended.

12.1.5 Post Combined License Activities

The post COL activities related to ALARA practices (part of the radiation protection program (RPP)) are discussed in Section 12.5.5 of this report.

12.1.6 Conclusion

The staff reviewed the WLS COL application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to ALARA, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff concludes that the relevant information presented in the WLS COL FSAR is acceptable based on meeting the relevant acceptance criteria provided in NUREG-0800, Section 12.1. The staff based its conclusion on the following:

- STD COL 12.1-1, relating to ALARA and operational policies and compliance with relevant regulatory guidance, is acceptable because the applicant incorporates approved references NEI 07-03A and NEI 07-08A into the WLS COL FSAR and meets the applicable regulatory requirements and guidance specified in Sections 12.1.3 and 12.1.4 of this report.

- STD SUP 12.1-1, relating to the use of video recording of equipment layout in areas where radiation fields are expected to be high, is acceptable because it is a statement of fact not requiring NRC approval.

12.2 Radiation Sources

12.2.1 Introduction

This section addresses the issues related to contained radiation sources and airborne radioactive material sources during normal operations, AOOs, and accident conditions affecting in-plant radiation protection.

12.2.2 Summary of Application

WLS COL FSAR, Revision 11, Section 12.2 incorporates by reference AP1000 DCD, Revision 19, Section 12.2. In addition, in WLS COL FSAR Section 12.2, the applicant provided the following:

Departures

- WLS DEP 6.4-1

The applicant provided additional information in Section 12.2 of the WLS COL FSAR about WLS DEP 6.4-1 related to design changes affecting habitability of the main control room and changes to the calculated doses to control room operators. This information, as well as related

WLS DEP 6.4-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.2 of this SER.

AP1000 COL Information Item

- STD COL 12.2-1

The applicant provided additional information in STD COL 12.2-1 to resolve COL Information Item 12.2-1 (COL Action Item 12.3.1-1) that addresses miscellaneous sources.

12.2.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. The regulatory requirements and acceptance criteria associated with the relevant requirements of NRC regulations for the radiation sources are given in NUREG-0800, Section 12.2.

The applicable regulatory requirements for STD COL 12.2-1 are as follows:

- 10 CFR 20.1801, "Security of stored material"
- 10 CFR 20.1802, "Control of material not in storage"
- 10 CFR Part 50, "Domestic licensing of production and utilization facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criteria (GDC) 61, "Fuel Storage and Handling and Radioactivity Control"

12.2.4 Technical Evaluation

The staff reviewed WLS COL FSAR Section 12.2 of the application and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to radiation sources. The results of the staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP) SER were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews.

- The staff compared the VEGP COL FSAR, Revision 5 to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed and that those changes were actualized in the WLS COL FSAR.

- The staff verified that the site-specific differences, if any, did not adversely affect any previous relevant evaluation or conclusion.

The staff completed its review and concluded that the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this report by use of italicized, double-indented formatting. Section 1.2.3 of this report provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) SER contains evaluation material from the SER for the BLN Units 3 and 4 COL application SER.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 12.2.4:

> *The following portion of this technical evaluation section is reproduced from Section 12.2.4 of the BLN SER:*
>
> *<u>AP1000 COL Information Item</u>*
>
> - *STD COL 12.2-1*
>
> *The applicant provided additional information in STD COL 12.2-1, related to miscellaneous sources, to resolve COL Information Item 12.2-1. COL Information Item 12.1-1 states:*
>
> > *The Combined License applicant will address any additional contained radiation sources not identified in subsection 12.2.1, including radiation sources used for instrument calibration or radiography.*
>
> *The same commitment was also captured as COL Action Item 12.3.1-1 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793).*
>
> *The applicant provided additional information in the BLN COL FSAR to address the plant STD COL 12.2-1 dealing with miscellaneous sources. The applicant stated that licensed sources containing byproduct, source and special nuclear material that warrant shielding consideration will meet the applicable requirements of 10 CFR Parts 20, 30, 31, 32, 33, 34, 40, 50 and 70. The applicant indicated that there are byproducts and source materials with known isotopes and activity manufactured for the purpose of measuring, checking, calibrating, or controlling processes quantitatively or qualitatively. Accordingly, written procedures will be established and implemented that address procurement, receipt, inventory, labeling, leak testing, surveillance, control, transfer, disposal, storage, issuance and use of these radioactive sources. Also, the applicant indicated that sources maintained on-site for instrument calibration purposes will be shielded while in storage to keep personnel exposure ALARA.*

The regulatory requirements cited in the above paragraph address the requirements applicable to sources that would likely be used in conjunction with construction, preoperational, and initial testing. The applicant will implement the practices for radioactive material control as described in NEI 07-03, Section 12.5.4.10, "Radioactive Material Control." In a letter dated March 18, 2009 (ML090510379), the NRC accepted NEI 07-03, Revision 7. Specifically, the NRC staff indicated that for COL applications, NEI 07-03, Revision 7 provides an acceptable template for assuring that the RPP meets the applicable NRC regulations and guidance. Since the BLN FSAR has not adopted the approved version of the NEI template, this is identified as Confirmatory Item 12.1-1.

The staff concludes that the information provided by the applicant with respect to radiation sources is acceptable and meets the requirements of 10 CFR Sections 20.1801 and 20.1802 and GDC 61. This conclusion is based on the applicant's commitment to the NEI 07-03 administrative controls to meet the regulatory requirements. These controls apply to the additional contained radiation sources discussed in the COL item. The staff notes that its review did not encompass the entire set of regulatory requirements cited by the applicant (10 CFR Parts 20, 30, 31, 32, 33, 34, 40, 50 and 70), since the staff's review is focused on radiation protection requirements on sources used in conjunction with the RPP.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 12.2.4:

Resolution of Standard Content Confirmatory Item 12.1-1

The NRC staff compared the VEGP and BLN COL applications regarding STD COL 12.2-1, and found them to be essentially identical, with the exception that VEGP FSAR Appendix 12AA references NEI 07-03A, whereas, the BLN FSAR references NEI 07-03, Revision 3. As indicated in Section 12.1.4 above, Confirmatory Item 12.1-1, is resolved for VEGP because the applicant has adopted the approved version of NEI 07-03, which is now designated as NEI 07-03A.

12.2.5 Post Combined License Activities

There are no post COL activities related to this section.

12.2.6 Conclusion

The staff reviewed the WLS COL application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to radiation sources, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements. The staff concludes that the relevant information presented in the WLS COL

FSAR is acceptable based on the relevant acceptance criteria provided in NUREG-0800, Section 12.2. The staff based its conclusion on the following:

- WLS DEP 6.4-1, related to design changes affecting habitability of the main control room and changes to the calculated doses to control room operators, is reviewed and found acceptable by the staff in Section 21.2 of this SER.

- The staff finds that STD COL 12.2-1, which addresses miscellaneous sources, is acceptable because the applicant has incorporated the approved reference NEI 07-03A into the WLS COL FSAR and meets the requirements of 10 CFR 20.1801, 10 CFR 20.1802, and GDC 61.

12.3 Radiation Protection Design Features

WLS COL FSAR Section 12.3, "Radiation Protection Design Features" and the following Section 12.4, "Dose Assessment," are treated as separate sections in this report (as well as in the AP1000 DCD). However, these two sections are listed as a single section, Section 12.3-12.4, "Radiation Protection Design Features," in NUREG-0800, with the material discussed under "Dose Assessment" included at the end of Section 12.3-12.4.

12.3.1 Introduction

This section addresses the issues related to radiation protection equipment and design features used to ensure that OREs are ALARA. The staff review takes into account design dose rates, AOOs, and accident conditions. Issues considered by the staff include the facility design features, shielding, ventilation, area radiation and airborne radioactivity monitoring instrumentation, and dose assessment.

12.3.2 Summary of Application

WLS COL FSAR, Revision 11, Section 12.3 incorporates by reference AP1000 DCD, Revision 19, Section 12.3. The WLS COL FSAR Section 12.3, provided the following additional information:

Departures

- WLS DEP 6.4-1

The applicant provided additional information in Section 12.3 of the WLS COL FSAR about WLS DEP 6.4-1 related to design changes affecting habitability of the main control room and changes to the calculated doses to control room operators. This information, as well as related WLS DEP 6.4-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.2 of this SER.

- WLS DEP 18.8-1

The applicant described the following portion of the Tier 2 departure (DEP) from the AP1000 DCD related to the radiation design protection features. The applicant proposed revising several DCD figures in WLS COL FSAR Section 12.3 to reflect the relocation of the Operations Support Center (OSC). Other aspects of this Tier 2 departure are evaluated in Sections 12.5, 13.3, and 18.8 of this report.

AP1000 COL Information Items

- STD COL 12.3-1

The applicant provided additional information in STD COL 12.3-1 to resolve COL Information Item 12.3-1 (COL Action Item 12.4.2-1) that addresses the administrative controls for use of the design features provided to control access to radiological restricted areas.

- STD COL 12.3-2

The applicant provided additional information in STD COL 12.3-2 to resolve COL Information Item 12.3-2 (COL Action Item 12.4.4-1) that addresses the criteria and methods for obtaining representative measurement of radiological conditions, including airborne radioactivity concentrations in work areas.

- STD COL 12.3-3

The applicant provided additional information in STD COL 12.3-3 to resolve COL Information Item 12.3-3 that addresses the groundwater monitoring program beyond the normal radioactive effluent monitoring program.

- STD COL 12.3-4

The applicant provided additional information in STD COL 12.3-4 to resolve COL Information Item 12.3-4 that addresses the program to ensure documentation of operational events deemed to be of interest for decommissioning.

Supplemental Information

- WLS SUP 11.2-3

In a September 20, 2011, letter, the applicant provided supplemental information in WLS COL FSAR Section 11.2.1.2.4. The supplemental information added to this WLS COL FSAR section clarifies the locations of radwaste monitoring and describes the features of the exterior liquid radwaste system discharge pipeline incorporated to minimize leakage to the environment. This section also states that the diluted liquid radwaste effluent will be discharged to the Broad River.

12.3.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. The regulatory requirements and acceptance criteria associated with the relevant requirements of NRC regulations for the radiation protection design features are given in NUREG-0800, Section 12.3-12.4.

The applicable regulatory requirements and guidance for STD COL 12.3-1 are as follows:

- 10 CFR Part 20
- RG 1.8, Revision 3
- RG 8.9, “Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program,” Revision 1
- RG 8.38, “Control of Access to High and Very High Radiation Areas in Nuclear Power Plants,” Revision 1
- NUREG-1736

The applicable regulatory requirements and guidance for STD COL 12.3-2 are as follows:

- 10 CFR Part 19, “Notices, instructions and reports to workers: inspection and investigations”
- 10 CFR Part 20
- 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities”
- NUREG-0737, ”Clarification of TMI Action Plan Requirements,” Item III.D.3.3
- RG 1.8, Revision 3
- RG 8.2, “Guide for Administrative Practices in Radiation Monitoring,” Revision 0
- RG 8.8, Revision 3
- RG 8.10, Revision 1-R
- RG 1.21, “Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants,” Revision 1, Appendix A, “Measuring Radioactive Materials in Liquid and Gaseous Effluents and Solid Waste”
- RG 1.97, Revision 4

The applicable regulatory requirements and guidance for STD COL 12.3-3 and STD COL 12.3-4 are as follows:

- 10 CFR 20.1406, “Minimization of contamination”
- 10 CFR 50.75, “Reporting and recordkeeping for decommissioning planning”
- RG 4.21, “Minimization of Contamination and Radioactive Waste Generation: Life Cycle Planning,” Revision 0

12.3.4 Technical Evaluation

The staff reviewed WLS COL FSAR Section 12.3 and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to radiation protection design features. The results of the staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the VEGP reference COL application SER were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed and that those changes were actualized in the WLS COL FSAR.

- The staff verified that the site-specific differences, if any, did not adversely affect any previous relevant evaluation or conclusion.

The staff completed its review and finds the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this report by use of italicized, double-indented formatting. Section 1.2.3 of this report provides an explanation of why the standard content material from the VEGP reference COL application SER contains evaluation material from the BLN Units 3 and 4 COL application SER. Any confirmatory items in the standard content material retain the numbers assigned in the VEGP SER. Confirmatory items that are first identified in this report have a WLS designation (e.g., WLS Confirmatory Item 12.3-1).

Departures

- WLS DEP 18.8-1

The location of the WLS Units 1 and 2 OSC differs from the location described in the AP1000 DCD. In the AP1000 DCD description, the OSC is located in the same room as the ALARA Briefing Room in the Annex Building. For WLS Units 1 and 2, the OSC has been relocated to where the DCD states that the TSC will be located. At WLS, the TSC, in turn, has been relocated to the basement of the Maintenance Support Building, which is separate from the Unit 1 and 2 Nuclear Islands. The staff finds the applicant's proposed revision to several DCD figures to reflect the relocation of the OSC is acceptable because the location of the OSC does not have an impact on the radiation protection facilities design.

The evaluation of the effect of the OSC relocation is addressed in Section 12.5 of this report for the health physics (HP) facilities, in Section 13.3 of this report for emergency preparedness, and in Section 18.8 of this report for the human system interface design.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 12.3.4:

> *The following portion of this technical evaluation section is reproduced from Section 12.3.4 of the BLN SER:*
>
> *AP1000 COL Information Items*
>
> - *STD COL 12.3-1*
>
> *The applicant provided additional information in STD COL 12.3-1, related to the administrative controls for radiological protection, to resolve COL Information Item 12.3-1. COL Information Item 12.3-1 states:*
>
> > *The Combined License applicant will address the administrative controls for use of the design features provided to control access to radiologically restricted areas, including potentially very high radiation areas, such as the fuel transfer tube during refueling operations and to the reactor cavity.*
>
> *The commitment was also captured as COL Action Item 12.4.2-1 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:*
>
> > *The COL applicant will address the administrative controls for use of the design features provided to control access to radiologically restricted areas, including potentially very high radiation areas, such as the reactor cavity and the fuel transfer canal during refueling operations. The hatch to the spent fuel transfer canal will be treated as an entrance to a very high radiation area under 10 CFR Part 20 and will be locked during spent fuel transfer operations.*
>
> *The applicant addressed this STD COL item in BLN COL FSAR, Appendix 12AA. This appendix incorporates by reference NEI 07-03, Revision 7* [sic]*. The NEI template directs COL applicants to describe the site-specific plant information for areas requiring administrative controls for very high radiation areas. To supplement NEI 07-03, Section 12.5.4.4, "Access Control," the applicant provided additional measures in Appendix 12AA for access controls such as signs, locks, plant manager (or designee) approval for entry, and radiation protection personnel accompaniment and exposure control for entry into very high radiation areas. The applicant also stated that a closed circuit television system may be installed in high radiation areas to allow remote monitoring of individuals entering high radiation areas by personnel qualified in radiation protection procedures.*

The COL applicant will apply the administrative controls for the use of the design features to control access to very high radiation areas, such as the fuel transfer tube during refueling and to the reactor cavity during operations, and other radiologically restricted areas to comply with 10 CFR Sections 20.1601 and 20.1602 The opening of the fuel transfer hatch is administratively controlled, treated as an entrance to a very high radiation area, and is in place during spent fuel transfer operation.

The staff finds the applicant's approach meets the requirements of 10 CFR Sections 20.1601 and 20.1602, and is consistent with RG 8.38, Regulatory Position C1 and C3, which will ensure that an individual is unable to gain unauthorized or inadvertent access to such areas.

In a letter dated March 18, 2009 (ML090510379), the NRC accepted NEI 07-03, Revision 7. Specifically, the NRC staff indicated that for COL applications, NEI 07-03, Revision 7 provides an acceptable template for assuring that the RPP meets the applicable NRC regulations and guidance. Since the BLN FSAR has not adopted the approved version of the NEI template, this is identified as Confirmatory Item 12.1-1.

The NRC staff reviewed STD COL 12.3-1 dealing with administrative controls for radiological protection, using the text added in Appendix 12AA. The BLN COL FSAR Appendix 12AA, incorporates by reference NEI 07-03.

In Appendix 12AA, the applicant has taken exception to NEI 07-03, Section 12.5 to not conform to the guidance of the following regulatory guides:

- *RG 8.20, "Applications for Bioassay for I-125 and I-131"*
- *RG 8.26* [sic]*, "Bioassay at Uranium Mills"*
- *RG 8.32, "Criteria for Establishing a Tritium Bioassay Program"*

The guidance documents were identified as outdated regulatory guidance in NUREG-1736, Consolidated Guidance: 10 CFR Part 20, "Standards for Protection Against Radiation," October 2001. NUREG-1736 describes that in conjunction with 10 CFR 20.1502(b), which requires licensees to monitor for likely intakes; 10 CFR 20.1204(a) and (b) prescribe how information obtained through monitoring is to be used when assessing exposures to workers from intakes. The NUREG recommends that licensees (and therefore applicants) consider the methods described in RG 8.9, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program," for estimating intakes of radionuclides and determining the frequency of bioassay measurements. RG 8.9 provides updated methods and guidance that was previously contained in positions of the three RGs above. The applicant's commitment to RG 8.9 is sufficient to assure proper monitoring for intake of radionuclides.

In BLN COL FSAR, Appendix 12AA, the applicant took exception to the first paragraph of NEI 07-03, Section 12.5.2 to describe the equivalent key radiological protection positions for the BLN site. The description of *organizational positions with specific radiation protection responsibilities is in BLN COL FSAR Section 13.1. BLN COL FSAR Section 13.1, "Organizational Structure of the Applicant," provides specific radiation protection responsibilities for key positions within the plant organization and the plant organization overall. Managers and supervisors within the plant operating organization are responsible for establishing goals and expectations for their organization and to reinforce behaviors that promote radiation protection. BLN COL FSAR Section 13.1.1, "Management and Technical Support Organization," and Section 13.1.2, "Operating Organization," provide the responsibilities of the organizations and positions to assure that radiological safety goals and expectations are adhered to.*

The staff finds that the applicant's exception to NEI 07-03, Section 12.5.2 is acceptable because BLN COL FSAR Section 13.1 provides the key radiological safety responsibilities and organization consistent with RG 1.8.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 12.3.4:

Correction of Errors in the Standard Content Evaluation Text

The NRC staff identified an error in the text reproduced above from the BLN SER, Section 12.3.4, that requires correction. The BLN SER states that Appendix 12AA of the BLN COL FSAR incorporates by reference NEI 07-03, Revision 7. The appendix actually incorporates by reference NEI 07-03, Revision 3. The NRC staff also identified an error in the text reproduced above from the BLN SER, Section 12.3.4 regarding the reference to RG 8.22, which was incorrectly referred to as RG 8.26.

Resolution of Standard Content Confirmatory Item 12.1-1

The NRC staff compared the VEGP and BLN COL applications regarding STD COL 12.3-1, and found them to be essentially identical, with the exception that VEGP FSAR Appendix 12AA references NEI 07-03A and BLN FSAR Appendix 12AA references Revision 3 of NEI 07-03. Additional clarifying information has been added to the VEGP FSAR regarding STD COL 12.3-1, which is discussed below. As indicated in Section 12.1.4 above, Confirmatory Item 12.1-1, is resolved for VEGP because the applicant has adopted the approved version of NEI 07-03, which is now designated as NEI 07-03A.

In addition, changes have been made in Revision 2 of the VEGP FSAR Chapter 12 that relate to STD COL 12.3-1. The changes are as follows:

1. *A new Table 12AA-201 has been added to Appendix 12AA that provides information concerning access to very high radiation areas (VHRA). The table provides VHRA locations, DCD cross references, radiation sources in the locations and other conditions and restrictions.*

2. *In FSAR Appendix 12AA, new text was added to Section 12.5.4.4 of NEI 07-03A. The text references new Table 12AA-201 and describes the*

> *information in it, discusses removal of the primary sources of radiation from the VHRA areas, and discusses verification walk downs of VHRA to ensure consistency with RG 8.38. In addition to the changes to Appendix 12AA discussed above, the applicant has also added text to Section 12.5.4 regarding the possible use of closed circuit television system to allow remote monitoring of individuals entering high radiation areas.*

These items (i.e., the addition of the table, reference to it and discussion of walk downs, and the closed circuit television system) are acceptable because they provide additional clarity and site-specific information regarding controls to VHRAs and more completely describe features that address STD COL 12.3-1.

The following portion of this technical evaluation section is reproduced from Section 12.3.4 of the BLN SER.

- *STD COL 12.3-2*

The applicant provided additional information in STD COL 12.3-2, related to the criteria and methods for radiological protection, to resolve COL Information Item 12.3-2. COL Information Item 12.3-2 states:

> *The Combined License applicant will address the criteria and methods for obtaining representative measurement of radiological conditions, including airborne radioactivity concentrations in work areas. The Combined License applicant will also address the use of portable instruments, and the associated training and procedures, to accurately determine the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.*

The same commitment was also captured as COL Action Item 12.4.4-1 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793).

The staff reviewed STD COL 12.3-2, dealing with criteria and methods for radiological protection. In BLN COL FSAR Section 12.3.4, the applicant presented the procedure detailing the criteria and methods for obtaining representative measurement of radiological conditions, including in-plant airborne radioactivity concentrations in accordance with applicable portions of 10 CFR Part 20 and consistent with the guidance in RGs 1.21, Appendix A, 8.2, 8.8, and 8.10.

The applicant also discussed the surveillance requirements and the frequency of scheduled surveillance that are consistent with the operational philosophy in RG 8.10. In Section 12.3.4, "Area Radiation and Airborne Radioactivity Monitoring Instrumentation," the applicant described the typical survey frequencies and varieties of surveys. The surveys described in general terms include radiation, contamination, airborne radioactivity, and job coverage surveys for occupational radiation workers during normal and off-normal conditions.

Appendix 12AA also describes qualification and training criteria for site personnel consistent with the guidance in RG 1.8 and as described in FSAR Chapter 13. Section 13.2, "Training," incorporates NEI 06-13A, "Template for an Industry Training Program Description." NEI 06-13A, Section 1.2.7, provides training for the use of survey instruments, use of analytical equipment, radiation protection procedures and emergency plan procedures.

The applicant discussed a portable iodine monitoring system used to determine the airborne iodine concentration in areas where plant personnel may be present routinely and during an accident which meets the guidance of NUREG-0737, Item III.D.3.3 and complies with 10 CFR Part 50, Appendix A. The applicant will incorporate the use of this sampling system into the emergency plan implementing procedures.

The NRC staff reviewed BLN COL FSAR Section 12.3.4 and Appendix 12AA, dealing with standards applied to the calibration and maintenance of portable radiation survey instruments. The applicant describes Area and Airborne Radioactivity Monitoring Instrumentation in BLN COL FSAR Section 12.3.4 and also in Section 14.2.9.4.27, "Portable Personnel Monitors and Radiation Survey Instruments."

The portable personnel monitor and radiation survey instrument testing verifies that the devices operate in accordance with their intended function in support of the RPP as described in Chapter 12. The applicant stated as a prerequisite that the monitors, instruments and certified test sources are on site. The applicant also stated that the general test method and acceptance criteria for the monitors and instruments would be source checked and tested in accordance with the manufactures' recommendations. The NRC staff determined that additional information should be provided in addition to the use of manufacturers' recommendations. Additional standards such as American National Standards Institute (ANSI) N42.17A-1989, as it relates to the accuracy and overall performance of portable survey instruments, and ANSI N323A-1997, as it relates to the calibration and maintenance of portable radiation survey instruments should be provided. In response to RAI 12.3-12.4-5, in a letter from the applicant, dated September 22, 2008; the applicant stated that it intends to revise the BLN COL FSAR to include maintenance and calibration of survey instruments and to update the version of the ANSI standard in a future revision of the COL application. The NRC staff finds that Revision 1 of the BLN COL FSAR adequately addresses the above. As a result, RAI 12.3-12.4-5 is closed.

- *STD COL 12.3-3*

The applicant provided additional information in STD COL 12.3-3, related to the groundwater monitoring program, to resolve COL Information Item 12.3-3. COL Information Item 12.3-3 states:

> *The Combined License applicant will establish a groundwater monitoring program beyond the normal radioactive effluent monitoring program. If and as necessary to support this*

groundwater monitoring program, the Combined License applicant will install groundwater monitoring wells during the plant construction process. Areas of the site to be specifically considered in this groundwater monitoring program are as follows:

- *West of the auxiliary building in the area of the fuel transfer canal*
- *West and south of the radwaste building*
- *East of the auxiliary building rail bay and the radwaste building truck doors*

The applicant added text in BLN COL FSAR Appendix 12AA, Section 12AA.5.4.14 to the information incorporated from NEI 07-03 regarding the groundwater monitoring program.

The applicant stated that a groundwater monitoring program beyond the normal radioactive effluent monitoring program will be developed, if, and as necessary to support this groundwater monitoring program, design features will be installed during the plant construction process. The applicant discussed areas of the site to be specifically considered in this groundwater monitoring program.

The NRC staff evaluated the applicant's groundwater monitoring program to the criteria in 10 CFR 20.1406. 10 CFR 20.1406 requires the applicant to provide a description of how facility design and procedures for operation will minimize, to the extent practicable, contamination of the facility and the environment; facilitate eventual decommissioning; and minimize, to the extent practicable, the generation of radioactive waste. The regulatory guidance which describes an acceptable method for meeting the regulation was published in June 2008, RG 4.21, Revision 0, "Minimization of Contamination and Radioactive Waste Generation: Life Cycle Planning."

The groundwater monitoring program as described in BLN COL FSAR Appendix 12AA included some implementation considerations, but the program lacked a description of the key components of the program such as, types and periodicity of routine samples, threshold activity to be detected, actions to be taken upon detection, and quality assurance practices to be used to ensure reasonable assurance of prompt identification of leakage into the groundwater (RAI 12.3-12.4-1 and RAI 12.3-12.4-2).

The applicant stated in a letter dated September 22, 2008, that it will adopt the NEI 08-08, "Generic FSAR Template Guidance for Life Cycle Minimization of Contamination," Revision 0 template. If approved by the NRC, the applicant will provide additional description of site specific design features and procedures for operation that minimize contamination of the facility, site, and environment. NEI 08-08 is currently under staff review. This is identified as Open Item 12.3-1.

As described in Section 11.2.1 2.4 of the AP1000 DCD, Revision 17, the exterior monitored liquid effluent discharge pipe is engineered to preclude leakage by

> *either enclosure within a guard pipe and leakage monitoring, or is accessible for visual inspection in total from the Radwaste Building to the licensed release point for dilution and discharge. No valves, vacuum breakers, or other fittings are incorporated outside of buildings. In a supplemental response dated December 16, 2008, to RAI 12.3-12.4-1, the applicant provided a proposed revision to the BLN COL FSAR to describe the site-specific design of the external radioactive waste discharge line. The staff agrees with the applicant that the site-specific design will minimize the potential for undetected leakage from this discharge to the environment at a non-licensed release point, and complies with 10 CFR 20.1406. The proposed change to the BLN COL FSAR is acceptable subject to a formal revision to the BLN COL FSAR. Accordingly, this is identified as Confirmatory Item 12.3-1.*

The following portion of this technical evaluation section is reproduced from VEGP SER Section 12.3.4:

> *Resolution of Standard Content Open Item 12.3-1*
>
> *Revision 2 of the FSAR references NEI 08-08A, which is the version of NEI 08-08 that has been accepted by NRC. Accordingly, Open Item 12.3-1 is resolved for VEGP.*
>
> *Resolution of Standard Content Confirmatory Item 12.3-1*
>
> *The NRC staff verified that Section 11.2.1.2.4 of the VEGP FSAR was updated to include the information identified in BLN Confirmatory Item 12.3-1; therefore, Confirmatory Item 12.3-1 is resolved for VEGP*.

Supplemental Information

- WLS SUP 11.2-3

The radwaste discharge piping, described above, runs from the auxiliary building to the radwaste building and then out of the radwaste building to the licensed release point for dilution and discharge. The last paragraph of the standard content evaluation of STD COL 12.3-3, reproduced from BLN SER Section 12.3.4 above, provides the staff's evaluation of the exterior radwaste discharge piping for BLN. In a May 22, 2009, letter to NRC, the WLS applicant endorsed BNL's response to BLN RAI 12.3-12.4-1.

Although the WLS applicant described the above mentioned portions of the radwaste discharge piping in WLS COL FSAR Section 11.2, the applicant did not initially provide a description of the site-specific design portions of the external radioactive waste discharge line in the WLS COL FSAR. In a September 20, 2011, letter, the applicant provided supplemental information in WLS COL FSAR Section 11.2.1.2.4 regarding site-specific design features of the external radioactive waste discharge line. In WLS SUP 11.2-3, the applicant stated that the exterior liquid radwaste system discharge pipe from the Radwaste Building for each unit (Units 1 and 2) is stainless steel and is enclosed within a high-density polyethylene (HDPE) guard pipe. The annular space between the liquid radwaste discharge pipe and the guard pipe is monitored for leakage at low points along the path to comply with the requirements of 10 CFR 20.1406. No valves or vacuum breakers are incorporated in exterior radwaste discharge pipe. The HDPE guard pipe surrounding the radwaste discharge pipe is continuous up to the underground pit on the western bank of the Broad River where the liquid radwaste pipe ties into the outfall pipe. It is at this point that the liquid radwaste effluent from the liquid radwaste pipe is diluted by the high volume circulating water system blowdown stream from the outfall pipe to meet the release limits of 10 CFR Part 20, Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewage," Table II, Column 2. The underground pit where this mixing takes place is monitored for leakage to comply with the requirements of 10 CFR 20.1406. The resulting diluted flow will be discharged to the Broad River via a submerged multi-port diffuser on the Ninety-Nine Islands Dam spillway.

RG 4.21 states that applicants should strive to minimize leaks and spills, provide containment in areas where such events might occur, and provide for detection that supports timely assessment and appropriate response. In accordance with the guidance of RG 4.21, and to comply with the requirements of 10 CFR 20.1406, the applicant will provide monitoring points to facilitate manual sampling for leakage from the liquid radwaste system discharge pipeline. NEI 08-08A, "Generic FSAR Template Guidance for Life Cycle Minimization of Contamination," states that the COL applicant should establish an on-site ground water monitoring program to ensure timely detection of inadvertent radiological releases to the ground water. In accordance with NEI 08-08A, the applicant has modified its radiation protection program described in WLS COL FSAR Section 12AA to include a description of a groundwater monitoring program. This groundwater monitoring program will include a network of wells to ensure timely detection of inadvertent radiological releases to the groundwater.

The staff finds that the design features of the site-specific design portions of the external radioactive waste discharge line, which are described in WLS SUP 11.2-3, will minimize the potential for undetected leakage from this discharge to the environment in accordance with the guidance of RG 4.21. On the basis of the staff's review of the information provided in WLS SUP 11.2-3, the staff concludes that the discharge piping design features and implementation of the groundwater monitoring program meet the requirements of 10 CFR 20.1406 for minimizing the potential for the contamination of the environment and finds WLS SUP 11.2-3 acceptable.

The following portion of this technical evaluation section is reproduced from Section 12.3.4 of the VEGP SER.

> *The following portion of this technical evaluation section is reproduced from Section 12.3.4 of the BLN SER.*
>
> - *STD COL 12.3-4*
>
> *The applicant provided additional information in STD COL 12.3-4, related to the record of operational events of interest for decommissioning, to resolve COL Information Item 12.3-4. COL Information Item 12.3-4 states:*
>
> > *The Combined License applicant will establish a program to ensure documentation of operational events deemed to be of interest for decommissioning, beyond that required by*

10 CFR 50.75. This or another program will include remediation of any leaks that have the potential to contaminate groundwater.

The applicant added text in Appendix 12AA, Section 12AA.5.4.15 to the information incorporated from NEI 07-03 dealing with a record of operational events of interest for decommissioning. The applicant discussed procedures established to document the operational events that are deemed of interest for decommissioning, beyond that required by 10 CFR 50.75. These documented operational events assist in developing a historical assessment of the nuclear facilities, thereby reducing time, effort, and hazards to personnel during decommissioning planning. This documentation will include identification of the remediation of any leaks, which have the potential to contaminate groundwater. The procedures that govern retention of these records, and the records themselves, should specify the retention period required to assure availability when they may be required (e.g., life of facility plus 30 years). The NRC staff requested in RAI 12.3-12.4-3 that the applicant include the operational and design COL information items that fully meet the objectives of RG 4.21, Revision 0 and hence the requirements of 10 CFR 20.1406, 'Minimization of Contamination."

In response to the RAI, in a letter dated September 22, 2008, the applicant stated that it intended to adopt NEI 08-08. This document is intended to provide the description of additional site procedures for decommissioning records which will demonstrate compliance with 10 CFR 20.1406. This is identified as Open Item 12.3-1.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 12.3.4:

<u>Resolution of Standard Content Open Item 12.3-1</u>

Revision 2 of the FSAR references NEI 08-08A, which is the version of NEI 08-08 that has been accepted by NRC. Accordingly, Open Item 12.3-1 is resolved for VEGP.

12.3.5 Post Combined License Activities

There are no post COL activities related to this section.

12.3.6 Conclusion

The staff reviewed the WLS COL application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to the radiation protection design features, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements. In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable. The staff based its conclusion on the relevant acceptance criteria provided in NUREG-0800, Section 12.3-12.4 and on the following:

- WLS DEP 6.4-1, related to design changes affecting habitability of the main control room and changes to the calculated doses to control room operators, is reviewed and found acceptable by the staff in Section 21.2 of this SER.

- WLS DEP 18.8-1, which the applicant proposed to relocate the OSC from the location described in the AP1000 DCD, Section 12.5.2.2, is acceptable from a radiation protection design features perspective. The location of the OSC does not have an impact on the radiation protection facilities design. The ALARA briefing room remains as stated in the ASP1000 DCD, so there is no impact on radiation protection facilities, programs or functions.

- STD COL 12.3-1, which addresses the administrative controls for use of the design features provided to control access to radiological restricted areas, is acceptable because the applicant has incorporated the approved reference NEI 07-03A into the WLS COL FSAR and meets the applicable regulatory requirements and guidance specified in Sections 12.3.3 and 12.3.4 of this report.

- STD COL 12.3-2, which addresses the criteria and methods for obtaining representative measurement of radiological conditions, including airborne radioactivity concentrations in work areas, is acceptable because the applicant has demonstrated compliance with the applicable regulatory requirements and guidance specified in Sections 12.3.3 and 12.3.4 of this report.

- STD COL 12.3-3 and WLS SUP 11.2-3, which address the groundwater monitoring program beyond the normal radioactive effluent monitoring program, are acceptable because the applicant has incorporated the approved reference NEI 08-08A into the WLS COL FSAR in order to demonstrate compliance with the applicable regulatory requirements and guidance specified in Sections 12.3.3 and 12.3.4 of this report.

- STD COL 12.3-4, which addresses the program to ensure documentation of operational events deemed to be of interest for decommissioning, is acceptable because the applicant has incorporated the approved reference NEI 08-08A into the WLS COL FSAR and meets the applicable regulatory requirements and guidance specified in Sections 12.3.3 and 12.3.4 of this report.

12.4 Dose Assessment

12.4.1 Introduction

This section of the report addresses the issues related to estimating the annual personnel doses associated with operation, normal maintenance, radwaste handling, refueling, ISI and special maintenance (e.g., maintenance that goes beyond routine scheduled maintenance, modification of equipment to upgrade the plant, and repairs to failed components), and construction.

12.4.2 Summary of Application

WLS COL FSAR, Revision 11, Section 12.4 incorporates by reference AP1000 DCD, Revision 19, Section 12.4. In addition, in WLS COL FSAR Section 12.4, the applicant provided the following:

Supplemental Information

- WLS SUP 12.4-1

The applicant provided site-specific supplemental information to address dose to construction workers by adding new sections after AP1000 DCD Section 12.4.1.8.

- STD SUP 12.4-1

The applicant provided supplemental information regarding conduct of radiological surveys in unrestricted and controlled areas and for radioactive materials in effluents discharged to unrestricted and controlled areas.

12.4.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. In addition, the regulatory requirements and acceptance criteria associated with the relevant requirements of NRC regulations for the dose assessment are given in NUREG-0800, Section 12.4.

The applicable regulatory requirements for WLS SUP 12.4-1 and STD SUP 12.4-1 are as follows:

- 10 CFR 20.1101
- 10 CFR 20.1301, “Dose limits for individual members of the public”
- 10 CFR 20.1302, “Compliance with dose limits for individual members of the public”

12.4.4 Technical Evaluation

The staff reviewed WLS COL FSAR Section 12.4 and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The staff’s review confirmed that the information in the application and incorporated by reference addresses the required information relating to dose assessment. The results of the staff’s evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff’s findings on standard content that were documented in the VEGP reference COL application SER were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed and that those changes were actualized in the WLS COL FSAR.

- The staff verified that the site-specific differences, if any, did not adversely affect any relevant evaluation of conclusion.

The staff completed its review and finds the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this report by use of italicized, double-indented formatting.

Supplemental Information

- WLS SUP 12.4-1

The applicant provided supplemental information regarding dose to construction workers in WLS COL FSAR Section 12.4.1.9 (comprised of Sections 12.4.1.9.1 through 12.4.1.9.6), "Dose to Construction Workers." The first section, 12.4.1.9.1, "Site Layout," describes the site layout as depicted in WLS COL FSAR Figure 2.1-201.

WLS COL FSAR Section 12.4.1.9.2, "Radiation Sources," identifies the sources of radiation that would be encountered by construction workers. Since there are currently no operating units on the site, construction workers are not exposed to any radiation sources other than background radiation until WLS Unit 1 becomes operational. At that time, construction workers would potentially receive radiation exposures from direct radiation, gaseous effluents, and liquid effluents from the operation of WLS Unit 1. As stated in AP1000 DCD Section 12.4.2, direct radiation from the WLS Unit 1 containment and other plant buildings is negligible. In addition, there is no direct dose contribution from the refueling water since the refueling water is stored inside the containment instead of in an outside storage tank.

WLS COL FSAR Section 12.4.1.9.3, "Construction Worker Dose Estimates," includes the assumptions and bases used to calculate the annual construction worker exposure estimates. The applicant used the XOQDOQ computer code (discussed in NRC's NUREG/CR-2919, "XOQDOQ: Computer Program for the Meteorological Evaluation of Routine Effluent Releases at Nuclear Power Stations") to determine the X/Q and D/Q values for the nearest location along the Unit 1 protected area fence in each direction as well as for the point on the Unit 2 shield building closest to Unit 1. The Unit 2 shield building is the principal construction area for Unit 2 workers. The applicant then used the NRC-accepted GASPAR computer code to estimate the gaseous effluent dose contribution from the operation of Unit 1 to Unit 2 construction workers. The applicant summed the individual dose contributions from external exposure to contaminated ground, external exposure to noble gas radionuclides in the airborne plume, and internal exposures from the inhalation of air in arriving at a maximum annual estimated dose contribution of 2.9 (micro Sieverts (µSv)) (0.29 mrem) to Unit 2 construction workers from the gaseous effluents from Unit 1. The calculated dose from routine gaseous effluents is to a construction worker at the point on the Unit 2 shield building closest to Unit 1. The staff performed an independent assessment of the construction worker exposure resulting from the Unit 1 gaseous effluent releases. The result of the staff's evaluation supported the applicant's conclusion regarding its conservative assessment of the construction worker dose from Unit 1 gaseous effluents.

In a December 20, 2012, letter, the applicant notified the NRC of a design change to the physical locations of the WLS Nuclear Islands. The WLS Unit 1 Nuclear Island is shifted 20 m (66 ft) south and 15 m (50 ft) east from the original location. The WLS Unit 2 Nuclear Island is shifted 20 m (66 ft) south. These changes resulted in a change in the relative distances between the two units as well as a change in the downwind distances to the construction worker locations. On the basis of these changes, the applicant calculated revised X/Q and D/Q values and then used the GASPAR computer code to estimate the revised gaseous effluent dose contributions from the operation of Unit 1 to Unit 2 construction workers. The revised annual estimated gaseous dose contribution to a Unit 2 worker at the nearest point of the Unit 2 shield building increased from 2.9 µSv (0.29 mrem) to 3.97 µSv (0.397 mrem), based on the relocation of the units. The applicant determined that the hourly dose due to routine gaseous effluents would occur where the highest dose rates could be expected, at the southeast sector of the Unit 1 fence line. Assuming the worker remains at this location on the fence line for all working hours of the entire year, the applicant estimated that the worker would receive 53.7 µSv (5.37 mrem) in a year. This would equate to a maximum dose in any one hour of 0.0285 µSv (2.85xE-3 mrem). The staff performed an independent assessment to verify the acceptability of these revised worker dose assessments from Unit 1 gaseous effluents.

Any potential dose contribution to Unit 2 construction workers from liquid effluents would be from construction performed to tie in the Unit 2 liquid effluent discharge and blowdown piping to the Unit 1 liquid effluent discharge and blowdown piping. The applicant stated, however, that any work done to tie in the Unit 2 discharge/blowdown piping after Unit 1 is operating will be performed under the Unit 1 radiation protection program. In addition, since the potable water for the site is provided by the Draytonville Water District, which is upstream of the WLS site, there will be no construction worker exposure from this pathway. Therefore, the applicant stated that there will be no dose contribution to Unit 2 construction workers from liquid effluents from the operation of Unit 1.

In WLS COL FSAR Section 12.4.1.9.4, "Compliance with Dose Regulations," the applicant stated that construction workers are classified as members of the public and identifies the regulatory requirements that are applicable to their exposures. Since Unit 2 construction workers are considered to be members of the general public, they are limited by 10 CFR 20.1301 and 10 CFR 20.1302 to 1 mSv (100 mrem) total effective dose equivalent (TEDE) in a year and less than 0.02 mSv (2 mrem) in any one hour, respectively. As stated above, the dose contribution to Unit 2 construction workers from both direct dose and liquid effluents would be negligible. Therefore, the applicant stated that the only dose contributions to Unit 2 construction workers from the operation of Unit 1 would be from gaseous effluents. Based on a construction worker exposure time of 2080 hours per year, the applicant estimates that Unit 2 construction workers would receive an annual dose of 3.97 µSv (0.397 mrem) TEDE from the operation of Unit 1. This is well below the 10 CFR 20.1301 annual dose limit of 1 mSv (100 mrem) TEDE to a member of the general public. The applicant's maximum hourly dose estimate to a construction worker of 0.0285 µSv (2.85×10^{-3} mrem) in any hour is also well below the 10 CFR 20.1302 dose limit of 0.02 mSv (2 mrem) in any one hour to a member of the general public. Once Unit 1 is operational, Unit 1 personnel will conduct radiological surveys in the unrestricted and controlled area and radiological surveys for radioactive materials in effluent discharged to unrestricted and controlled areas to ensure that the annual doses to Unit 2 construction workers do not exceed the 10 CFR 20.1301 dose limits to members of the general public.

WLS COL FSAR Section 12.4.1.9.5, “Collective Doses to Lee Nuclear Station Unit 2 Workers,” identifies the collective annual exposure estimate for WLS Unit 2 construction workers. Using the estimated annual construction worker dose of 3.79 µSv (0.379 mrem) and a maximum estimated construction workforce of 2100 persons, the applicant calculated a total annual construction worker collective dose of 8.34 person-mSv (0.834 person-rem).

In summary, the staff independently evaluated the construction worker doses and concluded that the applicant’s dose assessment, as listed in WLS COL FSAR Table 12.4-201, meets regulatory requirements. Therefore, the staff concluded that the information provided in WSL COL FSAR Table 12.4-201, regarding the dose to construction workers, in WLS COL FSAR Section 12.4.1.9, is acceptable.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 12.4.4:

- *STD SUP 12.4-1*

The applicant provided supplemental information regarding conduct of radiological surveys in unrestricted and controlled areas and for radioactive materials in effluents discharged to unrestricted and controlled areas. The supplemental text states that these surveys are conducted by the operating unit for the purposes of implementing 10 CFR 20.1302 and to demonstrate compliance with the standards of 10 CFR 20.1301 for construction workers. This text is acceptable because it is consistent with applicable regulatory requirements. The staff confirmed that the VEGP COL FSAR was appropriately revised, and Open Item 12.4 1 is, therefore, closed.

A portion of the standard technical evaluation from the VEGP COL SER is not included above. The staff determined that the omitted portion was not relevant to WLS.

12.4.5 Post Combined License Activities

There are no post COL activities related to this section.

12.4.6 Conclusion

The staff reviewed the WLS COL application and checked the referenced DCD. The staff’s review confirmed that the applicant addressed the required information relating to dose assessment, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements. In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable based on the relevant acceptance criteria provided in NUREG-0800, Section 12.3-12.4. The staff based its conclusion on the following:

- WLS SUP 12.4-1, which provides supplemental information to address dose to construction workers, is acceptable because the applicant has demonstrated compliance with applicable requirements of 10 CFR 20.1101; 10 CFR 20.1301; 10 CFR 20.1302; and the applicable acceptance criteria provided in NUREG-0800, Section 12.3-12.4.

- STD SUP 12.4-1, which provides supplemental information regarding conduct of radiological surveys in unrestricted and controlled areas and for radioactive materials in effluents discharged to unrestricted and controlled areas, is acceptable because the applicant has demonstrated compliance with applicable requirements of 10 CFR 20.1301 and 10 CFR 20.1302.

12.5 Health Physics Facilities Design

12.5.1 Introduction

This section of the WLS COL FSAR addresses the objectives and design of the HP facilities. The HP facilities are designed to (1) provide the capability for administrative control of the activities of plant personnel to limit personnel exposure to radiation and radioactive materials ALARA and within the requirements of 10 CFR Part 20, and (2) provide the capability for administrative control of effluent releases from the plant to maintain the releases ALARA and within the limits of 10 CFR Part 20 and the plant Technical Specifications.

12.5.2 Summary of Application

WLS COL FSAR, Revision 11, Section 12.5 incorporates by reference AP1000 DCD, Revision 19, Section 12.5. In WLS COL FSAR Section 12.5, the applicant provided the following:

Departures

- WLS DEP 18.8-1

The applicant described the following Tier 2 departure from the AP1000 DCD. The AP1000 DCD states that the ALARA briefing room and Operations Support Center (OSC) share the same location in the Annex Building. The applicant proposed to move the OSC from the location identified in the AP1000 DCD to a location described in the Emergency Plan and revise AP1000 DCD Section 12.5.2.2 to exclude the reference to the OSC.

AP1000 COL Information Item

- STD COL 12.5-1

The applicant provided additional information in STD COL 12.5-1 to resolve COL Information Item 12.5-1 (COL Action Item 12.6-1), which addresses the RPP description.

License Conditions

- Part 10, License Conditions and ITAAC, 3, “Operational Program Implementation,” License Conditions C, “Receipt of Materials,” Item C.1, D, “Fuel Receipt,” Item D.2, G, “Fuel Loading,” Item G.4, and K, “Waste Shipment,” Item K.1
- Part 10, License Condition 6, “Operational Program Readiness”

The applicant proposed a license condition to provide a schedule to support NRC inspection of operational programs including the RPP.

12.5.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements. In addition, the acceptance criteria associated with the relevant requirements of NRC regulations for the HP facilities’ design are given in NUREG-0800, Section 12.5.

The applicable regulatory requirements and guidance for STD COL 12.5-1 are as follows:

- 10 CFR Part 20
- RG 8.2, Revision 0
- RG 8.4, “Direct Reading and Indirect Reading Pocket Dosimeters,” Revision 0
- RG 8.6, “Standard Test Procedures for Geiger-Muller Counters,” Revision 0
- RG 8.8, Revision 3
- RG 8.9, Revision 1
- RG 8.10, Revision 1-R
- RG 8.28, “Audible Alarm Dosimeters,” Revision 0
- NUREG-1736

The applicable regulatory requirement for License Condition 3, Items C.1, D.2, G.4, and K.1 is 10 CFR 20.1101.

12.5.4 Technical Evaluation

The staff reviewed WLS COL FSAR Section 12.5 and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The staff’s review confirmed that the information in the application and incorporated by reference addresses the required information relating to the HP facilities design. The results of the staff’s evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed and that those changes were actualized in the WLS COL FSAR.

- The staff verified that the site-specific differences, if any, did not adversely affect any previous relevant evaluation or conclusion.

The staff completed its review and finds the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this report by use of italicized, double-indented formatting. Section 1.2.3 of this report provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) contains evaluation material from the SER for the BLN Units 3 and 4 COL application.

Departures

- WLS DEP 18.8-1

Since the location of the WLS Units 1 and 2 OSC differs from the OSC location described in the AP1000 DCD, the applicant proposes to eliminate the reference to the OSC that appears in the first sentence of AP1000 DCD, Section 12.5.2.2. Therefore, the applicant proposes to revise the text in the first sentence of DCD Subsection 12.5.2.2 to read: "The ALARA briefing room is located off the main corridor immediately beyond the main entry to the annex building." This departure is acceptable insofar as the HP facility design is concerned because the location of the OSC does not have an impact on the radiation protection facilities design. The location of the ALARA briefing room remains as stated in the AP1000 DCD, so there is no impact on radiation protection facilities, programs or functions. The staff's evaluation of the effect of the OSC relocation on emergency preparedness is addressed in Section 13-3 of this report and on the human system interface design is addressed in Section 18.8 of this report.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 12.5.4:

> *The following portion of this technical evaluation section is reproduced from Section 12.5.4 of the BLN SER:*
>
> *AP1000 COL Information Item*
>
> - *STD COL 12.5-1*

The applicant provided additional information in STD COL 12.5-1, addressing the RPP description, to resolve COL Information Item 12.5-1. COL Information Item 12.5-1 states:

> *The Combined License applicant will address the organization and procedures used for adequate radiological protection and to provide methods so that personnel radiation exposures will be maintained ALARA.*

The same commitment was also captured as COL Action Item 12.6-1 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793). The applicant stated that STD COL 12.5-1 is addressed in Appendix 12AA of the BLN COL FSAR. This appendix incorporates by reference NEI 07-03, Revision 3. The applicant described revisions to NEI 07-03 and supplemental information in Appendix 12AA of the BLN COL FSAR. The staff evaluated the revised text and supplemental information provided in conjunction with the referenced NEI 07-03, Revision 3 template. These revisions and supplements address STD COL Items 12.1-1, 12.3-1, 12.3-3, 12.3-4, and 12.5-1. The applicant's proposed revisions and supplements are:

1. *Specific organizational positions were described in Chapter 13 of BLN COL FSAR; and Sections 12.5.2.1 through 12.5.2.5 are not incorporated in Appendix 12AA.*

2. *Facilities, as described in general terms in NEI 07-03, Revision 3 are not incorporated in BLN COL FSAR Appendix 12AA; facilities, instrumentation, and equipment are described in DCD Section 12.5.2.*

3. *Supplemental information was provided for NEI 07-03, Section 12.5.3.3 to describe compliance with 10 CFR 20.1703(b) and 10 CFR 20.1705 when National Institute for Occupational Safety and Health (U.S. Public Health Service) tested and certified respiratory protection equipment is not used.*

4. *The following headings and associated material that are described in general terms in NEI 07-03, Revision 3 are not incorporated in Appendix 12AA. Radwaste Handling, Spent Fuel Handling, Normal Operation, and Sampling are described in DCD Section 12.5.3.*

5. *Supplemental information was provided for NEI 07-03, Section 12.5.4.4* [sic] *to describe the use of a closed circuit television system to allow remote monitoring for high radiation areas access.*

6. *Supplemental information was provided for NEI 07-03, Section 12.5.4.4 to describe access control measures for very high radiation areas. Locations and radiological controls of the radiation zones are described on plant diagrams in DCD Section 12.5.3.*

7. *Appendix 12AA revised NEI 07-03, Section 12.5.4.7 to clarify the location of the COL applicant's management policy, organizational responsibility*

authorities for implementing an effective ALARA program, and the establishment and implementation of radiation protection.

8. *The applicant revised the second bullet of NEI 07-03, Section 12.5.4.7 II to require that the functional manager in charge of radiation protection be responsible for defining the value for "Significant exposures" and the associated activities within written procedures. The example value described in NEI 07-03 includes activities that are estimated to involve greater than 1 person-rem of collective dose.*

9. *The COL applicant added text after the last bullet of NEI 07-03, Section 12.5.4.8 to adopt NEI 08-08 that is currently under review by the NRC staff.*

10. *The COL applicant added information to NEI 07-03, Section 12AA.5.4.14 and Section 12AA.5.4.15* [sic] *to adopt NEI 08-08 that is currently under review by the NRC staff.*

The applicant describes the exceptions and supplemental information to NEI 07-03 that reference additional design and site-specific information necessary to clearly identify the source of the information addressed in the RPP as described in Appendix 12AA. The applicant's description provides sufficient detailed information supporting the exceptions or revisions such that the information described provides clear direction as to organizational structure, facilities, management policy for ALARA, and where the threshold for significant with exposures will be described. The NRC staff agrees that the applicant's exceptions to NEI 07-03, noted above are acceptable because these exceptions and the supplemental information satisfy the regulatory requirements of 10 CFR 20.1106 (b), the acceptance criteria of Sections 12.1 and 12.5 of NUREG-0800 and the regulatory guidance in RG 8.8, Position C.1.b, RG 8.9, and RG 8.10, Positions C.1.a, and C.2.

The applicant added Appendix 12AA, "Appendix 12AA, Radiation Protection Program Description," after Section 12.5 of the DCD. In this appendix the applicant incorporates by reference NEI 07-03, Revision 3. The applicant indicated that Table 13.4-201 provides milestones for radiation protection operational program implementation.

The NRC staff reviewed STD COL 12.5-1 dealing with the RPP description in BLN COL FSAR Appendix 12AA. The additional controls described in STD COL 12.5-1 are consistent with the discussion in NUREG-1736 regarding Bioassay programs for personnel monitoring and are consistent with the applicant's commitment to RG 8.9. The staff reviewed the threshold for determining significant exposures. The applicant stated that the functional manager in charge of radiation protection determines the threshold within procedures. Initially, the staff did not consider that the applicant exercised sufficient control related to maintaining ALARA (RAI 12.5-1).

In response to RAI 12.5-1, in a letter dated September 22, 2008, the applicant provided additional information that the final NEI 07-03 template (Revision 7) would be incorporated without departure concerning significant exposures. In a letter dated March 18, 2009 (ML090510379), the NRC accepted NEI 07-03, Revision 7. Specifically, the NRC staff indicated that for COL applications, NEI 07-03, Revision 7 provides an acceptable template for assuring that the RPP meets the applicable regulations and guidance. Since the BLN COL FSAR has not yet adopted the approved version of the NEI template, this is identified as Confirmatory Item 12.1-1.

The NRC staff reviewed Revision 0 of the BLN COL FSAR Appendix 1AA, which listed the applicant's conformance with radiation protection related RGs. The applicant stated that it will conform in general to RG 8.28, "Audible Alarm Dosimeters," Revision 0, dated August 1981, and specifically stated that it conforms to ANSI N13.7-1981, which was reaffirmed in 1992. ANSI N13.7-1983 is the "American National Standard for Radiation Protection-Photographic Film Dosimeters Criteria for Performance." RG 8.28, Revision 0, endorsed ANSI N13.27-1981, "Performance Specifications for Pocket-Sized Alarming Dosimeters/Ratemeters." This discrepancy was identified in RAI 1-10. In response to RAI 1-10, the applicant stated that BLN COL FSAR Appendix 1AA would be revised to the correct reference of the ANSI standard in a future revision of the BLN COL FSAR. The NRC staff verified that Revision 1 of the BLN COL FSAR adequately addresses the proposed change. As a result, RAI 1-10 is closed.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 12.5.4:

The staff notes that the VEGP FSAR has not been updated to correct the discrepancy identified in RAI 1-10 regarding the reference to ANSI N13.27-1981. Revision 2 of the VEGP FSAR currently references the incorrect standard, ANSI N13.7-1981, under RG 8.28 in Appendix 1AA. Since the VEGP applicant has endorsed RAI 1-10, the staff expects this discrepancy to be corrected in a future revision of the VEGP FSAR. This is VEGP Confirmatory Item 12.5-2.

Correction of Error in the Standard Content Evaluation Text

The NRC staff identified two errors in the text reproduced above from the BLN SER, Section 12.5.4 that require correction. In the change numbered 5 above, the reference to "NEI 07-03, Section 12.5.4.4," is incorrect. The correct reference is to "NEI 07-03, Section 12.5.4.2." In the change numbered 10, above, the reference to "Section 12AA.5.4.14 and Section 12AA.5.4.15" is incorrect. The correct reference is to "Section 12.5.4.14 and Section 12.5.4.15."

Resolution of Standard Content Confirmatory Item 12.1-1

The NRC staff compared the VEGP and BLN COL applications regarding STD COL 12.5-1, and found them to be essentially identical, with the exception that VEGP FSAR Appendix 12AA references NEI 07-03A and BLN FSAR

Appendix 12AA references Revision 3 of NEI 07-03. Additional clarifying information has been added to the VEGP FSAR regarding STD COL 12.5-1, which is discussed below. As indicated in Section 12.1.4 above, Confirmatory Item 12.1-1, is resolved for VEGP because the applicant has adopted the approved version of NEI 07-03, which is now designated as NEI 07-03A.

In Revision 2 of the FSAR, the applicant modified parts of FSAR Chapter 12, Appendix 12AA, that relate to STD COL 12.5-1. The changes are as follows:

1. *Text describing a closed circuit television system associated with high radiation areas has been moved from Appendix 12AA to Section 12.5.2.2 (this text is associated with STD COL 12.3-1, and is evaluated in Section 12.3.4 of this SER).*

2. *References in NEI 07-03A have been revised to reflect the appropriate sections of the FSAR.*

3. *Proposed modifications to the second bullet of NEI 07-03, Section 12.5.4.7 have been withdrawn.*

4. *Bullet number 3 of NEI 07-03A, Section 12.5, has been revised to address aspects of the radiation program functional areas that must be in place at various milestones.*

5. *A cross reference to NEI 08-08A has been added in NEI 07-03A.*

6. *The first paragraph of Section 12.5.4.12 of NEI 07-03A has been revised to address 10 CFR 20.1101 and the Quality Assurance Program.*

Items 1, 2, and 5 are acceptable because they are editorial and do not affect content. The change described in Item 3 is acceptable because NEI 07-03A is acceptable without modification. The changes described in Item 4 are acceptable because they are consistent with the milestones described in FSAR Table 13.4-201 and with applicable regulatory requirements. The changes described in Item 6 are acceptable because they are consistent with 10 CFR 20.1101 and the Quality Assurance Program described in FSAR Section 17.5.

Resolution of VEGP Confirmatory Item 12.5-2

Appendix 1AA of the WLS COL FSAR correctly references ANSI N13.27-1981 under the conformance discussion of RG 8.28. Therefore, VEGP Confirmatory Item 12.5-2 is resolved for WLS COL application.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 12.5.4:

<u>Exceptions to RGs 8.2, 8.4, 8.6, and Section C.3.b of RG 8.8</u>

The following portion of this technical evaluation section is reproduced from Section 12.5.4 of the BLN SER.

The applicant took exception to RG 8.2, "Guide for Administrative Practices in Radiation Monitoring," regarding a reference to a previous version of 10 CFR Part 20 (10 CFR 20.401), because it is no longer valid. The staff agrees with the applicant's exception.

The applicant took exception to RG 8.4, "Direct Reading and Indirect Reading Pocket Dosimeters," regarding references to previous versions of 10 CFR Part 20 (10 CFR 20.202(a), and 10 CFR 20.401) because they are no longer valid. The staff agrees with the applicant's exception. The applicant also took exception to ANSI N13.5-1972 (R-1989), in that two performance criteria, accuracy and leakage, specified in the guidance, are to be met by acceptance standards in ANSI N322-1997, "ANSI Test, Construction, and Performance requirements for Direct Reading Electrostatic/Electroscope Type Dosimeters." The staff finds that by using ANSI N322-1997 for performance criteria, 10 CFR 20 requirements are still met, as the major change is the allowance of an additional one percent leakage over a comparable time period. Test and calibration intervals recommended by RG 8.4 are not affected.

The applicant took exception to RG 8.6, "Standard Test Procedures for Geiger Mueller Counters," to reference an instrument calibration program based upon ANSI Criteria N323A-1997 (with 2004 Correction Sheet), "Radiation Protection Instrumentation Test and Calibration, Portable Survey Instruments." This methodology is acceptable over the previous program referenced in RG 8.6 because the ANSI standard reflects current industry practices. The staff agrees with the applicant's position.

The applicant took exception to part of Position C.3.b in RG 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposure at Nuclear Power Stations will be ALARA." This exception was to the reporting requirements associated with operating exposure. The applicant's basis for justifying the exception to RG 8.8, Position C.3.b, is that reporting of operating exposure information is no longer required. The staff agrees with the applicant's exception to RG 8.8, Position C3.b, because this specific reporting requirement has been superseded. All licensees are now required to report records of ionizing exposure to the NRC annually in accordance with 10 CFR 20.2206.

<u>License Condition</u>

- *License Condition 3, Items C.1, D.2, G.4, and K.1*

Implementation milestones were provided by the applicant to address the RPP required by 10 CFR 20.1101. A phased-in implementation should include appropriate milestones in the construction of the facility. Staffing levels, equipment, facilities, and procedures necessary to ensure radiation safety of the workers and public for each phase of implementation should be identified. In

RAI 12.5-2, the staff requested that the applicant provide the specific programs to be implemented at each milestone identified in Table 13.4-201 of the BLN COL FSAR. In its response to the RAI, the applicant provided clarifying information regarding Table 13.4-201.

In a supplemental response to RAI 12.5-2, dated December 16, 2008, the applicant provided a proposed revision to BLN COL FSAR Table 13.4-201 to show the specific program(s) for each milestone and assignment of a Radiation Protection Manager and Supervisor. The proposed change to BLN COL FSAR Table 13.4-201 is acceptable subject to a formal revision to the BLN COL FSAR, based on the specific commitment to establish an individual responsible for each milestone. Accordingly, this is identified as Confirmatory Item 12.5-1.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 12.5.4:

<u>Resolution of Standard Content Confirmatory Item 12.5-1</u>

The NRC staff verified that the VEGP FSAR was updated to include the information identified in the initial and supplemental BLN response to RAI 12.5-2. Accordingly, Standard Content Confirmatory Item 12.5-1 is resolved for the VEGP COL FSAR.

<u>Resolution of Standard Content Confirmatory Item 12.5-1</u>

The NRC staff verified that the WLS FSAR includes the information identified in the initial and supplemental BLN response to RAI 12.5-2. Accordingly, the standard content Confirmatory Item 12.5-1 is resolved for the WLS COL FSAR.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 12.5.4:

- *Part 10, License Condition 6, Operational Program Readiness*

The applicant proposed a license condition to provide a schedule to support NRC inspection of operational programs, including the RPP. The proposed license condition is consistent with the policy established in SECY-05-0197, "Review of Operational Programs in a Combined License Application and General Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," and is acceptable.

12.5.5 Post Combined License Activities

The license condition language in this section has been clarified from previously considered language. In a letter dated March 22, 2016 (ADAMS Accession No. ML16084A099), the applicant did not identify any concerns with the clarified license condition language. The changes do not affect the staff's above analysis of the conditions, and therefore, for the reasons discussed in the technical evaluation section above, the staff finds the following license conditions acceptable:

- License Condition (12-1) – The licensee shall implement the Radiation Protection Program (RPP) (including the ALARA principle) or applicable portions thereof (as identified in FSAR Section 12.5) as described in the milestones below:

  1. RPP features (including the ALARA principle) applicable to receipt of by-product, source, or special nuclear materials (excluding exempt quantities as described in 10 CFR 30.18) implemented before initial receipt of such materials;

  2. RPP features (including the ALARA principle) applicable to new fuel implemented before receipt of initial fuel on site;

  3. All other RPP features (including the ALARA principle) except for those applicable to control radioactive waste shipment implemented before initial fuel load;

  4. RPP features (including the ALARA principle) applicable to radioactive waste shipment implemented before first shipment of radioactive waste;

- License Condition (12-2) – No later than 12 months after issuance of the COL, the licensee shall submit to the Director of the Office of New Reactors a schedule that supports planning for and conduct of NRC inspections of the operational program (RPP). The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until this operational program has been fully implemented.

12.5.6 Conclusion

The staff reviewed the WLS COL application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to the radiation protection design features, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable based on the relevant acceptance criteria provided in NUREG-0800, Section 12.5. The staff based its conclusion on the following:

- WLS DEP 18.8-1, in which the applicant proposed to relocate the OSC from the location described in the AP1000 DCD Section 12.5.2.2, is acceptable insofar as the HP facility design is concerned because the location of the OSC does not have an impact on the radiation protection facilities design. The ALARA briefing room remains as stated in the AP1000 DCD, so there is no impact on radiation protection facilities, programs or functions.

- STD COL 12.5-1, which addresses the RPP description, is acceptable because the applicant incorporates NEI 07-03A into the WLS COL FSAR in order to meet the applicable regulatory requirements and guidance specified in Sections 12.5.3 and 12.5.4 of this report.

13 CONDUCT OF OPERATIONS

13.1 Organizational Structure of Applicant

13.1.1 Introduction

Duke Energy Carolinas', LLC. (Duke) (the applicant) organizational structure includes the design, construction, and preoperational responsibilities of the organizational structure. The management and technical support organization includes a description of the corporate or home office organization, its functions and responsibilities, and the number and the qualifications of personnel. The applicant's organizational structure activities include facility design, design review, design approval, construction management, testing, and operation of the plant. The descriptions of the design and construction and preoperational responsibilities include the following:

- how these responsibilities are assigned by the headquarters staff and implemented within the organizational units
- the responsible working- or performance-level organizational unit
- the estimated number of persons to be assigned to each unit with responsibility for the project
- the general educational and experience requirements for identified positions or classes of positions
- early plans for providing technical support for the operation of the facility

This section of the of the William States Lee III Nuclear Station (WLS) combined license (COL) Final Safety Analysis Report (FSAR) also describes the structure, functions, and responsibilities of the onsite organization established to operate and maintain the plant.

13.1.2 Summary of Application

WLS COL FSAR, Revision 11, Section 13.1 incorporates by reference AP1000 Design Control Document (DCD), Revision 19, Section 13.1.

In addition, in WLS COL FSAR Section 13.1, the applicant provided the following:

AP1000 COL Information Items

- WLS COL 13.1-1

The applicant provided additional information in WLS COL 13.1-1 to resolve COL Information Item 13.1-1 (COL Action Item 13.1-1). COL Information Item 13.1-1 requires the COL applicant to describe its organizational structure. WLS COL 13.1-1 describes organizational positions of the nuclear power station and owner/applicant corporations and associated functions and responsibilities. WLS COL FSAR Table 1.8-202, “COL Item Tabulation,” provides WLS COL 13.1-1 cross-references.

- WLS COL 9.5-1

The applicant provided additional information in WLS COL 9.5-1, describing the fire protection program in WLS COL FSAR Section 9.5.1.8. For this WLS COL item, the applicant added a new WLS COL FSAR Section 13.1.1.2.10, “Fire Protection,” and a new WLS COL FSAR Section 13.1.2.1.2.9, “Engineer - Fire Protection.” WLS COL FSAR Table 1.8-202, “COL Item Tabulation,” provides WLS COL 9.5-1 cross-references.

- WLS COL 18.6-1

The applicant provided additional information in WLS COL 18.6-1, describing the qualifications of the nuclear plant technical support personnel. WLS COL 18.6-1 is addressed under WLS COL FSAR Section 13.1.1.4, “Qualifications of Technical Support Personnel,”; WLS COL FSAR Section 13.1.3.1, “Qualification of Nuclear Plan Personnel.”; WLS COL FSAR Table 13.1-201 “Generic Position/Site Specific Position Cross Reference,”; and WLS COL FSAR Table 13.1-202 “Minimum On-Duty Operations Shift Organization For Two-Unit Plant”; and WLS COL FSAR Table 1.8-202, “COL Item Tabulation,” provides WLS COL 18.6-1 cross-references.

- WLS COL 18.10-1

The applicant provided additional information in WLS COL 18.10-1 to address the responsibilities of the manager in charge of nuclear training. WLS COL 18.10-1 is addressed in WLS COL FSAR Section 13.1.1.3.2.4, “Functional Manager – Training and Development” WLS COL FSAR Table 1.8-202, “COL Item Tabulation,” provides WLS COL 18.10-1 cross-references.

13.1.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design,” and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of NRC regulations for WLS COL 13.1-1, WLS COL 9.5-1, WLS COL 18.6-1, and WLS COL 18.10-1 are given in WLS COL FSAR Sections 13.1.1, “Management and Technical Support Organization,” and 13.1.2-13.1.3, “Operating Organization,” of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition.”

The applicable regulatory guidance for the organizational structure of the applicant is as follows:

- American National Standards Institute (ANSI)/American Nuclear Society (ANS)-3.1-1993, “American National Standard for Selection, Qualification, and Training of Personnel for Nuclear Power Plants,” as endorsed and amended by Regulatory Guide (RG) 1.8, Revision 3, “Qualification and Training of Personnel for Nuclear Power Plants.”

The applicable regulations and regulatory guidance for the management, technical support, and operating organizations of the applicant are as follows:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34, “Contents of applications; technical information”
- 10 CFR 50.40, “Common standards”
- 10 CFR 52.47, “Contents of applications; technical information”
- 10 CFR 50.48, “Fire Protection”
- 10 CFR 50.50 Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants”
- 10 CFR 50.54, “Conditions of licenses”
- 10 CFR Part 55, “Operator's Licenses”
- 10 CFR 50.71, “Maintenance of records, making of reports”
- 10 CFR 52.79, “Contents of applications; technical information in final safety analysis report”
- RG 1.33, Revision 2, “Quality Assurance Program Requirements (Operation)”
- RG 1.8, “Qualification and Training of Personnel for Nuclear Power Plants”
- RG 1.28, “Quality Assurance Program Criteria (Design and Construction)”
- RG 1.33, “Quality Assurance Program Requirements (Operation)”
- RG 1.68, “Initial Test Programs for Water-cooled Nuclear Power Plants”
- RG 1.114, AGuidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit”
- RG 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants”
- RG 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis.”
- RG 1.175, “An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing”
- RG 1.177, “An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications”
- RG 1.178, “An Approach for Plant-Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping”
- RG 1.182, “Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants”

- RG 1.206 “Combined License Applications for Nuclear Power Plants (LWR Edition)”
- NUREG-0660, “NRC Action Plan Developed as a Result of the TMI-1 Accident”
- NUREG-0694, “TMI-Related Requirements for New Operating Licenses”
- NUREG-0711,“Human Factors Engineering Program Review Model”
- NUREG-0718, “Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License”
- NUREG-0737 and Supplement 1, “A Clarification of TMI Action Plan Requirements”

13.1.4 Technical Evaluation

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed WLS COL FSAR Section 13.1 and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff’s (the staff) review confirmed that the information in the application and incorporated by reference addresses the required information relating to the organizational structure of the applicant. The results of the staff’s evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the WLS COL FSAR:

AP1000 COL Information Items

- WLS COL 13.1-1

The staff reviewed WLS COL 13.1-1 related to the organizational structure of the COL applicant included under WLS COL FSAR Section 13.1. WLS COL FSAR Section 13.1 describes the organizational positions of a nuclear power plant and owner/applicant corporations and associated functions and responsibilities.

The applicant provided the following additional WLS site-specific COL information to resolve COL Information Item 13.1-1, which addresses the organizational structure of the COL applicant. COL Information Item 13.1-1 states:

> Combined License applicants referencing the AP1000 certified design will address adequacy of the organizational structure.

The commitment was also captured as COL Action Item 13.1-1 in NUREG-1793, Appendix F, which states:

> The COL applicant will describe its organizational structure.

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[1] See Section 1.2.2 of this report for a discussion of the staff’s review related to verification of the scope of information to be included in a COL application that references a design certification (DC).

The applicant provided additional information as part of the WLS COL FSAR to describe the organizational positions of a nuclear power station and owner/applicant corporations and associated functions and responsibilities. The position titles used in the text are generic and describe the function of the position. The applicant stated that WLS COL FSAR Table 13.1-201, "Generic Position/Site-Specific Position Cross-Reference" provides a cross-reference to identify site-specific position titles.

The applicant added new sections and information related to the site-specific organizational structure to WLS COL FSAR Section 13.1 beyond the structure given in RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)." The new section titles are:

Section 13.1.1, "Management and Technical Support Organization"

Section 13.1.2, "Operating Organization"

Section 13.1.3, "Qualifications of Nuclear Plant Personnel"

Section 13.1.4, "Combined License Information Item"

Section 13.1.5, "References"

Table 13.1-201, "Generic Position/Site-Specific Position Cross-Reference"

Table 13.1-202, "Minimum On-Duty Operations Shift Organization for Two-Unit Plant"

Figure 13.1-201, "Plant Management Organization"

Section 13.1-202, "Shift Operations Organization"

Section 13.1-203, "Nuclear Executive Organization"

Section 13.1-204, "Duke Energy Corporate"

Section 13AA-201, "Construction Management Organization"

Section 13AA-202 "Hiring Schedule for Plant Staff"

In addition, the applicant added a new appendix to Chapter 13 titled, "Appendix 13AA, Design and Construction." This appendix describes the applicant's construction organization. Once plant operation commences, this appendix will become historical.

The staff reviewed WLS COL 13.1-1 and concludes that the management, technical support, and operating organizations, as described are acceptable and meet the requirements of 10 CFR 50.40(b) based on the following.

The applicant described its organization for the management of, and its means of providing, technical support for the plant staff for the design, construction, and operation of the facility and described its plans to manage the project and utilize the nuclear steam system supplier (NSSS) vendor and architect-engineer (AE). These plans provide reasonable assurance that the applicant will establish an acceptable organization and that sufficient resources are available to provide offsite technical support and to satisfy the applicant's commitments for the design, construction, and operation of the facility.

The applicant described the assignment of plant operating responsibilities; the reporting chain up through the chief executive officer; the functions and responsibilities of each major plant staff group; the proposed shift crew complement for single-unit or multiple-unit operation; the qualification requirements for members of its plant staff; and staff qualifications. In WLS COL FSAR Table 1.9-202, "Conformance with SRP Acceptance Criteria," the applicant noted an exception to the criteria of NUREG-0800, Section 13.1.2-13.1.3 that suggests resumes of personnel holding plant managerial and supervisory positions be included in the WLS COL FSAR. The staff finds this exception to the criteria of NUREG-0800, Section 13.1.2-13.1.3 acceptable because resumes for management and principal supervisory and technical positions will be available for review after position vacancies are filled.

NUREG-0800, Section 13.1.2-13.1.3, "Operating Organization," provides the following acceptable characteristics for an applicant's operating organization:

1. The applicant is technically qualified, as specified in 10 CFR 50.40(b).

2. An adequate number of licensed operators will be available at all required times to satisfy the minimum staffing requirements of 10 CFR 50.54(j).

3. On-shift personnel are able to provide initial facility response in the event of an emergency.

4. Organizational requirements for the plant manager and radiation protection manager have been satisfied.

5. Qualification requirements and qualifications of plant personnel conform to the guidance of RG 1.8.

6. Organizational requirements conform to the guidance of RG 1.33.

The staff finds that the operating organization proposed by the applicant will comply with these characteristics. These findings contribute to the judgment that the applicant complies with the requirements of 10 CFR 50.40(b). That is, the applicant is technically qualified to engage in design and construction activities and to operate a nuclear power plant; that the applicant will have the necessary managerial and technical resources to support the plant staff in the event of an emergency; and that the applicant has identified the organizational positions responsible for fire protection matters and delegated the authorities to these positions to implement fire protection requirements as discussed under WLS COL 9.5-1 below.

- WLS COL 9.5-1

The applicant added text to WLS COL FSAR Section 13.1.1.2.10, "Fire Protection," indicating that the nuclear power station is committed to maintaining a fire protection program as described in WLS COL FSAR Section 9.5.1.8, and that the site executive in charge of plant management, through the engineer in charge of fire protection, is responsible for the fire protection program. The applicant added text to WLS COL FSAR Section 13.1.2.1.2.9, “Engineer - Fire Protection,” describing the responsibilities of the engineer in charge of the fire protection program.

The staff reviewed WLS COL 9.5-1 relative to the text added to Sections 13.1.1.2.10 and 13.1.2.1.2.9 of the WLS COL application. Based on the management descriptions provided in WLS COL FSAR Sections 13.1.1.2.10 and 13.1.1.3.2.1.4, the staff finds the applicant’s fire protection organization meets the guidance of NUREG-0800. The technical review for WLS COL 9.5-1, as it relates to the programmatic requirements, is addressed in Section 9.5.1.8 of this report.

- WLS COL 18.6-1

The staff reviewed WLS COL 18.6-1, which describes the qualifications of the nuclear plant technical support personnel.

In WLS COL FSAR Table 1.9-202, “Conformance with SRP Acceptance Criteria,” the applicant noted an exception to the criteria of NUREG-0800, Section 13.1.1. The SRP acceptance criteria suggest that the experience requirements of managers and supervisors of the technical support organization are to be included in the WLS COL FSAR. The staff finds this exception to the criteria of NUREG-0800, Section 13.1.1 acceptable because the applicant added text to WLS COL FSAR Section 13.1.1.4, “Qualifications of Technical Support Personnel,” stating the qualifications of managers and supervisors of the technical support organization will meet the education and experience requirements described in ANSI/ANS-3.1-1993 and RG 1.8.

The applicant added text to WLS COL FSAR Section 13.1.3, “Qualification of Nuclear Plant Personnel,” stating, in WLS COL FSAR Section 13.1.3.1, the qualifications of managers, supervisors, operators, and technicians of the operating organization will meet the education and experience requirements described in ANSI/ANS-3.1-1993 and RG 1.8. In addition, WLS COL FSAR Section 13.1.3.2 states that resumes and other documentation of the qualifications and experience of initial appointees to appropriate management and supervisory positions will be available for review after position vacancies are filled.

The applicant added WLS COL FSAR Table 13.1-202, “Minimum On-Duty Operations Shift Organization for Two-Unit Plant.” WLS COL FSAR Table 13.1-202 describes the minimum composition of the operating shift crew for all modes of operation. Position titles, license requirements and minimum shift manning for the various modes of operation are addressed in Technical Specifications and will be addressed in administrative procedures.

The staff reviewed the text added to WLS COL FSAR Sections 13.1.1.4 and 13.1.3.1 relative to WLS COL 18.6-1 and concludes that the qualification requirements are acceptable and meet the requirements of 10 CFR 50.40(b) based on the following.

The applicant described its organization for the management of, and its means of providing, technical support for the plant staff for the design, construction, and operation of the facility and described its plans for managing the project and utilizing the NSSS vendor and AE. These plans give reasonable assurance that the applicant will establish an acceptable organization and that sufficient resources are available to provide offsite technical support and to satisfy the applicant's commitments for the design, construction, and operation of the facility.

- WLS COL 18.10-1

The staff reviewed WLS COL 18.10-1 included under WLS COL FSAR Section 13.1.1.3.2.4, “Functional Manager – Training and Development.” This section describes the responsibilities of the manager in charge of nuclear training relative to the site training programs required for the safe and proper operation and maintenance of the plant. This item is cross-referenced to WLS COL FSAR Section 18.10 in WLS COL FSAR Table 1.8-202, “COL Item Tabulation.” The staff concludes that the qualification requirements are acceptable and meet the requirements of 10 CFR 50.40(b) and the regulatory guidelines in NUREG-0800, Sections 13.1.1 and 13.1.2-13.1.3 because the applicant described how the training manager will carry out his or her position responsibilities for designing, developing, implementing, and maintaining training programs for the safe and proper operation and maintenance of the plant.

13.1.5 Post Combined License Activities

There are no post COL activities related to this section.

13.1.6 Conclusion

The staff reviewed the application and checked the referenced DCD. The staff’s review confirmed that the applicant addressed the required information relating to the organizational structure of the applicant, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The applicant described clear responsibilities and definite resources for the design and construction of the facility and has described its plans for managing the project and utilizing the Nuclear Steam Supply System (NSSS) vendor and architect engineer (AE). The staff reviewed these plans and determined that they provide adequate assurance that an acceptable organization has been established and that sufficient resources are available to satisfy the applicant’s commitments for the design and construction of the facility. These findings contribute to the judgment that the applicant complies with the requirements of 10 CFR 50.34, 10 CFR 50.40, 10 CFR 50.48, 10 CFR Part 50 Appendix B, 10 CFR 52.47, 10 CFR 52.79, and 10 CFR 52.80, as applicable; that is, the applicant is technically qualified to engage in design and construction activities.

The applicant described its organization for the management of, and its means of providing, technical support for the plant staff during operation of the facility. These measures have been reviewed and the staff finds that the applicant has an acceptable organization and adequate resources to provide offsite technical support for the operation of the facility under both normal and off-normal conditions.

The applicant described the assignment of plant operating responsibilities; the reporting chain up through the chief executive office of the applicant; the proposed size of the regular plant staff; the functions and responsibilities of each major plant staff group; the proposed shift crew complement for single-unit or multiple-unit operation; the qualification requirements for members of its plant staff; and plant staff qualifications (through personnel resumes for management and principle supervisory and technical positions as submitted during the later stages of plant design, construction, and licensing).

The staff finds that the operating organization proposed by the applicant is acceptable because it meets the requirements of 10 CFR 50.40(b), as applicable. That is, the applicant is technically qualified to operate a nuclear power plant; and will have the necessary managerial and technical resources to support the plant staff in the event of an emergency and has identified the organizational positions responsible for fire protection matters and delegated the authorities to these positions to implement fire protection requirements.

In addition, the staff concludes that the information presented in the WLS COL FSAR is acceptable because it meets the acceptance criteria provided in NUREG-0800, Section 13.1. The staff based its conclusion on the following:

- WLS COL 13.1-1, as it relates to the organizational structure of the COL applicant, is acceptable because it meets the requirements of 10 CFR 50.40(b).

- WLS COL 9.5-1, as it relates to the fire protection organization meets the guidance of NUREG-0800, Section 13.1 and is acceptable.

- WLS COL 18.6-1, as it relates to the qualifications of nuclear plant technical support personnel, is acceptable because it meets the requirements of 10 CFR 50.40(b).

- WLS COL 18.10-1, as it relates to the qualification requirements for the manager in charge of nuclear training, is acceptable because it meets the requirements of 10 CFR 50.40(b).

13.2 Training

13.2.1 Introduction

This section of the WLS COL FSAR addresses the description and schedule of the training program for reactor operators (ROs) and senior reactor operators (SROs) (i.e., licensed operators). This section of the WLS COL FSAR addresses the scope of licensing examinations as well as training requirements. The licensed operator training program also includes the requalification programs as required in 10 CFR 50.54(i) (i-1) and 10 CFR 55.59, “Requalification.” In addition, this section of the WLS COL FSAR includes the description and schedule of the training program for non-licensed plant staff.

13.2.2 Summary of Application

WLS COL FSAR, Revision 11, Section 13.2, incorporates by reference AP1000 DCD, Revision 19, Section 13.2.

In addition, in WLS COL FSAR Section 13.2, the applicant provided the following:

AP1000 COL Information Items

- STD COL 13.2-1

The applicant provided additional information in Standard (STD) COL 13.2-1 to resolve COL Information Item 13.2-1 (COL Action Item 13.2-1), which incorporates the provisions of Nuclear Energy Institute (NEI) 06-13A, “Template for an Industry Training Program Description,” providing the description and scheduling of the training program for plant personnel, including the requalification program for licensed operators.

- STD COL 18.10-1

The applicant provided additional information in STD COL 18.10-1 to address training for those operators involved in the Human Factors Engineering (HFE) Verification and Validation Program, using a systematic approach to training and Westinghouse Commercial Atomic Power (WCAP)-14655, “Designer’s Input to the Training of the Human Factors Engineering Verification and Validation Personnel.”

License Conditions

- Part 10, License Condition 3, Items B.1, C.3

The applicant proposed a license condition in Part 10 of the WLS COL application, which provides the milestones for implementing the Reactor Operator Training (B.1) and the applicable portions of the Non-Licensed Plant Staff Training Program (C.3) related to radioactive material required in accordance with 10 CFR 50.120, “Training and Qualification of Nuclear Power Plant Personnel.” The license condition related to the portions of the Non-Licensed Plant Staff Training Program applicable to radioactive material is addressed in Chapter 1 of this report.

- Part 10, License Condition 6

The applicant proposed a license condition to provide a schedule to support the NRC’s inspection of operational programs included in WLS COL FSAR Table 13.4-201, including the Non-Licensed Plant Staff Training Program, required in accordance with 10 CFR 50.120, Reactor Operator Training Program, and the Reactor Operator Requalification Program.

13.2.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of NRC regulations for the description and schedule of the training program for licensed operators are given in WLS COL FSAR Sections 13.2.1 and 13.2.2 and in NUREG-0800, Chapter 18.

The applicable regulations and regulatory guidance documents for STD COL 13.2-1 are as follows:

- 10 CFR 50.54(m)

- 10 CFR Part 55, “Operators’ licenses”
- RG 1.8
- RG 1.149, “Nuclear Power Plant Simulation Facilities for Use in Operator Training and License Examinations”
- NUREG-1021, “Operator Licensing Examination Standards for Power Reactors”

The applicable regulations for the Non-Licensed Plant Staff Training Program are as follows:

- 10 CFR 50.120
- 10 CFR 52.79(a)(33), “Contents of applications; technical information”

The applicable regulations for the licensed operators training program are as follows:

- 10 CFR 55.13, “General exemptions”
- 10 CFR 55.31, “How to apply”
- 10 CFR 55.41, “Written examinations: Operators”
- 10 CFR 55.43, “Written examinations: Senior operators”
- 10 CFR 55.45, “Operating tests”

The applicable regulations for the licensed operator’s requalification program are found in the following:

- 10 CFR 50.34(b), “Final safety analysis report”
- 10 CFR 50.54(i)
- 10 CFR 55.59

The applicable regulatory guidance for STD COL 18.10-1 is as follows:

- NUREG-0711, “Human Factors Engineering Program Review Model”

13.2.4 Technical Evaluation

The staff reviewed WLS COL FSAR Section 13.2 and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The staff’s review confirmed that the information in the application and incorporated by reference addresses the required information relating to the description and schedule of the training programs for nuclear plant personnel. The results of the staff’s evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the design certification (DC) and use this review to evaluate subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (Vogtle Electric Generating Plant (VEGP), Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews.

- The staff compared the VEGP COL FSAR, Revision 5 to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from responses to requests for additional information (RAIs).

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this report by use of italicized, double-indented formatting. Section 1.2.3 of this report provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the Bellefonte Nuclear Station (BLN) Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.2.4:

> <u>*AP1000 COL Information Items*</u>
>
> - *STD COL 13.2-1*
>
> *The NRC staff reviewed STD COL 13.2-1 related to COL Information Item 13.2-1 (COL Action Item 13.2-1) included under Section 13.2 of the BLN COL FSAR. COL Information Item 13.2-1 states:*
>
>> *The Combined License applicants referencing the AP1000 certified design will develop and implement training programs for plant personnel. This includes the training program for the operations personnel who participate as subjects in the human factors engineering verification and validation. These Combined License applicant training programs will address the scope of licensing examinations as well as new training requirements.*
>
> *The commitment was also captured as COL Action Item 13.2-1 in Appendix F of the NRC staff FSER for the AP1000 DCD (NUREG-1793), which states:*
>
>> *The COL applicant will develop and implement training programs for plant personnel.*

The applicant provided the following text to supplement Section 13.2, "Training," of the AP1000 DCD, dealing with the training program for plant personnel.

> *This section incorporates by reference NEI 06-13* (sic) [NEI 06-13A], *Template for an Industry Training Program Description. See Table 1.6-201.*

This technical report provides a complete training program description for use with COL applications. The staff has endorsed NEI 06-13A, Revision 1, as it provides an acceptable template for describing licensed operators and non-licensed plant staff training programs. The applicant has incorporated by reference NEI 06-13A, Revision 1.

The applicant provided the following text to supplement Section 13.2, "Training," of the AP1000 DCD, which is included in the [design certification] DC amendment as part of the BLN COL FSAR to address STD COL 13.2-1, dealing with the training program for plant personnel.

> *Table 13.4-201 provides milestones for training implementation.*

NUREG-0800, Section 13.2.1, establishes milestones for the licensed operators and non-licensed plant staff training programs and for the licensed operator requalification training program. The BLN COL FSAR has identified those milestones in Table 13.4-201. The staff determined that this is acceptable, as the milestone information included in this table meets the criteria found in NUREG-0800.

- *STD COL 18.10-1*

The NRC staff reviewed STD COL 18.10-1, related to COL Information Item 18.10-1 (COL Action Item 18.10.3-1). COL Information Item 18.10-1 states:

> *Combined License applicants referencing the AP1000 certified design will develop and implement training programs for plant personnel. This includes the training program for the operations personnel who participate as subjects in the human factors engineering verification and validation. These Combined License applicant training programs will address the scope of licensing examinations as well as new training requirements*.

The commitment was also captured as COL Action Item 18.10.3-1 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:

> *With regard to the training program development, the COL applicant will: (1) address the training program development considerations in NUREG-0711, (2) address relevant concerns identified in this report [NUREG-1793], and (3) identify the minimum documentation that the COL applicant will provide to enable the staff to complete its review.*

This section refers to Sections 13.1, "Organizational Structure of Applicant" and 13.2, "Training" regarding the training program development.

The NRC staff reviewed the resolution to STD COL 18.10-1, related to staffing and qualifications included under Section 18.10 of the BLN COL FSAR. The applicant provided the referenced NRC-endorsed NEI 06-13A, Revision 1, to address COL Information Item 18.10-1.

NEI 06-13A, Revision 1 was written to provide COL applicants with a generic program description for use with COL application submittals. In a letter dated December 5, 2008, the staff stated that the training template of NEI 06-13A, Revision 1, was an acceptable means for describing licensed operator and non-licensed plant staff training programs. The staff finds the applicant's incorporation of NEI 06-13A, Revision 1 to be acceptable because it utilizes an NRC-endorsed methodology.

In Table 1.9-202, "Conformance with SRP Acceptance Criteria," of the BLN COL FSAR, the applicant identified two exceptions to the criteria of NUREG-0800, Section 13.2, which recommends following the guidance in NUREG-0711 and RG 1.149. Further, the applicant stated in Table 1.9-202 that NEI 06-13A is incorporated by reference into the BLN COL FSAR. The staff's safety evaluation report for NEI 06-13A (ML0709504790) states that NEI 06-13A complies with the guidance in NUREG-0711 and RG 1.149. Therefore, the staff finds the two exceptions to the criteria in NUREG-0800, Section 13.2 to be acceptable because NEI 06-13A complies with the guidance in NUREG-0711 and RG 1.149.

License Conditions

- *Part 10, License Condition 3, Item B1*

The NRC staff finds the implementation milestone for the Reactor Operator Training Program (18 months prior to schedule date of initial fuel load) to be acceptable because it is consistent with 10 CFR 50.120.

- *Part 10, License Condition 6*

The applicant proposed a license condition in Part 10 of the VEGP COL application to provide a schedule to support the NRC's inspection of operational programs, including the Non-Licensed Plant Staff Training Program, (required in accordance with 10 CFR 50.120), Reactor Operator Training Program, and Reactor Operation Requalification Program. The proposed license condition is consistent with the policy established in SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," for operational programs in general, and is acceptable.

13.2.5 Post Combined License Activities

For the reasons discussed in the Technical Evaluation section above, the staff finds the following license conditions acceptable:

- License Condition (13-1) – The licensee shall implement the Reactor Operator Training Program at least 18 months prior to schedule date of initial fuel load.

- License Condition (13-2) – No later than 12 months after issuance of the COL, the licensee shall submit to the Director of the Office of New Reactors (NRO) a schedule that supports planning for and conduct of NRC inspection of the operational programs (the Non-Licensed Plant Staff Training Program (required in accordance with 10 CFR 50.120), Reactor Operator Training Program, and Reactor Operation Requalification Program). The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until these operational programs have been fully implemented.

13.2.6 Conclusion

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to the description and schedule of the training program for licensed operators, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the information presented in the WLS COL FSAR is acceptable because it meets the acceptance criteria provided in NUREG-0800, Section 13.2. The staff based its conclusion on the following:

- STD COL 13.2-1 incorporates by reference Nuclear Energy Institute (NEI) 06-13A, Revision 1, which provides an acceptable template for describing licensed operators and non-licensed plant staff training programs. The staff finds this acceptable, as it applies an NRC-endorsed approach.

- STD COL 18.10-1, relating to training, references WLS COL FSAR Section 13.2, in which the applicant committed to use WCAP-14655 to ensure a systematic approach to training development and has referenced NEI 06-13A, Revision 1. The staff finds this acceptable because it applies an NRC-endorsed approach.

13.3 Emergency Planning

13.3.1 Introduction

This section of the WLS COL FSAR addresses the plans, design features, facilities, functions, and equipment necessary for radiological emergency planning (EP) that must be considered in a COL application (hereinafter referred to as "COLA" or "application"). This includes both the COL applicant's onsite emergency plan and State and local (offsite) emergency plans, which the

NRC and the Federal Emergency Management Agency (FEMA) evaluated to determine whether the plans are adequate, and that there is reasonable assurance that they can be implemented. The emergency plans are an expression of the overall concept of operation and describe the essential elements of advance planning that have been considered and the provisions that have been made to cope with radiological emergency situations.

Duke is the applicant for the WLS Units 1 and 2 COLs (hereinafter referred to as “Lee Nuclear Station” for discussions of the site or plant, “WLS” as a description of the applicant, or “applicant”). Duke submitted its COLA (Revision 0) on December 12, 2007, for two new nuclear reactors, which will be located in the eastern portion of Cherokee County in north central South Carolina, approximately (56 kilometers (km)) (35 miles (mi)) southwest of Charlotte, North Carolina (NC). WLS encompasses approximately 768.9 hectares (1900 acres) of property. In the early 1970s, the site was evaluated for construction of three nuclear units. The NRC docketed the application on February 25, 2008 (Docket Nos. 52-018 and 52-019.

The applicant submitted a complete and integrated emergency plan for WLS pursuant to 10 CFR 52.79(a)(21), which consists of the Lee Nuclear Station Emergency Plan in Part 5 of the COLA (hereinafter referred to as “Emergency Plan” or “WLS Emergency Plan”), supplemental information that includes the offsite radiological emergency response plans for the States of South Carolina and North Carolina and the Counties of Cherokee, Cleveland, York, and the Lee Nuclear Station Evacuation Time Estimate (ETE) Report No. KLD TR-407, Revision 2, “William S. Lee Nuclear Station– Development of Evacuation Time Estimates,” March 2010 (hereinafter referred to as “ETE Report”). The application also includes Table 3.8-1, “Emergency Plan Inspections, Tests, Analyses, and Acceptance Criteria,” in Part 10, “License Conditions and ITAAC,” which provides a listing of EP ITAAC that address required elements of emergency planning that cannot be completed during the COLA stage, and that will be completed before initial fuel load. The COLA also references the AP1000 standard design certification, NUREG-1793 “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design,” Revision 19.

As described below, in consultation with FEMA, the staff reviewed the COLA, ETE Report, the applicant’s responses to RAIs, and generally available reference material in accordance with the guidance provided in the Standard Review Plan (SRP) NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Revision 3, March 2007, Section 13.3, “Emergency Planning,” and Section 14.3.10, “Emergency Planning – Inspections, Tests, Analyses, and Acceptance Criteria.” FEMA reviewed the offsite radiological emergency response plans of the States of South Carolina and North Carolina and local government plans for Cherokee, Cleveland, and York Counties.

In a February 17, 2010, letter, FEMA provided the NRC with its Interim Finding Report for Reasonable Assurance for the WLS COLA, which found that all planning standards associated with their review are adequate; the State and local emergency plans are adequate; and there is reasonable assurance that the plans can be implemented with no corrections needed. The staff reviewed the FEMA findings, and the overall FEMA conclusions are reflected below in Sections 13.3.4 and 13.3.6 of this report.

13.3.2 Summary of Application

WLS COL FSAR, Revision 11, Section 13.3, incorporates by reference AP1000 DCD, Revision 19, Section 13.3. In addition, the applicant provided the following in the COLA.

Departures

In WLS COL FSAR Tier 2[2], Table 1.8-201, "Summary of FSAR Departures from the DCD," and WLS COLA Part 7, "Departures and Exemption Requests," the applicant identified one plant-specific departure from the AP1000 generic DCD, which is associated with emergency planning:

- WLS DEP 18.8-1

  The Technical Support Center (TSC) is not located in the control support area (CSA) as identified in AP1000 DCD Section 18.8.3.5; the TSC location is as described in the Emergency Plan. Additionally, the Operations Support Center (OSC) is also being moved from the location identified in AP1000 DCD Sections 12.5.2.2 and 18.8.3.6 and as identified on AP1000 DCD Figures 1.2-18, 9A-3 (Sheet 1 of 3), 12.3-2 (Sheet 11 of 15), and 12.3-3 (Sheet 11 of 16); the OSC location is as described in the Emergency Plan.

The staff's evaluation of the applicant's description of this AP1000 DCD departure is addressed below in Section 13.3.4.8 of this report.

AP1000 COL Information Items

Consistent with the AP1000 Tier 2 DCD, in WLS COL FSAR Table 1.8-202, "COL Item Tabulation," the applicant identified AP1000 DCD COL (information) items, including the AP1000 DCD subsections and WLS COL FSAR sections where each COL item is resolved. In WLS COL FSAR Section 13.3, the applicant identified the following two COL items relating to emergency planning:

- STD COL 13.3-1

The applicant provided additional information in STD COL 13.3-1 to address COL Information Item 13.3-1 (COL Action Item 13.3-1) of the AP1000 DCD, which states:

> Combined License applicants referencing the AP1000 certified design will address emergency planning including post-72 hour actions and its communication interface.

[2] The definitions of Tier 1, Tier 2, and Tier 2*, which reflect design-related information contained in the generic AP1000 DCD, are provided in 10 CFR Part 52, Appendix D, Section II.

- STD COL 13.3-2

The applicant provided additional information in STD COL 13.3-2 to address COL Information Item 13.3.1 of the AP1000 DCD, which states:

> Combined License applicants referencing the AP1000 certified design will address the activation of the emergency operations facility [EOF] consistent with current operating practice and NUREG-0654/FEMA-REP-1 ["Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1 (hereinafter referred to as "NUREG-0654")].

The applicant also identified the following three additional COL items in their respective WLS COL FSAR sections, which relate to emergency planning:

- WLS COL 9.5-9 and WLS COL 9.5-10

In WLS COL FSAR Sections 9.5.2.5.1, "Offsite Interfaces," and 9.5.2.5.2, "Emergency Offsite Communications," the applicant provided additional information to address AP1000 DCD COL Information Items 9.5-9 and 9.5-10. As addressed by the applicant, offsite interfaces and emergency offsite communication are described in the Emergency Plan. COL Information Items 9.5-9 and 9.5-10 are as follows:

> WLS COL 9.5-9 – Combined License applicants referencing the AP1000 certified design will address interfaces to required offsite locations; this will include addressing the recommendations of NRC Bulletin (BL)-80-15 (COL Reference 21)[3] regarding loss of the emergency notification system due to a loss of offsite power.
>
> WLS COL 9.5-10 – The emergency offsite communication system, including the crisis management radio system, will be addressed by the Combined License applicant.

- WLS COL 18.2-2

In WLS COL FSAR Section 18.2, "Human Factors Engineering Program Management," the applicant provided additional information in WLS COL 18.2-2 to address AP1000 DCD COL Information Item 18.2-2 (COL Action Item 18.2.3.1-1). Specifically, the applicant stated that the EOF and TSC communication strategies and human factors attributes are described in the Emergency Plan. COL Information Item 18.2-2 states:[4]

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[3] NRC IE Bulletin No. 80-15 (BL-80-15), "Possible Loss of Emergency Notification System (ENS) with Loss of Offsite Power," June 18, 1980.

[4] See also, Section 18.2.7, "Evaluation of COL Information Item 18.2-2 (no comparable NUREG-1793 section)," of NUREG-1793, Supplement 2, Volume 2, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Plant Design – Docket No. 52-006," August 5, 2011 (published September 2011).

> Specific information regarding EOF and TSC communications, and EOF and TSC human factors attributes will be provided by the Combined Operating License applicant to address the Combined License information requested in this [DCD] subsection [i.e., DCD Tier 2 Subsection 18.2.6].

The staff's evaluation of the applicant's resolution of these five COL items is addressed below in Section 13.3.4.18 of this report.

Supplemental Information

- STD SUP 13.3-1

In WLS COL FSAR Section 13.3, "Emergency Planning," the applicant provided supplemental information in STD SUP 13.3-1, which states that WLS COL FSAR Table 13.4-201, "Operational Programs Required by NRC Regulations," provides milestones for emergency planning program implementation. STD SUP 13.3-1 is evaluated by the staff as part of its evaluation of License Condition 6 in Section 13.3.4.19 of this report.

- WLS SUP 14.3-1

The applicant provided the following statement in WLS COL FSAR Section 14.3.2.3.1, "Emergency Planning ITAAC (EP-ITAAC)," with regard to EP ITAAC:

> EP-ITAAC have been developed to address implementation of elements of the Emergency Plan. Site-specific EP-ITAAC are based on the generic ITAAC provided in Appendix C.II.1-B of Regulatory Guide 1.206. These ITAAC have been tailored to the specific reactor design and emergency planning program requirements.

The EP ITAAC are identified below in Table 13.3-1 of this report, and WLS SUP 14.3-1 is evaluated by the staff in Section 13.3.4.19 of this report.

Onsite Emergency Plan

Emergency planning for Units 1 and 2 is addressed throughout WLS COL FSAR, with the Radiological Emergency Plan for Units 1 and 2 (WLS Emergency Plan) provided in COLA Part 5. The WLS Emergency Plan addresses guidance and meets the intent of the criteria established in NUREG-0654. The WLS Emergency Plan consists of a full and integrated emergency plan. In addition, the WLS Emergency Plan is structured to have ten appendices (listed below), one of which is not used, which provide additional detailed information on specific aspects of emergency planning.

- Appendix 1 [Not Used]
- Appendix 2 Radiological Assessment and Monitoring
- Appendix 3 Public Alert and Notification System Description
- Appendix 4 Evacuation Time Estimate

- Appendix 5 Implementing Procedures
- Appendix 6 Emergency Equipment and Supplies
- Appendix 7 Certification Letters
- Appendix 8 Cross References to Regulations, Guidance, and State and Local Plans
- Appendix 9 Justification for Common EOF
- Appendix 10 Technical Support Center Description

Offsite Emergency Plans

Pursuant to 10 CFR 50.33, "Contents of Applications; General Information," paragraph (g), a COL applicant is required to submit the radiological emergency response plans of State and local governments that are wholly or partially within the 16-kilometer (km) (10-mi) plume exposure pathway emergency planning zone (EPZ), as well as plans of State governments wholly or partially within the 80-km (50-mi) ingestion pathway EPZ (hereinafter referred to as the "10-mi EPZ" and "50-mi EPZ"). The WLS COLA includes supplemental information, consisting of the offsite radiological emergency response plans of the States of South Carolina and North Carolina and local government plans for Cherokee, Cleveland, and York Counties. The supplemental information also includes the detailed ETE Report for the 16-km (10-mi) EPZ, which is discussed in Section 13.3.4.17 of this report.

ITAAC

Part 10, "Proposed License Conditions (Including ITAAC)," of the WLS COL application provides information regarding EP ITAAC. The EP ITAAC is evaluated in Section 13.3.4.19 of this report.

License Conditions

COLA Part 10, "License Conditions (Including ITAAC)," includes the following proposed license conditions related to EP:

- Part 10, License Condition 1

The applicant proposed a license condition to incorporate EP ITAAC into the COL, which are identified in Table 3.8-1 of Appendix B to Part 10 of the WLS COL application.

- Part 10, License Condition 4 (Emergency Planning Actions)

  The licensee shall submit a fully developed set of site-specific Emergency Action Levels (EALs) to the NRC in accordance with the NRC-endorsed version of NEI 07-01, Rev. 0, with no deviations. The EALs shall have been discussed and agreed upon with State and local officials. These fully developed EALs shall be submitted to the NRC for confirmation not less than 180 days prior to the date scheduled for initial fuel load. (Identified below as License Condition (13-3))

Prior to the full participation exercise to be conducted in accordance with the requirements of Appendix E to 10 CFR Part 50, Duke Energy shall identify the specific locations of the reception centers and relocation sites and shall obtain Letters of Agreement for locations not under Duke Energy's control. (Identified below as License Condition (13-4))

At least two (2) years prior to scheduled initial fuel load, Duke Energy shall have performed an assessment of emergency response staffing in accordance with NEI 10-05, "Assessment of On-Shift Emergency Response Organization Staffing and Capabilities," Revision 0. (Identified below as License Condition (13-5))

Prior to the full-participation exercise to be conducted in accordance with the requirements of Appendix E to 10 CFR Part 50, Duke Energy will have available for NRC inspection Letters of Agreement with the entities listed in Appendix 7 of the Lee Nuclear Station COLA Part 5, Emergency Plan. These Letters of Agreement will detail each entity's specific emergency planning responsibilities, including response to hostile action affecting the plant site, and certify the entity's concurrence with their responsibilities. (Identified below as License Condition (13-6))

Prior to fuel load, Duke Energy will demonstrate the integrated capability and functionality of the Emergency Operations Facility (EOF) for activation and operation of the facility to respond to emergency events at both the Lee Nuclear Station and one additional nuclear facility that is supported by the EOF. Integrated communication and data capability and functionality will include the Technical Support Centers for Lee Nuclear Station and one additional nuclear facility, and other Federal, State, and local coordination centers as appropriate. (Identified below as License Condition (13-7))

- Part 10, License Condition 6, Items a. and e. (Operational Program Readiness)

  a. The licensee shall submit to the appropriate Director of the NRC, a schedule, no later than 12 months after issuance of the COL, that supports planning for and conduct of NRC inspections of operational programs listed in the operational program FSAR Table 13.4-201. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until either the operational programs in the FSAR table have been fully implemented or the plant has been placed in commercial service, whichever comes first. This schedule shall address:

  e. An emergency response data system (ERDS) implementation program plan consistent with 10 CFR Part 50, Appendix E, Section VI. (Identified below as License Condition (13-9))

- Part 10, License Condition 12.C (Fukushima Actions – Emergency Planning Actions)

  Staffing (Identified below as License Condition (13-10))

At least two (2) years before the latest date set forth in the schedule for completing the inspections, tests, and analyses in the ITAAC submitted in accordance with 10 CFR 52.99(a), the Licensee shall have performed assessments of the on-site and augmented staffing capability to satisfy the regulatory requirements for responding to a multi-unit event. The staffing assessments will be performed in accordance with NEI 12-01, Revision 0.

- Incorporation of corrective actions identified in the staffing assessments required by this condition, and
- Identification of how the augmented staff will be notified given degraded communications capabilities.

Communications (Identified below as License Condition (13-11))

At least two (2) years before the latest date set forth in the schedule for completing the inspections, tests, and analyses in the ITAAC submitted in accordance with 10 CFR 52.99(a), the Licensee shall have performed an assessment of on-site and off-site communications systems and equipment relied upon during an emergency event to ensure communications capabilities can be maintained during an extended loss of ac power. The communications capability assessment shall be performed in accordance with NEI 12-01, “Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities,” Revision 0.

At least one hundred eighty (180) days before the date scheduled for initial fuel load set forth in the notification submitted in accordance with 10 CFR 52.103(a), the Licensee shall complete implementation of corrective actions identified in the communications capability assessment described above, including any related emergency plan and implementing procedure changes and associated training.

ITAAC

WLS COLA Part 10 proposes License Condition 1 (described above) that incorporates into the COL the ITAAC identified in Appendix B of Part 10. WLS COLA Part 10, Appendix B includes Table 3.8-1 (EP ITAAC) and incorporates by reference AP1000 DCD Tier 1, Table 3.1-1 (ITAAC). The EP ITAAC are evaluated below in this report.

13.3.3 Regulatory Basis

The regulatory basis of the AP1000 DCD information incorporated by reference is addressed in NUREG-1793, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design,” and its supplements. The applicable regulatory requirements and guidance for emergency planning are as follows:

- 10 CFR 52.79(a)(21) requires that the FSAR include emergency plans that comply with the requirements of 10 CFR 50.47, “Emergency plans,” and 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” Appendix E, “Emergency Planning and Preparedness for Production and Utilization Facilities.” In addition, 10 CFR 52.79(a)(22)(i) requires emergency pllanning certifications from State and local governmental agencies with emergency planning responsibilities. Under 10 CFR 50.47(a)(1)(ii), no initial COL under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," will be issued unless a finding is made by the NRC that there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. In addition, 10 CFR 50.47(a)(2), states that the NRC will base its findings on a review of the FEMA findings and determinations as to whether State and local emergency plans are adequate, and whether there is reasonable assurance that they can be implemented, and on the NRC assessment as to whether the applicant's onsite emergency plans are adequate and whether there is reasonable assurance that they can be implemented.

- The staff also considered the applicable requirements in 10 CFR 50.33(g); 10 CFR 52.80, "Contents of applications; additional technical information"; 10 CFR 52.83, "Finality of referenced NRC approvals; partial initial decision on site suitability"; and 10 CFR 100.21, "Non-seismic siting criteria."

- NUREG-0800 identifies NUREG-0654 and other related guidance that the staff should consider during its review. The related acceptance criteria are identified in NUREG-0800, Section 13.3.II, and the applicable regulatory guidance for reviewing emergency preparedness as an operational program is established in NUREG-0800, Section 13.4. In addition, the staff considered NUREG/CR-7002, "Criteria for Development of Evacuation Time Estimate Studies" (November 2011), the current guidance for conducting and evaluating evacuation time estimates; NUREG/CR-6863, "Development of Evacuation Time Estimate Studies for Nuclear Power Plants" (January 2005); and Interim Staff Guidance (ISG) NSIR/DPR-ISG-01.[5]

- 44 CFR Part 350, "Review and Approval of State and Local Radiological Emergency Plans and Preparedness," and 44 CFR Part 352, "Commercial Nuclear Power Plants: Emergency Preparedness Planning," provide procedures for FEMA's review and evaluation of the adequacy of offsite radiological emergency planning and preparedness. Pursuant to 44 CFR Part 353, "Memorandum of Understanding Between Federal Emergency Management Agency and Nuclear Regulatory Commission Relating to Radiological Emergency Planning and Preparedness," Appendix A, "Memorandum of Understanding Between Federal Emergency Management Agency and Nuclear Regulatory Commission" (58 FR 47996, September 14, 1993), FEMA provided its findings and determinations on offsite planning to the NRC for NRC's use in the licensing process.

13.3.4 Technical Evaluation

The staff reviewed the information in WLS COL FSAR Section 13.3, "Emergency Planning," and the WLS Emergency Plan for conformance with applicable standards and requirements

[5] NSIR/DPR-ISG-01, Revision 0, "Interim Staff Guidance - Emergency Planning for Nuclear Power Plants," November 2011, provides updated guidance based on changes to emergency planning regulations in 10 CFR 50.47 and 10 CFR Part 50, Appendix E, that were published as a Final Rule in the *Federal Register* (FR) on November 23, 2011 (76 FR 72560), and on integrated offsite response organization event response concepts with onsite emergency planning programs.

identified in NUREG-0800, Sections 13.3 and 14.3.10. The complete set of emergency planning ITAAC for the new reactors is provided in Table 13.3-1, "WLS Units 1 & 2 ITAAC," of this report, and various ITAAC are discussed throughout this section of the report. In addition, the staff reviewed selected portions of the emergency response plans for the States of South Carolina and North Carolina and local government plans for Cherokee, Cleveland, and York Counties. The staff completed this review for understanding and content, in relation to consistency with various sections of the WLS Emergency Plan that address offsite support and response. The staff checked the referenced DCD to ensure that the combination of the DCD and the COLA represents the complete scope of information relating to this review topic.[6] The staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to emergency planning pursuant to 10 CFR 52.77, 10 CFR 52.79, 10 CFR 52.80, and 10 CFR 100.21. The results of the staff's evaluation of the referenced DCD are documented in NUREG-1793 and its supplements.

The staff's and FEMA's technical reviews of the WLS COLA addressed all of the relevant evaluation criteria in the 16 planning standards (i.e., A through P) of NUREG-0654, consistent with NUREG-0800, Section 13.3, which cites the applicable regulations.

In WLS COLA Part 1, the applicant incorporated by reference the AP1000 DCD. WLS COL FSAR Section 13.3 further incorporates by reference AP1000 DCD Section 13.3, "Emergency Planning." WLS COLA Part 5 provides the WLS Emergency Plan, which consists of the basic emergency plan and ten appendices. The basic plan follows the format of NUREG-0654, and provides detailed information regarding each of the 16 planning standards and associated evaluation criteria in NUREG-0654. The format of the staff's review of the WLS Emergency Plan is patterned after the 16 planning standards, which reflect the requirements in 10 CFR 50.47(b)(1) through (b)(16). 10 CFR Part 50, Appendix E, provides additional requirements that duplicate or supplement the evaluation criteria associated with the planning standards. The staff's review of the various aspects of 10 CFR Part 50, Appendix E is included within the associated planning standards review.

13.3.4.1 Assignment of Responsibility (Organizational Control)

The regulation in 10 CFR 50.47(b)(1), as reflected in NUREG-0654 as Planning Standard A, 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. In addition, 10 CFR Part 50, Appendix E, Section III requires that the emergency plans incorporate information about the emergency response roles of supporting organizations and offsite agencies, and that information shall be sufficient to provide assurance of coordination among the supporting groups and with the licensee. 10 CFR Part 50, Appendix E. Section IV.A requires, among other things, a description of the local offsite services to be provided in support of the licensee's emergency organization; the identification of, and assistance expected from, appropriate local, State, and Federal agencies with responsibilities for coping with emergencies, including hostile action at the site; identification of the State and

[6] See Section 1.2.2 of this report for a discussion on the staff's review related to verification of the scope of information to be included within a COL application that references a design certification (DC).

local officials responsible for planning for, ordering, and controlling appropriate protective actions, including evacuations when necessary; and a detailed analysis demonstrating that on-shift personnel assigned emergency functions are not assigned responsibilities that would prevent timely performance of their assigned functions as specified in the WLS Emergency Plan.

The regulatory guidance provided in NUREG-0654 Evaluation Criterion II.A.3 states, in part, that each plan shall include written agreements referring to the concept of operations developed among Federal, State, and local agencies and other support organizations having an emergency response role within the EPZs. In addition, 10 CFR 52.79(a)(22)(i) states that the COLA must contain all emergency plan certifications that have been obtained from the State and local governmental agencies with emergency planning responsibilities. These certifications must state that (1) the proposed emergency plans are practicable; (2) these agencies are committed to participating in any further development of the plans, including any required field demonstrations; and (3) these agencies are committed to executing their responsibilities under the plans in the event of an emergency.

In the WLS Emergency Plan, Section II.A, “Assignment of Responsibility (Organizational Control),” the applicant described the primary responsibilities and organizational control of WLS, Federal, State, county and other emergency response organizations (EROs) within the 16-km (10-mi) EPZ and the 80-km (50-mi) EPZ. The staff reviewed this section, as well as other relevant portions of the application, to determine whether the application conforms to the applicable guidance and complies with the pertinent regulatory requirements. The staff’s primary focus was to evaluate the WLS Emergency Plan against NUREG-0654, Planning Standard A, “Assignment of Responsibility (Organization Control).” Planning Standard A provides the detailed evaluation criteria that the staff should consider in determining whether the WLS Emergency Plan meets the applicable regulatory requirements in 10 CFR 50.47(b)(1).

WLS Emergency Plan, Section II.A, “Assignment of Responsibility (Organizational Control),” describes the relationships and concept of operations for the organizations and agencies that are a part of the overall ERO, and identifies the various Federal, State, county and local government agencies and organizations that are involved in a response to an emergency at WLS. WLS Emergency Plan, Figure II-1, “Emergency Response Organization Interrelationships,” illustrates the interrelationships of organizations that will be participating in emergency response. WLS Emergency Plan, Appendix 7, “Certification Letters,” contains certification letters signed by the supporting State and local agencies.

WLS Emergency Plan, Section II.A.1.b, “Assignment of Responsibility,” footnote 4, states that in the event of a security related attack on the site by a hostile force, a brief notification (site name, emergency classification, if determined, and nature of threat) is provided to the NRC following notification of the designated State and local authorities and within approximately 15 minutes of the discovery of the event.

WLS Emergency Plan, Section II.A.1.d, “Individual in Charge of Emergency Response,” identifies the individual in charge for coordinating the emergency response as the Operations Shift Manager, who will assume the role as Emergency Coordinator. The Operations Shift Manager is relieved as Emergency Coordinator when the Station Manager or a qualified alternate reports to the station, and he or she is updated as to the status of the unit, the emergency actions taken, and the current status of the emergency. Once the EOF is activated the EOF director assumes responsibility for ensuring that the appropriate offsite interface activities are performed (e.g., notifications of emergency status to State and local governments and NRC; and recommending offsite protective measures to the State).

WLS Emergency Plan, Sections II.A.1.e, “24-Hour Emergency Response Capability,” and II.A.4, “Continuous Operations,” states, in part, that Duke Energy and WLS maintains a 24-hour emergency response capability, communication links are staffed, and multiple responders are trained for key emergency response positions, consistent with the training requirements established in WLS Emergency Plan, Section II.O, “Radiological Emergency Response Training.” The Emergency Coordinator or EOF Director is identified as the individual from the principal organization in charge, and he or she has the responsibility for ensuring continuity of technical, administrative, and material resources during emergency operations. In addition, Section II.B.7, “Corporate Off-site Support for Plant Staff,” of the WLS Emergency Plan states, “The EOF is capable of 24 hours/day operation for a protracted period.”

WLS Emergency Plan, Section II.A.3, “Written Agreements,” states that Appendix 7 includes initial certification letters established between Duke Energy, the State and local government agencies, and private sector organizations that will be supporting the emergency response effort. As previously described in Section 13.3.2, “Summary of Application,” of this report, the applicant will develop updated letters of agreement (LOAs) in accordance with License Condition (13-6), which is evaluated in Section 13.3.4.3 of this report.

In RAI 25, Question 13.03-54(A), the staff requested that the applicant provide the title of the State and local officials who will be responsible for implementing offsite protective actions. In a December 23, 2008, response, the applicant provided additional information related to State and local officials who will be responsible for implementing offsite protective actions and provided proposed revisions to WLS Emergency Plan, Section II.A.1.b. The States of South Carolina and North Carolina, and counties of Cherokee, Cleveland, and York have emergency response plans that specify the responsibilities and functions for the major agencies, departments, and key individuals of their emergency response organizations. The governors for these states have the overall command authority for radiological and non-radiological aspects of a nuclear incident, and will provide for public protection through assignment of appropriate States’ resources and agencies. Within the State of South Carolina, should a rapidly-developing emergency condition arise that requires implementation of protective actions before the State Emergency Operations Center can be activated, affected county managers may implement the facility-recommended protective actions without prior consultation with the Director of the Emergency Management Division or the governor. Within the State of North Carolina, should a rapidly-developing emergency condition arise that requires the implementation of protective actions before the State Emergency Operations Center can be activated, affected Chairmen of the Board of County Commissioners may implement the facility recommended protective actions without prior consultation with the State agencies or governor. Since the above referenced States and counties reside within the 16-km (10-mi) and 80-(km (50-mi) EPZs, the applicant included the respective emergency response plans as supplemental information to the WLS COLA pursuant to 10 CFR 50.33(g).

In its Interim Finding Report for Reasonable Assurance, FEMA found that the offsite emergency plans are adequate for this planning standard and the associated evaluation criteria found in NUREG-0654.

The staff reviewed the information provided by the applicant and finds that the applicant adequately assigned primary responsibilities for emergency response, and has the necessary staffing to respond to and augment its initial response on a continuous basis. The staff notes that the applicant is capable of providing 24-hour-per-day emergency response and staffing of communication links, including continuous (24-hour) operations for a protracted period. In addition, the applicant identified the appropriate organizations that are intended to be part of the overall response organization, and established the emergency responsibilities of the various supporting organizations, including providing adequate written agreements. The applicant specified the concept of operations and its relationship to the total effort, illustrated the interrelationships in a block diagram, and has identified the individuals in charge of the emergency response and for ensuring continuity of resources.

In addition, the staff confirms that the applicant incorporated information about the emergency response roles of supporting organizations and offsite agencies into the WLS Emergency Plan, and finds that the information provided by the applicant and reviewed by the staff is sufficient to provide assurance of coordination among the supporting groups and with the licensee. Further, the applicant described the local offsite services to be provided in support of the licensee's emergency organization, and identified the assistance expected from appropriate local, State, and Federal agencies, including State and/or local officials responsible for planning for, ordering, and controlling appropriate protective actions.

The staff confirms that the certification letters in WLS Emergency Plan, Appendix 7 state that (1) the proposed emergency plans are practicable; (2) these agencies are committed to participating in any further development of the plans, including any required field demonstrations; and (3) these agencies are committed to executing their responsibilities under the plans in the event of an emergency. Therefore, the staff finds that the WLS COLA meets the requirements of 10 CFR 52.79(a)(22)(i).

The staff finds the additional information and clarifications provided by the applicant in the December 23, 2008, response to RAI 25, Question 13.03-54(A) acceptable because it conforms to the guidance in NUREG-0654. Accordingly, the staff considers these questions resolved.

Conclusion

Subject to License Condition (13-6), the staff concludes that the information provided in the WLS COLA conforms to the evaluation criteria in NUREG-0654, Planning Standard A. Accordingly, the staff finds the information acceptable and meets the relevant requirements of 10 CFR 52.79(a)(22)(i), 10 CFR 50.47(b)(1) and 10 CFR Part 50, Appendix E, Sections III and IV.A, insofar as the information describes the primary responsibilities for emergency response by the applicant, State and local organizations within the EPZs, and various supporting organizations, and that each principal response organization has staff to respond to and augment its initial response on a continuous basis.

13.3.4.2 Onsite Emergency Organization

The regulation in 10 CFR 50.47(b)(2), as reflected in NUREG-0654 as Planning Standard B, "Onsite Emergency Organization," 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 interfaces among various onsite response activities and offsite support and response activities are specified. In addition, 10 CFR Part 50, Appendix E, Section IV.A requires a description of the organization for coping with radiological emergencies, including definition of authorities, responsibilities, and duties of individuals assigned to the licensee's emergency organization, and the means for notification of such individuals in the event of an emergency. This discussion shall include a description of the normal plant operating organization, onsite emergency response organization, headquarters personnel who will augment the onsite emergency organization, and local offsite services to be provided in support of the licensee's emergency organization. The emergency plan shall identify persons within the licensee organization who will be responsible for making offsite dose projections, and other employees with special qualifications for coping with emergency conditions that may arise. Other persons with special qualifications, who are not licensee employees and who may be called upon for assistance, shall also be identified, including a description of the special qualifications. 10 CFR Part 50, Appendix E, Section IV.A.9 requires a detailed analysis demonstrating that on-shift personnel assigned emergency plan implementation functions are not assigned responsibilities that would prevent the timely performance of their assigned functions, as specified in the emergency plan.

In WLS Emergency Plan, Section II.B, "On-site Emergency Organization," the applicant described the ERO, its key positions and associated responsibilities, including outlining the staffing requirements that provide initial emergency response actions and provisions for timely augmentation of on-shift personnel. The staff reviewed this section, as well as other relevant portions of the application, to determine whether the application conforms to the applicable guidance and complies with the pertinent regulatory requirements. The staff's primary focus was to evaluate the emergency plan against NUREG-0654, Planning Standard B, "Onsite Emergency Organization." Planning Standard B provides the detailed evaluation criteria that the staff should consider to determine whether the emergency plan meets the applicable regulatory requirements in 10 CFR 50.47(b)(2).

WLS COL FSAR Section 13.1, "Organizational Structure of Applicant," (referenced from the DCD) and WLS COL FSAR Table 13.1-201, "Generic Position/Site Specific Position Cross Reference," provides a description of the proposed operating plant staffing, including position titles and functions. The normal plant personnel complement is established with the Site Executive – Plant Management directly responsible for the management and direction of activities associated with the efficient, safe, and reliable operations of the station. The Site Executive is assisted in management and technical support activities by the Plant Manager. The Plant Manager is responsible for onsite activities necessary for safe operation and maintenance of the plant (e.g., Operations, Chemistry, and Outage Management). Additionally, the Plant Manager has overall responsibility for occupational and public radiation safety. The Functional Manager in charge of operations reports to the Plant Manager and has overall responsibility for the day-to-day operation of the plant with assistance from the Assistant Functional Manager - Operations. The manager in charge on-shift (Shift Manager) is a licensed senior reactor operator (SRO) responsible for the control room command function, and is the Plant Manager's direct management representative for the conduct of operations. As such, the manager in charge on-shift has the responsibility and authority to direct the activities and personnel onsite as required to do the following:

- Protect the health and safety of the public, the environment, and personnel on the plant site

- Protect the physical security of the plant
- Prevent damage to site equipment and structures
- Comply with the operating license

WLS Emergency Plan, Section II.B states that the Shift Manager position is staffed at all times. In an emergency, he or she will act as the Emergency Coordinator until relieved by a qualified member of management or termination of the emergency. The Emergency Coordinator is responsible for initiating required emergency response actions (e.g., activation of emergency personnel and facilities, and authorizing emergency exposure limits). Other Emergency Coordinator responsibilities include: emergency classification, authorizing notification to the NRC, State, and local authorities, and the decision to notify and recommend protective actions to authorities responsible for offsite emergency measures. These responsibilities are designated as non-delegable. The Emergency Coordinator may also request assistance from any organization deemed necessary to mitigate the emergency. Once the EOF is activated the EOF director assumes responsibility for ensuring that the appropriate offsite interface activities are performed (e.g., notifications of emergency status to State and local governments and NRC; and recommending offsite protective measures to the State). At any time during an emergency should the Shift Manager be rendered unable to fulfill the duties and responsibilities of the Emergency Coordinator due to illness or injury, the Unit Supervisor (present on shift at all times) will assume the Emergency Coordinator position until relieved by a qualified member of the management staff.

The plant also has personnel on-shift at all times that can provide an initial response to an emergency event. WLS Emergency Plan, Table II-2, “Plant Staff Emergency Functions,” describes positions and major tasks to be performed by persons assigned to the functional areas of emergency activity. Upon declaration of an emergency, members of the plant staff assume positions in the ERO consistent with their training and management assignments, and provide for key functions of accident assessment, radiological monitoring and analysis, security, fire-fighting, first aid and rescue, and communication. On-shift staffing will be augmented with additional ERO personnel at an Alert and higher emergency classifications or earlier, as deemed necessary. WLS Emergency Plan, Section II.B states that the minimum staff required to conduct routine and immediate emergency operations is maintained at the station consistent with 10 CFR Part 50, Appendix E. In addition, minimum staffing was established based on the guidance provided in NUREG-0654, Table B-1, and provisions of other emergency plans from currently licensed Duke Energy facilities. The positions, titles and major tasks to be performed by station emergency responders are further described in emergency plan implementing procedures (EPIPs). Additional personnel may be designated by station management or the EOF Director as emergency responders providing special expertise deemed beneficial, but not mandatory, to the planned response based on the technical requirements of the position. WLS Emergency Plan, Figure II-2, “Emergency Response Organization–Site Only,” and WLS Emergency Plan, Figure II-3, “Offsite Emergency Response Organization,” illustrate the high level organizations that will be located in respective emergency response facilities (ERFs), which are the TSC, the OSC, EOF and Joint Information Center (JIC). WLS Emergency Plan, Section II.B.6, “Interface Between Functional Areas,” and WLS Emergency Plan, Figure II-1, “Emergency Response Organization Interrelationships,” identify and illustrate the interfaces among functional areas of the station emergency response activity, Duke Energy’s corporate support, and the affected State and local government response organizations. The applicant proposed EP-ITAAC 10.1 to verify that the emergency plan implementing procedures provide minimum and augmented on-shift staffing levels consistent with WLS Emergency Plan, Table II-2.

In RAI 25, Question 13.03-55(A), the staff requested that the applicant provide additional information related to staffing of accountability, decontamination, and public information positions. In a December 23, 2008, response, the applicant stated that on-shift security personnel are responsible for accountability; decontamination activities are conducted by on-shift Radiation Protection Technicians initially, and public information is handled by the EOF.

In RAI 25, Questions 13.3-55(M), (P), (P.2), and (Q), the staff requested that the applicant provide additional information related to staffing of the on-shift dose assessment as a part of the continual assessment capability. In a December 23, 2008, response, the applicant stated that the position and function would be staffed and performed in the EOF. While evaluating WLS Emergency Plan, Table II-2, “Plant Staff Emergency Functions,” the staff noted that Footnote 3 indicated that there will be personnel assigned to the shift, who are trained and qualified to perform dose assessment functions. Furthermore, in a supplement to the initial response to RAI 25, Questions 13.3-55(M), (P), (P.2), and (Q), the applicant stated that there will be an individual on-shift with the qualification to perform offsite dose projections until relieved by staff augmentation. This provides for the on-shift capability to perform dose assessment in the determination of emergency classification, onsite protective action, and offsite protective action recommendations.

WLS Emergency Plan, Section II.B.7, “Corporate Off-site Support for the Plant Staff,” states that upon declaration of an Alert, Site Area Emergency, or General Emergency, the Emergency Coordinator directs the activation and notification of the onsite and offsite ERFs. Duke Energy management, technical, and administrative personnel staff the EOF and provide or coordinate augmented support for the plant staff. The Duke Energy corporate staff provides management, technical, and administrative support as needed to support the plant staff and to relieve the plant staff of external coordination responsibilities, including notification of and coordination with offsite authorities and release of information to the media. In addition to the activities identified in WLS Emergency Plan, Table II-2, Duke Energy corporate staff provides logistical support for plant personnel; technical support for planning and recovery operations; management-level interface with governmental authorities; and coordination with and the release of information to the news media.

WLS Emergency Plan, Section II.B.8, “Support from Contractor and Private Organizations,” states that the Institute of Nuclear Power Operations (INPO), when notified of an emergency classification, will serve as a clearinghouse for industry-wide support and provide requested emergency response technical assistance, including emergency staffing and equipment. The applicant may request that the reactor vendor, Westinghouse, provide technical support for emergency response activities. If required at the time of the event, additional resources can be obtained from consultants and vendors through purchase agreements with the supporting institutions based on their expertise and plant needs. In addition, the applicant has established and will maintain agreements for emergency response support services, including firefighting, rescue squad, and medical and hospital services. WLS Emergency Plan, Section II.L describes the arrangements for medical support services, including hospital and ambulance support, and is addressed in Section 13.3.4.12 of this report. WLS Emergency Plan, Appendix 7 provides the certification letters for organizations providing these services. (Emergency response support and resources are further described in Section 13.3.4.3 of this report.)

WLS Emergency Plan, Section II.B.9, “Local Emergency Response Support,” describes the agreements between the applicant and local emergency response support services, including firefighting, rescue squad, medical and hospital services. The applicant’s emergency plan implementing procedure, “Site Response to Security Events,” provides information regarding measures to integrate offsite response resources and capabilities into the onsite response activities.

Fukushima Dai-ichi – NTTF Recommendation 9.3

On March 12, 2012, the NRC requested information pursuant to the 10 CFR 50.54(f) process from all power reactor licensees and holders of construction permits, associated with the NRC Near-Term Task Force (NTTF) review of the accident at the Fukushima Dai-ichi nuclear facility. In NTTF Recommendation 9.3, the NTTF addressed staffing and communication provisions for enhancing emergency preparedness. On January 23, 2013, the NRC issued a follow-up letter, which identified eight generic technical issues that need to be addressed as part of NTTF Recommendation 9.3 for conducting the communication capability assessment.

In an April 25, 2012, letter, the NRC informed the existing licensees and COL applicants that the staff would issue an RAI concerning the implementation of the NTTF recommendations in SECY-12-0025. In RAI 105, Question 01.05-4, the staff requested that the applicant address Recommendation 9.3, "Provisions for Enhancing Emergency Preparedness." The NRC issued this information request regarding the power supplies for communication systems and staffing to determine if additional regulatory action is warranted. This request was based upon NTTF Recommendation 9.3, which proposed that facility emergency plans provide for a means to power communication equipment needed to communicate onsite and offsite during an extended loss of alternating current power and staffing to fill all necessary positions to respond to a mult-unit event. In a June 11, 2012, response to RAI 105, Question 01.05-4, the applicant proposed License Condition 12.C “Emergency Planning Actions” in Part 10 of the WLS COLA. With respect to staffing, the staff refers to the following as License Condition (13-10):

Proposed License Condition (13-10):

> Staffing:
>
> At least two (2) years before the latest date set forth in the schedule for completing the inspections, tests, and analyses in the ITAAC submitted in accordance with 10 CFR 52.99(a), the Licensee shall have performed assessments of the on-site and augmented staffing capability to satisfy the regulatory requirements for responding to a multi-unit event. The staffing assessments will be performed in accordance with NEI 12-01, Revision 0.
>
> At least 180 days before the date scheduled for initial fuel loading set forth in the notification submitted in accordance with 10 CFR 52.103(a), the Licensee shall revise the Emergency Plan to include the following:
>
> - Incorporation of corrective actions identified in the staffing assessments required by this condition, and

- Identification of how the augmented staff will be notified given degraded communications capabilities.

In WLS COLA Part 10, the applicant proposed License Condition 12.C, “Fukushima Actions – Emergency Planning Actions,” which addresses both the staffing and communication areas addressed in NTTF Recommendation 9.3. The staff reviewed License Condition (13-10) and finds it acceptable, except for the scheduling of the assessment, because it is consistent with NTTF Recommendation 9.3 and reflects the use of NEI technical report NEI 12-01, “Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities,” Revision 0, which the NRC has endorsed as an acceptable method for licensees to employ when addressing NTTF Recommendation 9.3.[7] To address the scheduling of completing the actions for staffing assessments from “2 years” to “18 months,” the staff identified License Condition (13-10) below. (Emergency communication and the other part of License Condition 12.C (referred to as License Condition (13-11)) are addressed in Section 13.3.4.6 of this report.)

License Condition (13-10):

- At least 18 months before the latest date set forth in the schedule submitted in accordance with 10 CFR 52.99(a) for completing the inspections, tests, and analyses in the ITAAC, the licensee shall have performed assessments of the onsite and augmented staffing capability for response to a multi-unit event. The staffing assessments will be performed in accordance with NEI 12-01, “Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities,” Revision 0.

  At least 180 days before the date scheduled for initial fuel load set forth in the notification submitted in accordance with 10 CFR 52.103(a), Duke Energy shall revise the WLS Emergency Plan to include the following:

  - incorporation of corrective actions identified in the staffing assessments described above

  - identification of how the augmented staff will be notified given degraded communication capabilities

As previously referenced, with the staff’s revisions incorporated above, the staff finds License Condition (13-10) and the response to RAI 01.05-acceptable.

Enhancements to Emergency Preparedness Regulations

On November 23, 2011, the NRC published a Final Rule, “Enhancements to Emergency Preparedness Regulations” (hereinafter referred to as “Final Rule”), which included a new

[7] See (1) NRC May 15, 2012, letter, ‘U.S. Nuclear Regulatory Commission Review of NEI 12-01, “Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities,” Revision 0, May 2012’ (ADAMS Accession No. ML12131A043); (2) NEI May 3, 2012, letter, ‘Transmittal of NEI 12-01, “Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities,” Revision 0, May 2012’ (ADAMS Accession No. ML12125A411); and (3) NEI Report No. 12-01, Revision 0, “Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities,” May 2012 (ADAMS Accession No. ML12125A412).

requirement in 10 CFR Part 50, Appendix E, Section IV.A associated with on-shift ERO personnel. Specifically, 10 CFR Part 50, Appendix E, Section IV.A.9 requires nuclear power reactor licensees conduct a detailed analysis demonstrating that on-shift personnel assigned emergency plan implementation functions are not assigned responsibilities that would prevent the timely performance of their assigned functions as specified in the emergency plan.

As part of the issuance of the Final Rule, the NRC issued associated guidance in ISG Nuclear Security and Incident Response (NSIR)/Division of Preparedness and Response (DPR)-ISG-01. In NSIR/DPR-ISG-01, Section IV.C, "On-Shift Staffing Analysis," the NRC endorsed NEI technical report NEI 10-05, "Assessment of On-Shift Emergency Response Organization Staffing and Capabilities," Revision 0, June 2011 – stating, in part, that NEI 10-05 establishes a standard methodology for a licensee to perform the 10 CFR Part 50, Appendix E, Section IV.A.9 required staffing analysis, and that the NRC has reviewed NEI 10-05 and found it an acceptable methodology for this purpose.

In WLS COLA Part 10, as part of the applicant's proposed License Condition 4 "Emergency Planning Actions," which addresses the requirements in 10 CFR Part 50, Appendix E, Section IV.A.9 for a detailed on-shift staffing analysis associated with the emergency plan. The applicant proposed the following license condition, which the staff refers to as License Condition (13-5).

Proposed License Condition (13-5):

> At least two (2) years prior to scheduled initial fuel load, Duke Energy shall have performed an assessment of emergency response staffing in accordance with NEI 10-05, "Assessment of On-Shift Emergency Response Organization Staffing and Capabilities," Revision 0.

With the staff's revisions incorporated below, the staff finds License Condition (13-5) acceptable because it is consistent with the Final Rule and NSIR/DPR-ISG-01:

License Condition (13-5):

> At least 18 months before the latest date set forth in the scheduled submitted in accordance with 10 CFR 52.99(a) for completing the inspections, tests, and analyses in the ITAAC, Duke Energy shall have performed a detailed staffing analysis, in accordance with NEI 10-05, "Assessment of On-Shift Emergency Response Organization Staffing and Capabilities," Revision 0.
>
> At least one hundred eighty (180) days before the date scheduled for initial fuel load set forth in the notification submitted in accordance with 10 CFR 52.103(a), Duke Energy shall revise the WLS Emergency Plan to incorporate any changes identified in the staffing analysis that are needed to bring staff to the required levels.

The staff reviewed the information provided by the applicant and finds that the applicant adequately designated an individual as the Emergency Coordinator who has the authority and responsibility to initiate emergency actions, including recommending protective actions to the authorities responsible for implementing offsite emergency measures. The staff also finds that the applicant clearly specified which responsibilities may not be delegated to other elements of the emergency organization, and has identified an adequate line of succession for the Emergency Coordinator position.

The staff reviewed WLS Emergency Plan, Table II-2, and other associated sections and figures as described above, and finds that the required minimum on-shift staff and augmentation staffing in support of WLS are acceptable because they are consistent with NUREG-0654, Table B-1 in and the guidance provided in Emergency Preparedness Position (EPPOS)-3[8], "On Requirement For Onshift Dose Assessment Capability."

Subject to License Condition (13-5) and License Condition (13-10), the staff finds that the applicant explicitly defined its responsibilities for emergency response, has adequate staffing to provide and maintain at all times initial facility accident response in key functional areas, including response to a hostile action, and is capable of timely augmentation of the response capabilities. In addition, the staff finds that the applicant adequately specified the interfaces among various onsite and offsite support and response activities; described the organization for coping with radiological emergencies, including the authorities, responsibilities, and duties of individuals assigned to the licensee's emergency organization and the means for their notification in the event of an emergency; and described the normal plant operating organization, the onsite ERO, and the headquarters and local offsite personnel and services that will augment and support the onsite organization. Licensee employees that are responsible for making offsite dose projections, and licensee and other persons with special qualifications for coping with emergency conditions, are also identified.

The staff finds the additional information and clarifications provided by the applicant in the December 23, 2008, response to RAI 25, Question 13.03-55(A), (M), (P)(P.2), and (Q) acceptable because they conform to the guidance in NUREG-0654. Accordingly, the staff considers these questions resolved.

Conclusion

Subject to License Condition (13-5) and License Condition (13-10), the staff concludes that the information provided in the WLS COLA conforms to the evaluation criteria in NUREG-0654, Planning Standard B. Therefore, the staff finds the information acceptable and meets the relevant requirements of 10 CFR 50.47(b)(2) and 10 CFR Part 50, Appendix E, Section IV.A, insofar as the information describes the applicant's on-shift responsibilities for emergency response, which are unambiguously defined; adequate staffing to provide initial facility accident response in key functional areas maintained at all times; timely augmentation of response capabilities; and interfaces among various onsite and offsite response support activities.

13.3.4.3 Emergency Response Support and Resources

As stated in NUREG-0654, Planning Standard C, "Emergency Response Support and Resources," 10 CFR 50.47(b)(3) requires that arrangements for requesting and effectively using assistance resources have been made, arrangements to accommodate State and local staff at the licensee EOF have been made, and other organizations capable of augmenting the planned response have been identified. In addition, 10 CFR Part 50, Appendix E, Section III requires

[8] EPPOS 3 (1995) ADAMS Accession No. ML023040473

that the emergency plans incorporate information about the emergency response roles of supporting organizations and offsite agencies, and that the information shall be sufficient to provide assurance of coordination among the supporting groups and with the licensee. 10 CFR Part 50, Appendix E, Section IV.A.7 requires identification of, and a description of the assistance expected from, appropriate local, State, and Federal agencies with responsibilities for coping with emergencies, including hostile action at the site.

In WLS Emergency Plan, Section II.C, "Emergency Response Support and Resources," the applicant addressed the responsibilities and concepts of operations for the various organizations that would support the Lee Nuclear Station in an emergency. The staff reviewed this section, as well as other relevant portions of the application, to determine whether the application conforms to the applicable guidance and complies with the pertinent regulatory requirements. The staff's primary focus was to evaluate the emergency plan against NUREG-0654, Planning Standard C, which provides the detailed evaluation criteria that the staff should consider to determine whether the emergency plan meets the applicable regulatory requirements in 10 CFR 50.47(b)(3).

WLS Emergency Plan, Section II.C.1 "Federal Response Capability," provides general information related to support expected from Federal Radiological Monitoring and Assessment Center (FRMAC), Department of Energy (DOE) Savannah River, DOE Oak Ridge and Radiation Emergency Assistance Center/Training Site (REAC/TS), and the NRC. WLS Emergency Plan, Section II.C.1.a, identifies that the EOF Director or Radiological Assessment Manager may request FRMAC assistance through the NRC for offsite radiological monitoring support. Additionally, DOE Savannah River may provide radiological monitoring assistance (DOE Radiological Assistance Program). DOE Oak Ridge may provide medical support from the REAC/TS. The FRMAC Advance Party could arrive at the Lee Nuclear Station within 3 to 4 hours following the order to deploy. Assistance from the NRC Region II office in Atlanta, Georgia, could arrive 7 to 8 hours following notification. The timeframe could be reduced if air travel were used.

WLS Emergency Plan, Section II.C.1.e states that facilities and resources needed to support the Federal response through the EOF will be provided. This includes office space and telephones. The applicant will also provide limited office space and telephone communication facilities for the NRC personnel in the TSC. In addition, WLS Emergency Plan, Section II.C.1.c, "Federal Response Capability," describes provisions for incorporating the Federal response capability into its operation plan, including specific licensee, State and local resources available to support the Federal response, (e.g., air fields, command posts, telephone lines, radio frequencies, and telecommunication centers).

WLS Emergency Plan, Section II.A.1.b, "Concept of Operations," provides general information related to assistance that will be provided from all Federal, State, and local agencies. WLS Emergency Plan, Section II.B.9, "Local Emergency Response Support," states that the applicant has established and maintains agreements with local emergency response support services. WLS Emergency Plan, Section II.E.1, "Notification of State and Local Authorities," provides an overview of the notification systems for prompt notification of State, local and Federal authorities. WLS Emergency Plan, Section II.L, "Medical and Public Health Support," discusses local hospital and medical support, including first aid and ambulance transport, and REAC/TS responsibilities during emergencies.

WLS Emergency Plan, Section II.C.2, “Off-site Organization Representation in the EOF,” states that designated work areas have been provided in the EOF for the State and county Emergency Management Liaisons and State Radiation Protection Liaisons.

WLS Emergency Plan, Section II.C.3, “Radiological Laboratories,” identifies radiological laboratories in South Carolina Departments of Health and Environmental Control, Bureau of Radiological Health; North Carolina Department of Environment and Natural Resources, Radiation Protection Section; and the DOE Radiological Assistance Team. Mobile monitoring and assessment capabilities in addition to fixed facilities for gross counting and spectral analysis are also identified. Other applicant facilities at the McGuire, Oconee, and Catawba Nuclear Stations are available to provide additional support within 1 to 4 hours, if needed. In RAI 25, Question 13.03-56(C), the staff requested that the applicant provide additional information related to the location of the station counting laboratory and when it will be used, the criterion that would be used to determine when the additional facilities would be needed, and the process for requesting additional support. In a December 23, 2008, response, the applicant stated that facilities used for health physics monitoring and assessment are discussed in AP1000 DCD Section 12.5, “Health Physics Facilities Design”; WLS COL FSAR Chapter 12, “Radiation Protection”; and WLS COL FSAR Section 13.1.1.2.4, “Chemistry.” The Radiological Assessment Manager, working with the EOF Director, determines staffing needs and assigns resources in support of efforts to coordinate radiological aspects of an emergency. The Radiological Assessment Manager also has the authority to seek assistance from other organizations within the applicant’s resources. Fixed radiological facilities at the Catawba, McGuire, and Oconee Nuclear Stations may also be used in an emergency.

WLS Emergency Plan, Section II.C.4, “Other Supporting Organizations,” describes arrangements with State and local emergency management authorities that establish cooperation for fire, medical and local law enforcement in accordance with 10 CFR Part 50, Appendix E, Section IV.A.7. A statewide mutual aid agreement in the South Carolina Emergency Response Plan provides coordination with State law enforcement for additional resources in the event of hostile action against the site and evacuation of the public. WLS Emergency Plan, Section II.C.4 also identifies additional emergency response support from: INPO Fixed Nuclear Facility Voluntary Assistance Agreement signatories and REAC/TS. Certification letters are provided in WLS Emergency Plan, Appendix 7, “Certification Letters.” LOAs for INPO or REAC/TS were not included. In RAI 25, Question 13.03-56(D), the staff requested that the applicant provide LOAs or other appropriate supporting documentation related to the emergency assistance provided by INPO and REAC/TS. In a December 23, 2008, response, the applicant stated LOAs with INPO and REAC/TS will be incorporated into WLS Emergency Plan, Appendix 7 in a future revision to the WLS Emergency Plan once they have been reached, or INPO and REAC/TS will be removed from WLS Emergency Plan, Section II.C.4, “Other Supporting Organizations.” In WLS COLA Part 10, the applicant identified License Condition 4, which includes a license condition regarding the schedule to update LOA’s. The staff identified this as License Condition (13-6). The staff has evaluated License Condition (13-6) in Section 13.3.4.16 of this report.

The staff finds the additional information and clarifications provided by the applicant in response to RAI 25, Question 13.03-56(C) and (D) acceptable because they conform to the guidance in NUREG-0654. Accordingly, the staff considers these questions resolved.

In its Interim Finding Report for Reasonable Assurance, FEMA found that the offsite emergency plans are adequate for this planning standard and the associated evaluation criteria in NUREG-0654.

The staff finds the information provided by the applicant and finds that the WLS Emergency Plan adequately describes the applicant's operational role, its concept of operations, and its relationship to the total effort. The staff finds this acceptable because it meets the applicable requirements in 10 CFR Part 50, Appendix E.

Conclusion

Subject to License Condition (13-6), the staff concludes that the information provided in the WLS COLA conforms to the evaluation criteria in NUREG-0654, Planning Standard C. Therefore, the staff finds the information acceptable and because it meets the relevant requirements of 10 CFR 50.47(b)(3) and 10 CFR Part 50, Appendix E, Sections III and IV.A.7, insofar as the information describes the arrangements for requesting and effectively using assistance resources; arrangements to accommodate State and local staff at the applicant's Emergency Operations Facility; and identification of other organizations capable of augmenting the planned emergency response.

13.3.4.4 Emergency Classification System

As stated in NUREG-0654, Planning Standard D, "Emergency Classification System," 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 that State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures. In addition, 10 CFR Part 50, Appendix E, Section IV.B requires a description of the means to be used for determining the magnitude, and for continually assessing the impact, of the release of radioactive materials, including emergency action levels (EALs) that are to be used as criteria for determining the need for offsite agency notifications and participation, and when and what types of protective measures should be considered. The EALs must include hostile actions that might adversely affect the nuclear power plant. The initial EALs shall be discussed and agreed upon by the applicant or licensee and State and local governmental authorities, and approved by the NRC. Thereafter, EALs shall be reviewed with State and local governmental authorities on an annual basis. 10 CFR Part 50, Appendix E, Section IV.C requires a description of EALs and emergency conditions that involve alerting or activating the total emergency organization, including communication steps to be taken under each emergency class. The emergency classes defined shall include (1) notification of unusual event, (2) alert, (3) site area emergency, and (4) general emergency. 10 CFR Part 50, Appendix E, Section IV.C.2 requires the capability to assess, classify, and declare an emergency condition within 15 minutes after the availability of indications to plant operators, which positions are defined in NSIR/DPR-ISG-01, that an EAL threshold has been exceeded, and to promptly declare the emergency conditions as soon as possible after the identification of the appropriate emergency classification level.

In WLS Emergency Plan, Section II.D, "Emergency Classification System," the applicant described the emergency classification and action level scheme used to determine the minimum response to an abnormal event at the plant. The staff reviewed this section, as well as other relevant portions of the application, to determine whether the application conforms to the applicable guidance and complies with the pertinent regulatory requirements. The staff's primary focus was to evaluate the emergency plan against NUREG-0654, Planning Standard D, which provides detailed evaluation criteria that the staff should consider to determine whether the emergency plan meets the applicable regulatory requirements in 10 CFR 50.47(b)(4).

In WLS Emergency Plan, Section II.D, "Emergency Classification System," the applicant described its capability to declare an emergency classification level within 15 minutes after the availability of the indications to trained and qualified staff that an emergency action level threshold has been exceeded are described in the emergency plan implementing procedures. The responsibilities for declaring emergencies at WLS are also included, as well as, the responsibility for terminating the emergency.

WLS Emergency Plan, Section II.D.1, "Classification System," includes a standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters. The following emergency classes are identified: "Notification Of Unusual Event," "Alert," "Site Area Emergency," and "General Emergency." The applicant also proposed EP-ITAAC Acceptance Criterion 1.1.1, which states, "The specific parameters identified in the Emergency Action Thresholds in the emergency plan implementing procedure addressing 'Emergency Classification' have been retrieved and displayed in the control room, TSC, and EOF." In addition, the applicant also proposed EP-ITAAC Acceptance Criterion 1.1.2, which states, "The ranges available in the control room, TSC, and EOF encompassed the values for the specific parameters identified in the Emergency Action Level Thresholds in the emergency plan implementing procedure addressing 'Emergency Classification.'"

For a COL application, the requisite EAL information is limited and the applicant is required to address four critical elements: (1) An overview of the EAL scheme, including a definition of the four emergency classification levels and general list of licensee actions; (2) a commitment to develop the remainder of the EAL scheme using a specified NRC-endorsed guidance document; (3) a proposed license condition that addresses EAL completion, agreement with State and local officials (as appropriate), and submission of the fully developed EALs to the NRC; and (4) maintaining the EALs in a document subject to 10 CFR 50.54(q). The information associated with these critical elements provides a sufficient level of applicable detail to support the staff's reasonable assurance evaluation.

In RAI 83, Question 13.03-75, the staff requested that the applicant submit either an entire EAL scheme or a revised WLS Emergency Plan, Section D, "Emergency Classification System" to address the four critical elements of the EAL scheme. In a June 12, 2009, response, the applicant provided a revised WLS Emergency Plan, Section D, and proposed a license condition to submit a fully developed set of site-specific EALs in accordance with the NRC-endorsed version of NEI 07-01, Revision 0, with no deviations. In WLS COLA Part 10, the applicant committed to meet EP-ITAAC (Table 3.8-1) and has proposed License Condition 4 "Emergency Planning Actions" related to the development and schedule for EALs. The staff refers to this as License Condition (13-3):

Proposed License Condition (13-3):

> The licensee shall submit a fully developed set of site-specific Emergency Action Levels (EALs) to the NRC in accordance with the NRC-endorsed version of NEI 07-01, Rev. 0, with no deviations. The EALs shall have been discussed and agreed upon with State

> and local officials. These fully developed EALs shall be submitted to the NRC for confirmation not less than 180 days prior to the date scheduled for initial fuel load.

The staff finds the description of the EAL scheme is acceptable because it is consistent with 10 CFR Part 50, Appendix E, Section IV.C, and addresses critical element (1). The staff considers the applicant's incorporation of the fully developed site-specific EAL scheme into implementing procedures acceptable because it ensures that the EALs are maintained in a document subject to the required change control process in 10 CFR 50.54(q) (i.e., EPIPs) and, therefore, addresses critical element (4). With regard to critical elements (2) and (3), in WLS COLA Part 10, the applicant proposed License Condition (13-3) (EALs), which includes a commitment to develop an EAL scheme with fully developed site-specific EALs in accordance with NRC-endorsed guidance document NEI 07-01, Revision 0. In addition, License Condition (13-3) requires a discussion and agreement with State and local officials, and submission of the fully developed EALs to the NRC. The EAL scheme is also addressed in NRC Bulletin 2005-02, "Emergency Preparedness and Response Actions for Security Based Events," (BL 2005-02), which requested, in part, that all holders of operating licenses provide information regarding the identification of emergency classification levels and EALs for security-based events. In NEI 07-01, Revision 0, the emergency classification scheme for security events, including hostile actions, is addressed in Section 5.9, "Hazards or Other Conditions Affecting Plant Safety EALs."

The staff reviewed License Condition (13-3) and, with the exception of the timeframe for submission of the EALs, finds that it is acceptable because it is consistent with NEI 07-01, Revision 0. The staff proposes a similar timeframe for submission of the EALs to the NRC, which is based on the date scheduled for initial fuel load set forth in the notification submitted in accordance with 10 CFR 52.103(a). Therefore, the staff identified License Condition (13-3), which includes the staff's proposed timeframe for submission of the EALs to the NRC.

License Condition (13-3):

> No later than 180 days before the date scheduled for initial fuel load set forth in the notification submitted in accordance with 10 CFR 52.103(a), Duke Energy shall submit a fully developed set of plant-specific emergency action levels (EALs) for WLS, Units 1 and 2, in accordance with NEI 07-01, "Methodology for Development of Emergency Action Levels – Advanced Passive Light Water Reactors," Revision 0, with no deviations. The EALs shall have been discussed and agreed upon with State and local officials.

The staff considers the proposed EAL scheme and License Condition (13-3) acceptable because they meet the requirements of 10 CFR Part 50, Appendix E and conform to the guidance provided in NUREG-0654.

Letters of Certification with State and local governments are included in WLS Emergency Plan, Appendix 7, "Certification Letters." These letters state that the signature on the letter indicates that the parties concurred with the emergency classification system, initiating conditions, and EALs for the Lee Nuclear Station. In RAI 83, Question 13.03-83, the staff requested that the applicant address when the initial EALs will be discussed and agreed upon with State and local governmental authorities. In a December 11, 2009, response, the applicant stated that in its proposed license condition, it would gain approval of the revised EAL scheme by local and State officials prior to submitting it to the NRC. In addition, in RAI 83, Question 13.03-78, the staff requested that the applicant provide a confirmation that the EAL scheme would be coordinated with State and local offsite response organizations. In a December 11, 2009, response, the applicant stated this would be accomplished through a proposed license condition, which is discussed above.

The staff finds the additional information and clarifications provided by the applicant in response to RAI 83, Questions 13.03-75, 13.03-78, and 13.03-83 acceptable because they conform to the guidance in NUREG-0654. Accordingly, the staff considers these questions resolved.

In its Interim Finding Report for Reasonable Assurance, FEMA found that the offsite emergency plans are adequate for this planning standard and the associated evaluation criteria in NUREG-0654.

Subject to License Condition (13-3), the staff finds that the applicant established a standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, which includes the four emergency classes identified above. The applicant described EALs and emergency conditions that involve ERO activation, including steps to be taken under each emergency class. The applicant also described the means to determine the magnitude of, and for continually assessing the impact of, the release of radioactive materials, and EALs (including those pertaining to hostile actions) that are used to determine the need for offsite notifications and protective measures. In addition, the applicant has the capability to assess, classify, and declare an emergency condition within 15 minutes after the availability of indications to plant operators that an EAL threshold has been exceeded, and to promptly declare the emergency condition.

Conclusion

Subject to License Condition (13-3), the staff concludes that the information provided in the WLS COLA is consistent with the guidelines in NUREG-0654, Planning Standard D. Therefore, the staff finds the information acceptable and that it meets the relevant requirements of 10 CFR 50.47(b)(4) and 10 CFR Part 50, Appendix E, Sections IV.B and IV.C, insofar as the information describes the essential elements of advanced planning and the provisions made to cope with emergency situations.

13.3.4.5 Notification Methods and Procedures

As stated in NUREG-0654, Planning Standard E, “Notification Methods and Procedures,” 10 CFR 50.47(b)(5) requires that procedures have been established for notification, by the licensee, of State and local response organizations and for notification of emergency personnel by all organizations; the content of initial and follow-up messages to response organizations and the public has been established; and that the means to provide early notification and clear instruction to the populace within the 16-km (10-mi) plume exposure pathway EPZ have been established. In addition, 10 CFR Part 50, Appendix E, Section IV.A.4 requires a description of how offsite dose projections will be made and the results transmitted to State and local authorities, NRC, and other appropriate governmental entities. 10 CFR Part 50, Appendix E, Section IV.C requires a description of EALs and emergency conditions that involve alerting or activating the emergency organization, including communication steps to be taken under each class of emergency, and the existence of a message-authentication scheme. 10 CFR Part 50,

Appendix E, Section IV.D.1 requires a description of administrative and physical means for notifying local, State, and Federal officials and agencies and agreements reached with these officials and agencies for the prompt notification of the public and for public evacuation or other protective measures. The description shall include identification of the appropriate officials, by title and agency, of the State and local government agencies within the EPZs. 10 CFR Part 50, Appendix E, Section IV.D.3 requires the licensee to have the capability to notify responsible State and local governmental agencies within 15 minutes after declaring an emergency. The licensee shall demonstrate that appropriate governmental authorities have the capability to make a public alerting and notification decision promptly on being informed by the licensee of an emergency condition, and that administrative and physical means have been established for alerting and providing prompt instructions to the public within the plume exposure pathway EPZ. The alerting and notification capability shall include a backup method. Finally, 10 CFR 50.72(a)(3) requires NRC notification no later than 1 hour after declaring an emergency.

NRC notifications are further addressed in Commission Orders issued on February 25, 2002, as well as BL 2005-02, which requested in part that all holders of operating licenses provide information regarding the implementation of an NRC notification time period of approximately 15 minutes from discovery of a security-based event, as well as subsequent NRC guidance.

In WLS Emergency Plan, Section II.E, “Notification Methods and Procedures,” the applicant described the specific methods and sequencing of notifications that will be covered in the appropriate implementing procedures for WLS in an emergency. The staff reviewed this section, as well as other relevant portions of the application, to determine whether the application conforms to the applicable guidance and complies with the pertinent regulatory requirements. The staff’s primary focus was to evaluate the emergency plan against NUREG-0654, Planning Standard E, which provides the detailed evaluation criteria that the staff should consider to determine whether the emergency plan meets the applicable regulatory requirements in 10 CFR 50.47(b)(5).

WLS Emergency Plan, Section II.E, “Notification Methods and Procedures,” states that onsite emergencies are immediately reported to the Shift Manager on duty. Offsite response is the responsibility of local government officials in accordance with the State plans. Procedures for notification of State and local response organizations and licensee emergency responders reference the prepared messages in the State plans. Notification is initiated by the Emergency Coordinator within 15 minutes of emergency declaration based on EALs. All affected organizations (warning points) are listed. The NRC is notified following notification of State and local authorities and within 1 hour of declaration of an emergency. The notification system consists of a primary and a back-up system maintained through the use of commercial telephones (Section II.F.1, “Description of Communications Links”). In RAI 25, Question 13.03-58(B) and RAI 83, Question 13.03-79, the staff requested that the applicant specify the “officials” to be notified as described in 10 CFR Part 50, Appendix E, Section IV.D.1 and describe procedures and physical means for making notifications to offsite agencies. In a December 11, 2009, response, the applicant explained that North Carolina, South Carolina, and county warning point Cherokee County, York County and Cleveland County duty officers are notified via the Selective Signaling System within 15 minutes of a declared emergency at the site. The State and county warning point duty officers will then notify State and local officials according to their respective procedures. This process and the procedures have been cooperatively developed between the States of North and South Carolina and the counties of

Cherokee, York, and Cleveland. The applicant proposed EP-ITAAC 2.1 to test the emergency notification capabilities.

WLS Emergency Plan, Section II.E.1, “Notification of State and Local Authorities,” states that systems and procedures needed to provide prompt notification of the NRC Operations Center following the declaration of any emergency condition are maintained. The NRC is notified as soon as is practical following the notification of State and local authorities and within 1 hour of the emergency declaration, including escalation or de-escalation of any emergency declaration. WLS Emergency Plan, Section II.A.1.b, “Assignment of Responsibility,” footnote #4, states that in the event of a security-related attack on the site by a hostile force, a brief notification (site name, emergency classification, if determined, and nature of threat) is provided to the NRC following notification of the designated State and local authorities and within approximately 15 minutes of the discovery of the event.

WLS Emergency Plan, Section II.E.2, “Notification and Mobilization of Licensee Response Organizations,” is directed by the Emergency Coordinator. WLS has an evacuation alarm and a Telephone/Page System. There is redundant notification through the paging system and an automated telephone system. A siren tone generator and public address system speakers can be activated from the control room in case of emergency (see AP1000 DCD Section 9.5.2.2, “Communications Systems-System Design”). ERO personnel are notified by alpha-numeric pagers following procedures in the EPIPs. The applicant proposed EP-ITAAC 2.2 to test the capabilities of the system used to notify licensee response personnel.

WLS Emergency Plan, Section II.E.3, “Message Content,” states, “The content of the messages has been established in conjunction with the State and local governments and include the class of emergency, whether a release is in progress, and potentially affected areas and populations, and any recommended protective measures. Additional information is provided as it becomes available.”

In RAI 25, Question 13.03-58(A), the staff requested that the applicant provide documentation detailing the notification process. In a December 23, 2008, response, the applicant stated the Emergency Coordinator provides emergency notification directly to the State and county governments through the Selective Signaling Telephone System discussed in WLS Emergency Plan, Section II.F. Emergency notification forms are transmitted to the 24-hour warning points in North Carolina and South Carolina as soon as they are online and hourly updates are provided throughout the emergency. Warning points implement their respective emergency plans and notify the appropriate State or local officials specified in their plans once notified. Commercial and satellite phones can be used as backup.

WLS Emergency Plan, Section II.E.4, “Follow-up Messages to Off-site Authorities,” states that there are dedicated communicators for continuous communication allowing regular updates. Communication with designated authorities is to be continuous with the NRC, and provides updates approximately every 60 minutes to State and local authorities. Follow-up messages shall include all information listed in NUREG-0654, Evaluation Criterion E.4.a-n, (as appropriate). In RAI 25, Question 13.03-58(D), the staff requested that the applicant provide information identifying the communicators, where they will be located during an emergency, and how they will obtain the necessary information for the follow-up messages. In a December 23, 2008, response, the applicant stated that the Selective Signaling Telephone System is used for follow-up communication with State and local authorities as described in WLS Emergency Plan, Section II.F.1.b and the response to RAI 25, Question 13.03-58(A). Communication support is provided for by communicators in the TSC or EOF. Follow-up communication during notification of an unusual event is provided by the Control Room (CR).

WLS Emergency Plan, Section II.E.6, “Instructions to the Public in the Plume Exposure EPZ (Emergency Planning Zone),” states that the Alert and Notification System (ANS) is used and that it includes an outdoor acoustic warning system designed to meet the acceptance criteria of NUREG- 0654/FEMA-REP-1, Revision 1, as revised by Supplement 4 (October 2011) Appendix 3, “Means for Providing Prompt Alerting and Notification of Response Organizations and the Population,” Section B, “Criteria for Acceptance,” The physical description of the primary and backup systems is detailed in a FEMA-approved ANS design report. The design objective of the primary system is to have the capability to essentially complete the initial alerting and initiate notification of members of the public within the plume exposure pathway EPZ, including those in remote and low population areas within 15 minutes following a decision by cognizant offsite agencies to notify the public. The capability includes any transient populations in remote and rural areas, open water, rivers, hunting, recreational and low population areas that may require special alerting procedures. Furthermore, as a back-up, State and local plans maintain the alert mechanism via systems such as emergency vehicles, automated dialing systems, and Public Alerting (PA) systems to also alert the public to monitor commercial broadcasts for emergency information. The primary and back-up alert systems may include any combination of fixed sirens, tone alert radios, National Oceanographic and Atmospheric Agency (NOAA) weather radios or route alerting. The notification systems may consist of a combination of Emergency Alert System (EAS), NOAA weather radios or route alerting. Each county controls the activation of the sirens within its boundaries. Individuals by title that will initiate alarm are listed in the referenced State plans. The applicant proposed EP-ITAAC 2.3 and EP-ITAAC Acceptance Criteria 8.1.1.2.B.4 to confirm the means to notify and provide instructions to the populace in the plume exposure pathway EPZ.

WLS Emergency Plan, Section II.E.7, “Written Messages to the Public,” states that written pre-scripted messages are released to the local media by the State Director of Emergency Management or local Director of Emergency Management. The messages give instruction to specific actions to be taken, the nature of the emergency and recommended protective actions, including sheltering, evacuation, and the use of potassium iodide, as appropriate. The WLS Emergency Plan also states that the applicant will assist with the development of the messages. In RAI 25, Question 13.03-58(E), the staff requested that the applicant provide details related to the applicant’s support for written messages to the public. In a December 23, 2008, response, the applicant stated that the EOF News Manager manages the communication organization, which is responsible for coordinating plant status updates to State and local authorities and the media. The applicant provides detailed information regarding the emergency to support the preparation of the emergency messages. The applicant also provided Duke Energy’s corporate procedure for emergency communication between corporate and the EOF as an example. This example was reviewed by the NRC staff and found acceptable. Actual procedures for WLS will be developed and submitted in accordance with License Condition (13-8). This License Condition is evaluated in Section 13.3.4.19 of this report.

NRC notifications are further addressed in BL 2005-02, which requested in part that all holders of operating licenses provide information regarding the implementation of an NRC notification time period of approximately 15 minutes from discovery of a security-based event.

WLS Emergency Plan, Section II.E.1, "Notification of State and Local Authorities," states that systems and procedures needed to provide prompt notification of the NRC Operations Center following the declaration of any emergency condition are maintained. The NRC is notified as soon as is practical following the notification of State and local authorities and within 1 hour of the emergency declaration, including escalation or de-escalation of any emergency declaration.

WLS Emergency Plan, Section II.A.1.b, "Assignment of Responsibility," footnote #4, states that in the event of a security-related attack on the site by a hostile force, a brief notification (site name, emergency classification, if determined, and nature of threat) is provided to the NRC following notification of the designated State and local authorities and within approximately 15 minutes of the discovery of the event.

WLS Emergency Plan, Section II.F, "Emergency Communications," states that dedicated communicators are available to maintain a continuous channel of communication with the NRC as requested and to provide regular updates to State and local officials approximately every 60 minutes, when conditions change, or as otherwise agreed.

The staff finds the additional information and clarifications provided by the applicant in response to RAI 25, Questions 13.03-58(A), (B), (D) and (E), and RAI 58, Question 13.03-79 acceptable because they conform to the guidance in NUREG-0654. Accordingly, the staff considers these questions resolved.

In its Interim Finding Report for Reasonable Assurance, FEMA found that the offsite emergency plans are adequate for this planning standard and the associated evaluation criteria in NUREG 0654.

After reviewing information provided by the applicant, the staff finds that procedures for notification of State and local response organizations and emergency personnel by all organizations have been established, and the licensee has the capability to notify offsite officials and agencies, including State and local governmental agencies within 15 minutes, and NRC no later than 1 hour after declaring an emergency. The appropriate officials of the States and local government agencies within the EPZs have been identified. The licensee has described the entire spectrum of emergency conditions that involve alerting or activating the emergency organization, including EALs for offsite agency notification and communication steps to be taken under each class of emergency. Message authentication is described in the States' and local emergency plans. The applicant has also described how appropriate governmental authorities have the capability to make a public alerting and notification decision promptly following notification of an emergency by the licensee, and administrative and physical means have been established for alerting and providing prompt instruction to the public within the plume exposure pathway EPZ (including a backup methods to alert populations), and for public evacuation and other protective measures.

Conclusion

The staff concludes that the information provided in the WLS COLA is consistent with the guidelines in NUREG-0654, Planning Standard E. Therefore, the staff finds the information acceptable and that it meets the relevant requirements of 10 CFR 50.47(b)(5), 10 CFR 50.72(a)(3), and 10 CFR Part 50, Appendix E, Sections IV.A.4, IV.C, IV.D.1, and IV.D.3, insofar as the information describes the essential elements of advanced planning and the provisions made to cope with emergency situations.

13.3.4.6 Emergency Communications

As stated in NUREG-0654, Planning Standard F, "Emergency Communications," 10 CFR 50.47(b)(6) requires that provisions exist for prompt communication among principal response organizations, to emergency personnel, and to the public. In addition, 10 CFR Part 50, Appendix E, Section IV.E.9 requires onsite and offsite communication systems with backup power sources, including provisions for communication with State and local governments within the plume exposure pathway EPZ, and Federal emergency response organizations and the NRC. Also required are provisions for communication among the Control Room, TSC, EOF, principal State and local emergency operations centers (EOCs), and field assessment teams. Communication systems shall be tested at designated frequencies.

WLS Emergency Plan, Section II.F, "Emergency Communications," describes the station communication systems. The staff reviewed this section, as well as other relevant portions of the WLS COL application, to determine whether the application conforms to the applicable guidance and complies with the pertinent regulatory requirements. The staff's primary focus was to evaluate the emergency plan against NUREG-0654, Planning Standard F, which provides the detailed evaluation criteria that the staff should consider to determine whether the emergency plan meets the applicable regulatory requirements in 10 CFR 50.47(b)(6).

WLS Emergency Plan, Section II.F, "Emergency Communications," states that the applicant maintains systems and procedures that provide for prompt communication between its ERFs and between the site and offsite ERFs. Dedicated communicators are available to maintain continuous communication with the NRC and to provide regular updates to State and local officials approximately every 60 minutes, when conditions change or as otherwise agreed.

The communication systems are designed to provide redundant means to communicate with essential areas of the station during normal operation and under accident conditions. Communication systems vital to operation and safety are designed so that failure of one component would not impair the reliability of the total communication systems. This is accomplished within the station by using diverse systems. The EPIPs define the responsibilities of designated personnel for use of the communication systems.

The communication systems include those systems described in AP1000 DCD Section 9.5.2, "Communication System," and the following emergency communication systems:

- Wireless telephone system
- Telephone/page system
- Private automatic branch exchange (PABX) system
- Emergency offsite communication
- Security communication system

In the event of a natural disaster, WLS maintains a satellite phone system described in WLS COL FSAR Section 9.5.2.2.3.1.3.

WLS Emergency Plan, Section II.F, "Emergency Communications," states that responsibilities of designated personnel for the communication systems can be found in State and local plans and in the EPIPS. The station maintains capabilities for 24 hours per day emergency notification to the State and county emergency response network. All State and county warning points are staffed 24 hours per day.

WLS Emergency Plan, Section II.F.1.a, "Description of Communication Links," states that the applicant maintains capabilities for 24 hours per day emergency notification to the State and county emergency response network. In RAI 25, Question 13.03-59(D), the staff requested that the applicant discuss who is designated to use communication systems and what responsibilities they have for using those communication systems. In a December 23, 2008, response, the applicant stated that a communicator will be assigned by the Operations Shift Manager/Emergency Coordinator from the on shift staff. The position will be filled by a Control Room Operator or Non-Licensed Operator from the unaffected unit that has been trained to perform this function. Full-time communication positions in the ERO include the TSC Off-site Agency Communicator, the EOF Off-site Agency Communicator, and the NRC Communicator.

WLS Emergency Plan, Section II.F.1.b, "Description of Communications Links," identifies communication links (EOF to State and county warning points). The applicant proposed EP-ITAAC 3.1 to test that the means exist for communication among the CR, TSC, EOF, principal State and local EOCs, and radiological field assessment teams. The applicant proposed ITAAC 3.2 to test the communication capabilities from the control room, TSC, EOF to the NRC Headquarters Operation Center (HOC) and Region II EOC, which also addresses the establishment of the availability of an access port for ERDS and transfer of data from the units to the NRC HOC.

WLS Emergency Plan, Section II.F.1.c, "Description of Communications Links," identifies dedicated communication links with the NRC through Emergency Notification System (ENS), Health Physics Network (HPN), Reactor Safety Counterpart Link (RSCL), Protective Measures Counterpart Link (PMCL), ERDS, Management Counterpart Link (MCL), and Local Area Network (LAN) systems.

WLS Emergency Plan, Section II.F.1.d, "Description of Emergency Communications Links," states that the applicant provides capability for communication between CR or TSC and the EOF, county, and State EOCs. This section states that communication between the TSC, EOF and offsite monitoring teams is via radio. In RAI 25, Question 13.03-59(C), the staff requested that the applicant provide clarification regarding the testing frequency from the licensee to the NRC Headquarters and appropriate NRC Regional Office Operations Center. In a December 23, 2008, response, the applicant revised Section II.N.2.a to clarify communication systems testing between the facility, NRC Headquarters, and NRC Regional Operations will be performed monthly.

WLS Emergency Plan, Section II.F.1.e, "Description of Communications Links," refers back to WLS Emergency Plan, Section II.E.2, "Notification and Mobilization of Licensee Response Organizations," for notification, alerting, and activation of emergency response personnel in the TSC, OSC, and EOF.

WLS Emergency Plan, Section II.F.1.f describes that WLS communicates with the NRC Headquarters Operations Center by the Emergency Telecommunications System (ETS) located in the CR, TSC and EOF or by private telephone systems, and between the CR, TSC and EOF to the NRC regional office by the normal private telephone capability. The applicant maintains radio communication between the TSC, EOF and offsite radiological monitoring teams.

WLS Emergency Plan, Section II.F, “Emergency Communications,” provides communication system descriptions but does not identify communication between the licensee and Federal EROs other than NRC. Additional information related to communication between the licensee and Federal EROs was requested in RAI 25, Question 13.03-59(B). In a December 23, 2008, response, the applicant stated that the Radiological Assessment Manager may contact the DOE-Savannah River and/or REAC/TS for radiological monitoring assistance as discussed in WLS Emergency Plan, Section II.C.1.b. The NRC is the primary interface for communication with other Federal agencies. WLS Emergency Plan, Section II.N.2.a, “Communications Drills,” states that communication testing with NRC Headquarters’ and Region’s Operation Centers is tested monthly.

The applicant’s communication plans have arrangements for emergencies, including titles and alternates for those in charge at both ends of the communication links and the primary and backup means of communication.

WLS Emergency Plan, Section II.F.2, “Communication with Fixed and Mobile Medical Support Facilities,” states Duke Energy maintains radio and telephone communication systems that allow for communication between Lee Nuclear Station and fixed and mobile medical support facilities.

WLS Emergency Plan, Section II.F.3, “Communication System Reliability,” discusses system reliability. WLS COL FSAR Section 9.5.2.2.3.1.1, “NRC Offsite Interfaces,” states the design addresses the recommendations of NRC Bulletin 80-15, “Possible Loss of Emergency Notification System (ENS) with Loss of Offsite Power.” WLS Emergency Plan, Section F, “Emergency Communications,” states, “The communications systems include those systems described in Section 9.5.2 of the AP1000 DCD,” and that a normal source power supply failure does not impact offsite communication systems since backup power sources are provided, in most cases. Furthermore, the section states that onsite communication systems are periodically tested and that dedicated telephone lines are checked according to specified schedules.

In RAI 25, Question 13.03-59(A), the staff requested that the applicant clarify the use of backup power. In a December 23, 2008, response, the applicant stated that systems are maintained to communicate within the station and offsite as discussed in WLS COL FSAR Section 9.5.2.2.3, “Private Automatic Branch Exchange System.” The selective signaling system is used as the primary means of communication between the station and offsite agencies. The system has sufficient backup power sources with automatic transfer capability to maintain communication if power is lost. Commercial telephone company lines and the Duke Energy Radio network can be used as secondary means of communication.

WLS Emergency Plan, Section II.N.2.a, “Communications Drills,” states that communication testing with State and local governments within the 16-km (10-mi) EPZ is performed monthly; communication testing with State and Federal EROs within the 80-km (50-mi) EPZ and outside the 16-km (10-mi) EPZ is performed quarterly; and from WLS to NRC Headquarters Operations Center and to the Region Operations Center monthly. Additionally, Duke Energy conducts a communication drill annually. The drills' acceptance criteria are established for both operability of the systems and intelligibility of the messages. WLS Emergency Plan, Appendix 8, "Cross References to Regulations, Guidance, and State and Local Plans," provides a cross reference between the WLS Emergency Plan and the State and local plans.

In WLS COLA Part 10, the applicant proposed License Condition 12C, "Emergency Planning Actions," to address the NTTF Recommendation 9.3 for communication capabilities. With respect to communications, the staff refers to this as License Condition (13-11):

Proposed License Condition (13-11):

> Communications –
>
> At least two (2) years before the latest date set forth in the schedule for completing the inspections, tests, and analyses in the ITAAC submitted in accordance with 10 CFR 52.99(a), the Licensee shall have performed an assessment of on-site and off-site communications systems and equipment relied upon during an emergency event to ensure communications capabilities can be maintained during an extended loss of ac power. The communications capability assessment shall be performed in accordance with NEI 12-01, "Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities," Revision 0.
>
> At least one hundred eighty (180) days before the date scheduled for initial fuel load set forth in the notification submitted in accordance with 10 CFR 52.103(a), the Licensee shall complete implementation of corrective actions identified in the communications capability assessment described above, including any related emergency plan and implementing procedure changes and associated training.

With the exception of the timeframes for completion and submission of the communication capability assessment, the staff finds the license condition acceptable because it is consistent with NTTF Recommendation 9.3 and reflects the use of (NEI technical report NEI 12-01, "Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities," which the NRC has endorsed as an acceptable method for licensees to employ when addressing NTTF Recommendation 9.3.[9]

The staff proposes a similar timeframe for completion of the communication capabilities assessment as for the staffing assessment in License Condition (13-10). Therefore, consistent with the applicant's proposed License Condition (13-11), the staff identified License Condition

[9] See (1) NRC May 15, 2012, letter, 'U.S. Nuclear Regulatory Commission Review of NEI 12-01, "Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities," Revision 0, May 2012' (ADAMS Accession No. ML12131A043); (2) NEI May 3, 2012, letter, 'Transmittal of NEI 12-01, "Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities," Revision 0, dated May 2012' (ADAMS Accession No. ML12125A411); and (3) NEI Report No. 12-01, Revision 0, "Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities," May 2012 (ADAMS Accession No. ML12125A412).

(13-11), which addresses enhanced communication capabilities and includes the timeframes for completion of the assessments and their submission to the NRC.

License Condition (13-11):

> At least 18 months before the latest date set forth in the schedule submitted in accordance with 10 CFR § 52.99(a) for completing the inspections, tests, and analyses in the ITAAC, Duke Energy shall have performed an assessment of onsite and offsite communications systems and equipment relied upon during an emergency event to ensure communication capabilities can be maintained during an extended loss of alternating current power. The communications capability assessment shall be performed in accordance with NEI 12-01, "Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities," Revision 0.
>
> At least one hundred eighty (180) days before the date scheduled for initial fuel load set forth in the notification submitted in accordance with 10 CFR § 52.103(a), Duke Energy shall complete implementation of corrective actions identified in the communications capability assessment described above, including any related emergency plan and implementing procedure changes and associated training.

The staff finds License Condition (13-11), as modified, acceptable because it meets the requirements of 10 CFR Part 50, Appendix E and conforms to the guidance provided in NUREG-0654/FEMA REP 1, Revision 1 and in NEI 12-01, Revision 0.

After review, the staff finds the additional information and clarifications provided by the applicant in the December 23, 2008, response to RAI 25, Question 13.03-59 (A through D) acceptable because it conforms to the guidance in NUREG-0654.

In its Interim Finding Report for Reasonable Assurance, FEMA found that the offsite emergency plans are adequate for this planning standard and the associated evaluation criteria in NUREG-0654.

Subject to License Condition (13-11) as modified, and after review of information provided by the applicant, the staff finds that provisions exist for prompt communication among principal response organizations, to emergency personnel, and to the public. Specifically, the applicant established a reliable primary and backup means of communication for alerting and activating the response organizations and personnel, including 24-hour staffing of communication links. Provisions also exist for communication among the Control Room, TSC, EOF, State and local governments within the EPZs, and field assessment teams. In addition, the applicant provided a coordinated communication link for fixed and mobile medical support facilities. Onsite and offsite communication systems have backup power sources and are tested at designated frequencies.

Conclusion

Subject to License Condition (13-11), the staff concludes that the information provided in the COLA is consistent with the guidelines in NUREG-0654, Planning Standard F, and in NEI 12-01, Revision 0. Therefore, the staff finds the information acceptable and meets the relevant requirements of 10 CFR 50.47(b)(6) and 10 CFR Part 50, Appendix E, Section IV.E.9, insofar as the information describes the essential elements of advanced planning and the provisions made to cope with emergency situations.

13.3.4.7 Public Education and Information

As stated in NUREG-0654, Planning Standard G, “Public Education and Information,” 10 CFR 50.47(b)(7) requires that information be made available periodically to the public concerning notification methods and initial actions the public should take in an emergency, for example, 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) be established in advance, and procedures for coordinating dissemination of information to the public be established. In addition, 10 CFR Part 50, Appendix E, Section IV.D.2 requires a description of provisions for yearly dissemination to the public within the plume exposure pathway EPZ of basic emergency planning information, such as methods for public notifications and protective actions planned if an accident occurs, general information as to the nature and effects of radiation, and a listing of local broadcast stations that will be used for dissemination of information during an emergency. Signs or other measures shall also be used to disseminate information to any transient population within the plume exposure pathway EPZ.

In WLS Emergency Plan, Section II.G, “Public Education and Information,” the applicant described the public education and information program for the WLS, including the process for keeping the public within the 16-km (10-mi) EPZ informed in the event of an emergency. The staff reviewed this section, as well as other relevant portions of the application, to determine whether the application conforms to the applicable guidance and complies with the pertinent regulatory requirements. The staff’s primary focus was to evaluate the emergency plan against NUREG-0654, Planning Standard G, which provides the detailed evaluation criteria that the staff should consider to determine whether the emergency plan meets the applicable regulatory requirements in 10 CFR 50.47(b)(7). The applicant commits to coordinating with the State and local authorities to disseminate information to the public on responding to a radiological emergency at the WLS.

WLS Emergency Plan, Section II.G.1, “Public Information Program,” states that information provided to the public includes educational information on radiation, point of contact for additional information, protective measures (evacuation routes, relocation centers, sheltering, respiratory protection, etc.) and information addressing special needs of the handicapped on an annual basis.

WLS Emergency Plan, Section II.G.2, “Distribution and Maintenance of Public Information,” lists how written information may be provided to permanent residences and transient populations. WLS COL FSAR Section II.G.2 states that information for transient populations, which may be staying in hotels, motels and campgrounds, will be provided by public postings and publications, which will provide local information sources for emergencies. However, the WLS Emergency Plan does not address who will be responsible for creating the material and having the material disseminated. In RAI 25, Question 13.03-60(A), the staff requested that the applicant identify the individual responsible for coordinating and disseminating educational information for the general public with State and local authorities and discuss the process. In a December 23, 2008, response, the applicant stated that the Emergency Communications Manager is responsible for operation and maintenance of the JIC, and coordinating the creation and distribution of public informational materials in cooperation with State and local authorities for the WLS. Educational material is distributed to commercial and residential addresses within the plume exposure pathway EPZ annually. The applicant provided examples of this material used at the Catawba site as examples of the types of public information material that would be developed and distributed for the WLS. Public education material for Duke Energy's operating nuclear plants is also available via the Duke Energy Nuclear Emergency Preparedness website. The applicant further stated that details regarding the creation and distribution of public information materials will be developed on a schedule that supports NRC inspection activities and execution of the emergency exercise required by 10 CFR Part 50, Appendix E, Section IV.F.2.

WLS Emergency Plan, Section II.G.3, “News Media Coordination states that the JIC is located in the Energy Center located in Charlotte, North Carolina. The section also indicates that the News Manager and Public Spokesperson are the primary contacts for the news media. The JIC, co-located with the EOF is where approved news releases will be provided to the media for dissemination to the public. The JIC is equipped with appropriate seating, lighting, and visual aids to allow for public announcements and briefings to be given to the news media. The JIC is activated at the declaration of an “Alert” or higher classification. Additionally, WLS Emergency Plan, Section II.G.3 states that an onsite media center can be promptly established and provide space for a limited number of media. The applicant has proposed EP-ITAAC 4.1 to ensure that the licensee has provided space that may be used for a limited number of the news media.

WLS Emergency Plan, Section II.B, “On-site Emergency Organization,” Figure II-3, “Off-site Emergency Response Organization,” shows the JIC organization as part of the EOF Director’s span of control.

WLS Emergency Plan, Section II.G.4.a, “Information Exchange,” indicates that the public spokesperson has access to all the required information related to the emergency and provides plant status information during news conferences and briefings. The Chief Nuclear Officer and his direct reports are the designated public spokespersons.

WLS Emergency Plan, Section II.G.4.b, “Information Exchange,” states that liaisons coordinate with licensee and designated members of the State and local EROs on a periodic basis. WLS Emergency Plan, Appendix 9, “Justification for Common EOF,” states that the applicant and state officials cooperate in releasing information to the media. In RAI 25, Question 13.03-60(B), the staff requested that the applicant provide details on how timely and accurate information is provided to the media during an emergency. In a December 23, 2008, response, the applicant stated that WLS Emergency Plan, Sections II.G.3, “News Media Coordination,” and II.G.4, “Information Exchange,” address arrangements for news media coordination and the exchange of information among designated spokespersons. The applicant has specified that Duke Energy has corporate procedures are in place, which describe the public information responsibilities of emergency response personnel. The Chief Nuclear Officer and his or her direct reports fill the position of public spokesperson to provide plant status and company information during news conferences and media briefings conducted on an hourly basis. The applicant committed to revise their corporate JIC Activation Procedure to incorporate the WLS on a schedule to support the requirements of 10 CFR Part 50, Appendix E. The applicant proposed EP-ITAAC 9.1 to verify that the emergency plan implementing procedures provide for 24-hour per day emergency response, continuous staffing of communication links and operations for a protracted period.

WLS Emergency Plan, Section II.G.4.c, “Information Exchange,” states that contact between the designated spokespersons and by the activities of a licensee liaison in the JIC serves to control rumors. Customer inquiries are handled by Customer Contact Centers. Employees are updated through company intranet. Elected officials and regulatory agencies are updated through the Corporate Communications and Governmental Affairs Departments. Industry groups assist in disseminating information to other industry groups.

WLS Emergency Plan, Section II.G.5, “News Media Training,” states that information regarding emergency plans and radiation hazards, and points of contact for release of public information is provided annually to media organizations.

The staff finds the additional information and clarifications provided by the applicant in the December 23, 2008, response to RAI 25, Question 13.03-60 (A) and (B) acceptable because they conform to the guidance in NUREG-0654.

In its Interim Finding Report for Reasonable Assurance, FEMA found that the offsite emergency plans are adequate for this planning standard and the associated evaluation criteria in NUREG-0654.

The staff finds that the applicant has provided for a coordinated and periodic dissemination of information to the public, including the permanent and transient population within the 16-km (10-mi) EPZ, regarding how they will be notified and what their actions should be in an emergency. The applicant has also established the principal points of contact with the news media for dissemination of information during an emergency, and procedures for coordinated dissemination of information to the public. In addition, the applicant has described the provisions for yearly dissemination to the public within the plume exposure pathway EPZ of basic emergency planning information, including the use of signs or other measures to disseminate information to any transient population within the plume exposure pathway EPZ.

Conclusion

The staff concludes that the information provided in the WLS COLA is consistent with the guidelines in NUREG-0654, Planning Standard G. Therefore, the staff finds the information acceptable and meets the relevant requirements of 10 CFR 50.47(b)(7) and 10 CFR Part 50, Appendix E, Section IV.D.2, insofar as the information describes the essential elements of advanced planning and the provisions made to cope with emergency situations.

13.3.4.8 Emergency Facilities and Equipment

As stated in NUREG-0654, Planning Standard H, “Emergency Facilities and Equipment,” 10 CFR 50.47(b)(8) requires that adequate emergency facilities and equipment to support the emergency response be provided and maintained. In addition, 10 CFR Part 50, Appendix E, Section IV.E.8 requires that adequate provisions be made and described for emergency facilities and equipment, including a licensee’s onsite OSC and TSC, as well as an EOF from which effective direction can be given and effective control can be exercised during an emergency. 10 CFR Part 50, Appendix E, Section IV.E.8.b addresses various requirements associated with EOF locations and required provisions, which are not applicable to an existing EOF pursuant to 10 CFR Part 50, Appendix E, Section IV.E.8.e. 10 CFR Part 50, Appendix E, Section IV.E.8.c requires various EOF capabilities, which include supporting response to multiple reactors/sites and simultaneous events, as applicable. 10 CFR Part 50, Appendix E, Section IV.E.8.d requires an alternative facility (for use when onsite emergency facilities cannot be safely accessed during hostile actions) that would be accessible and could function as a staging area for augmentation of emergency response staff. 10 CFR Part 50, Appendix E, Section IV.G requires a description of provisions to be employed to ensure that the emergency plan, its implementing procedures, and emergency equipment and supplies are maintained up to date. 10 CFR Part 50, Appendix E, Section VI.1 requires an ERDS data link between the licensee's onsite computer system and the NRC Operations Center, through which a limited data set of selected parameters can be automatically transmitted.

In WLS Emergency Plan, Section II.H, "Emergency Facilities and Equipment," Appendix 10, "Technical Support Center Description," and Appendix 9, "Justification for Common EOF," the applicant described the ERFs and the equipment that will be used to assess an accident and monitor functions following the declaration of an emergency. The staff reviewed the section and appendices in the WLS Emergency Plan, as well as other relevant portions of the application, to determine whether the application conforms to the applicable guidance and complies with the pertinent regulatory requirements. The staff's primary focus was to evaluate the emergency plan against NUREG-0654, Planning Standard H; NUREG-0696; NUREG-0737; and NSIR/DPR-ISG-01, which provide the detailed evaluation criteria that the staff should consider to determine whether the emergency plan meets the applicable regulatory requirements in 10 CFR 50.47(b)(8) and Appendix E to 10 CFR Part 50.

Technical Support Center

WLS Emergency Plan, Section II.H, "On-site Emergency Response Facilities," provides a discussion about the TSC and the OSC, and it states the facilities were designed to meet the intent of the guidance in NUREG-0696 and the clarification in NUREG-0737, Supplement 1. In RAI 25, Question 13.03-61(I), the staff requested that the applicant provide additional information regarding how the facilities will meet the guidance in NUREG-0696 and the clarification in NUREG-0737, Supplement 1. In a December 23, 2008, response, the applicant stated that a design description addressing the criteria provided in NUREG-0696, Sections 2.1 through 2.10 is included as WLS Emergency Plan, Appendix 10, "Technical Support Center Description," of the WLS Emergency Plan. The applicant also stated that the design satisfies the criteria established in the AP1000 DCD with the exception of being within a 2-minute walk of the CR.

The TSC is common to WLS Units 1 and 2 and is not located in the nuclear island CSA as described in the AP1000 DCD, rather, it is located in the Maintenance Support Building to provide centralized response management oversight for the site. Consequently, the applicant proposed departure WLS DEP 18.8-1 in WLS COLA Section 10, "Departures and Exemption Requests," to address the location for the TSC. In WLS COLA Part 7, the applicant evaluated WLS DEP 18.8-1 pursuant to 10 CFR Part 52, Appendix D, Section VIII.B.5.b, and found that the departure has no safety significance. Specifically, the departure is for a non-safety-related system, the location of the TSC meets applicable requirements, and relocating the TSC does not adversely affect its function. The staff agrees with the applicant's evaluation, for the reasons described below.

WLS Emergency Plan, Section II.H.1, "On-site Emergency Response Facilities," in part, describes the mission of the TSC and the purpose of the TSC. During an emergency, the TSC will provide a workspace and resources for the site personnel to support the CR by relieving the operators from tasks that are not directly related to reactor plant control manipulations. The TSC will provide communication, radiological, and engineering support. Command and control of the emergency response will transfer from the CR to the TSC. To provide technical support the TSC is equipped with the safety parameter display system (SPDS) that provides plant parameters. The TSC is provided radiological protection similar to the CR. WLS Emergency Plan, Section II.H.1, states in the event that all offsite alternating current (ac) power is unavailable, the TSC could be evacuated and function transferred to an unaffected location. A description of the procedure and locations to be considered was not provided. Therefore, in RAI 25, Question 13.03-61(A), the staff requested that the applicant provide information related to this procedure. In a December 23, 2008, response, the applicant stated that procedures for relocating the WLS TSC will be similar to those used at other Duke Energy nuclear plants. However, the applicant stated that alternate locations for the TSC and OSC have not been determined. The applicant proposed EP-ITAAC 5.1 to validate the location for the TSC and EP-ITAAC 5.1.3 Acceptance Criterion to verify back-up power was available to the TSC. The applicant also provided Catawba Nuclear Station procedures for activation of the TSC and OSC as Attachments 1 and 2 in the response to RAI 25, Question 13.03-55 as examples for future WLS procedures. WLS Emergency Plan, Appendix 10, “Technical Support Center Description” provides additional information on the TSC.

WLS Emergency Plan, Appendix 10 also states the TSC provides working space for the personnel assigned to the TSC at the maximum level of occupancy. The working space is sized for a minimum of 25 persons. Minimum size of working space is approximately 75 square feet per person. The applicant proposed EP-ITAAC 5.1 to inspect the as-built TSC and OSC. The applicant also proposed EP-ITAAC Acceptance Criterion 5.1.1 to confirm that the TSC has been located in the Maintenance Building. Also, additional ITAAC for the as-built TSC and OSC are addressed in AP1000 Design Control Document Tier 1, Revision 19, Table 3.1-1.

Due to the proposed TSC location, WLS Emergency Plan, Appendix 10 states that the TSC may not be within a 2-minute walk of either unit’s CR as identified in NUREG-0696. The applicant also states that the capability to retrieve and display plant data and use the communication systems, listed below, reduce the need for face-to-face meetings between the TSC and CR personnel. The communication systems consist of the following:

a. selective signaling telephone system

b. orivate telephone capability to the county and State warning points/EOCs

c. satellite telephone capability available in the TSC and EOF and via portable units

d. dedicated radio networks to the county and State warning points and EOCs

e. separate telephone lines are dedicated for communication with the NRC

f. separate radio system that provides for communication capabilities between the CR, TSC and EOF to the radiological monitoring teams in the field

The communication facilities include the means for reliable primary and backup communication. In addition, AP1000 DCD Tier 1, Revision 19, ITAAC Table 3.1-1, Item 2 will confirm that communication equipment is installed, and voice transmission and reception are accomplished for the TSC.

WLS Emergency Plan, Section II.H.1 states that plant data that is available in the TSC via the SPDS is described in AP1000 DCD Section 18.8.2, "Safety Parameter Display System (SPDS)." AP1000 DCD Tier 2, Chapter 1.9, "Compliance with Regulatory Criteria," Section 1.9, "Three Mile Island Issues," Section (2)(iv), "Safety Parameter Display System," states the purpose of the plant safety parameter display console (or SPDS) is to display important plant variables in the CR in order to assist in rapidly and reliably determining the safety status of the plant. In addition, displays are available at the operator workstations, the remote shutdown workstation, and at the TSC.

In RAI 25, Question 13.03-61(I), the staff requested that the applicant provide additional information regarding how the facilities meet the intent of the guidance in NUREG-0696 and the clarification in NUREG-0737, Supplement 1. In a December 23, 2008, response, the applicant stated that a design description addressing the criteria provided in NUREG-0696, Sections 2.1 through 2.10 is included in WLS Emergency Plan, Appendix 10. The applicant also stated that the design satisfies the criteria established in the AP1000 DCD with the exception of being within a 2-minute walk of the CR. AP1000 DCD Tier 1, Revision 19, EP-ITAAC Table 3.1-1, Item 3 will confirm that the plant parameters listed in AP1000 DCD Tier 1, Revision 19, Table 2.5.4-1, minimum inventory table, with a "Yes" in the "Display" column, can be retrieved in the TSC.

In RAI 25, Question 13.03-61(I)(b), the staff requested that the applicant address management plans, facility staffing, and task assignments of TSC personnel. In a December 23, 2008, response, the applicant stated that management, staffing, and assignments of TSC personnel are addressed in EPIPs. These procedures will be similar to Catawba Nuclear Stations TSC activation procedure, which was included as Attachment 1 to the response to RAI 25, Question 13.03-55. This Attachment 1 was reviewed by the NRC staff and found to be an acceptable example. Actual procedures for WLS will be developed and submitted in accordance with License Condition (13-8). This License Condition is discussed in Section 13.3.4.19 of this report.

In RAI 25, Question 13.03-61(I)(c), the staff requested that the applicant provide a detail staffing plan for the TSC to address the overall management of licensee resources and the continuous evaluation and coordination of licensee activities during and after an accident. In a December 23, 2008,response, the applicant stated that WLS Emergency Plan, Section II.A.4, "Continuous Operation," outlines the capability for continuous operations through training of multiple responders for key emergency response positions allowing for multiple shifts for extended response operations. Additional information on staffing of the TSC is provided in the response to RAI 25, Question 13.03-55. The staff reviewed this information and finds it acceptable.

In RAI 25, Question 13.03-61(I)(d), the staff requested that the applicant provide the TSC staff assignments to address that TSC management of licensee onsite and offsite radiological monitoring, to perform radiological evaluations, and to interface with offsite officials. The staff also requested that the applicant address whether the personnel assigned to the TSC varies according to the emergency class. In a December 2008, response, the applicant stated that TSC staff assignments will be similar to that in use at other Duke Energy nuclear stations. In accordance with procedures, Radiation Protection personnel are responsible for activating and dispatching field monitoring teams. TSC offsite agency communicators ensure that communicators in the EOF are aware of information affecting offsite agencies. Staffing levels are not varied based on the emergency classification. Procedures will contain provisions for emergency response managers to request additional support from other organizations to assess and mitigate the emergency condition. The Catawba Nuclear Station procedure for activation of the TSC was included as Attachment 1 in the response to RAI 25, Question 13.03-55 as an example. This Attachment 1 was reviewed by the staff and found to be an acceptable example. Actual procedures for WLS will be developed and submitted in accordance with License Condition (13-8). This License Condition is discussed in Section 13.3.4.19 of this report.

In RAI 25, Question 13.03-61(I)(e), the staff requested that the applicant address procedures for and training of personnel to use the data systems and instrumentation and include limitations of instrumentation. In a December 23. 2008, response, the applicant stated that information regarding the Emergency Response Training program is discussed in the response to RAI 25, Question 13.03-61(I)(a) and the training program description within the WLS Emergency Plan, Section II.O.4. The training program requires TSC staff to receive an overview of the site emergency plan and training on facility operations, technical assessment function, and task-specifics consistent with assigned duties. This task-specific training includes, for example, use of data systems and instrumentation, including the limitation of instrumentation for assigned personnel.

In RAI 25, Question 13.03-61(I)(f), the staff requested that the applicant address how the TSC staff maintain proficiency (participation in drills). In a December 23, 2008, response, the applicant stated that the exercise and drill program is discussed in WLS Emergency Plan, Sections II.N.1.a, “Exercise Scope and Frequency”; II.N.1.b, “Exercise Scenarios and Participation”; and II.N.2, “Drills.” The applicant also provided additional information related to the goals and primary objectives of drills and exercises. The applicant further stated that TSC staffs participate in these exercises and drills to maintain their proficiency.

In RAI 25, Question 13.03-61(I)(g), the staff requested that the applicant address whether there are means for facsimile transmission capability between the EOF, TSC, and NRC Operations Center. In a December 23, 2008, response, the applicant stated that facsimile transmission between the EOF, TSC, and NRC Operations Center will be supported at the TSC. New advancements in technology will be considered before incorporating transmission systems into the facilities. An EP-ITAAC regarding this capability was proposed in AP1000 DCD Tier 1, Table 3.1-1 and WLS COLA Part 10, Table 3.8.1.

WLS Emergency Plan, Appendix 10 states that the TSC is designed in accordance with the Uniform Building Code (UBC) to withstand earthquakes and high winds. Support facilities are located within the TSC to support long term operation of the TSC. WLS Emergency Plan, Appendix 10 further states that the TSC is provided with reliable power and backup power supplies. Lighting is powered by the normal and backup electrical supply system. An emergency battery operated lighting system is installed. Power for vital information systems is provided by reliable power supplies including a battery backed uninterruptible power supply (UPS) system.

WLS Emergency Plan, Appendix 10 states that the TSC is environmentally controlled to provide room air temperature, humidity, and cleanliness appropriate for personnel and equipment. The ventilation system includes high efficiency particulate air (HEPA) filters and charcoal filters. The ventilation system is designed to maintain exposures at or below 0.05 Sievert (Sv) (5 roentgen equivalent man (rem)) TEDE as defined in 10 CFR 50.2, "Definitions," for the duration of an accident. The TSC structure, shielding, and ventilation system are also designed to protect the TSC personnel from radiological hazards. The ventilation system is operated in accordance with approved procedures and is manually controlled from the TSC. In RAI 25, Question 13.03-61(J), the staff requested that the applicant provide details pertaining to ventilation design such as air inlet flow rates, recirculation flow rates, unfiltered air inleakage, and other factors necessary to complete a radiological assessment. In December 2008 and March 16, 2015, responses, the applicant provided a TSC Design Description Document and a detailed radiological assessment. The TSC Design Description Document stated that the TSC heating, ventilation, and air-conditioning (HVAC) system functions to provide normal environmental control for personnel and equipment operational requirements; and provides environmental control for habitability through filtration of potentially radioactive particulates and adsorption of iodine during emergency conditions. The applicant also stated that the TSC is designed to comparable levels of habitability, such as humidity and temperature, as described in the AP1000 DCD, as well as the same radiological habitability as the CR, under accident conditions. A design description for the location of the TSC is provided in WLS Emergency Plan, Appendix 10 of the application, which addresses criteria in NUREG-0737, Supplement 1, Section 8.2.1.

Furthermore, in TSC Design Description Document the applicant stated that radiation monitoring systems are available to personnel in the TSC. These monitoring systems may be composed of installed monitors or portable monitoring equipment. These systems continuously indicate radiation dose rates and airborne radioactivity concentrations inside the TSC while it is in use during an emergency. These monitoring systems include local alarms with trip levels set to provide early warning to TSC personnel of adverse conditions that may affect the habitability of the TSC. These detectors are able to distinguish the presence or absence of radioiodines at concentrations as low as 10^{-7} microcuries per cc. In addition, Appendix 10 states that portable radiation monitors are available to personnel in the TSC. EP-ITAAC Acceptance Criterion 5.1.2 has been proposed to confirm that the TSC includes radiation monitors and a ventilation system with a HEPA and charcoal filter.

Equipment and supplies are provided in accordance with WLS Emergency Plan, Appendix 6, "Emergency Equipment and Supplies." WLS Emergency Plan, Appendix 6 provides a general list of equipment located in the ERFs, including the TSC.

WLS Emergency Plan, Appendix 10 describes the technical and operational data and information that is available for each Lee Nuclear Station unit within the TSC. The TSC is equipped with a computer system, which provides source term and meteorological data and technical data displays to allow TSC personnel to perform detailed analysis and diagnosis of abnormal plant conditions, including assessment of any significant release of radioactivity to the environment. EP-ITAAC 6.4 has been proposed to verify the capability to perform an inspection of the TSC to verify the availability of the meteorological data is available in the TSC. Also, HFE is incorporated into the design of the TSC related to the display and availability of plant data.

WLS Emergency Plan, Appendix 10 states that the TSC has ready access to plant records and provides a list of specific documents, procedures, reports, and drawings that will be maintained in the TSC.

WLS Emergency Plan, Section II-H.4, “Activation and Staffing of Emergency Response Facilities,” states that all WLS ERFs will activate at an Alert or higher classification. Following notification, emergency response personnel report to their assigned ERF and undertake activities necessary to make the ERF fully functional. Additionally, should security or other conditions make activating the onsite ERFs impracticable or hazardous, the Emergency Coordinator will implement alternate plans as described in the EPIPs, which may direct the ERO to alternate offsite ERFs to assemble and await instructions until favorable conditions exist to activate the onsite ERFs.

The staff finds that the common TSC provides an area that meets the applicable regulatory guidance in NUREG-0696 and NUREG-0737, Supplement 1 and, as such, will adequately support its intended emergency response functions. Therefore, the staff concludes that WLS DEP 18.8-1 is acceptable with respect to the TSC location.

Operations Support Center

WLS Emergency Plan, Section II.H.1, “On-site Emergency Response Facilities,” describes the primary function of the OSC staff is to dispatch assessment, corrective action, and rescue personnel to locations in the plant as directed by the TSC and CR. Each unit has a separate OSC, and provides a centralized area and the necessary supporting resources during emergency conditions. Designated plant support personnel, as indicated in WLS Emergency Plan, Section II.B, “Onsite Emergency Organization,” assemble in the designated OSC to provide support to both the CR and TSC. WLS Emergency Plan, Section II.B.6, “Plant Emergency Response Staff,” states that WLS Emergency Plan, Figure II-2, “WSL Emergency Response Organization-TSC/OSC Only,” illustrates the plant staff emergency organization. The figure identifies an OSC Director as directing the OSC staff and reporting to the Site Emergency Director. WLS Emergency Plan, Section II.B.5, “Plant Emergency Response Staff,” describes the OSC Director as the one who directs repair teams, performs damage assessment, and coordinates OSC teams to provide proper briefings and accompaniment by radiation protection personnel as applicable.

WLS DEP 18.8-1 states that the OSC location will be described in the WLS Emergency Plan. WLS Emergency Plan, Section II.H states that the OSCs are located in the space formerly designated in the AP1000 DCD for the TSC. In WLS COLA Part 7, the applicant evaluated WLS DEP 18.8-1 pursuant to 10 CFR Part 52, Appendix D, Section VIII.B.5.b and found that the departure is for a non-safety-related system, that the location of the OSC meets applicable requirements, and that relocating the OSC does not adversely affect its function. The staff agrees with the applicant’s evaluation for the reasons described below.

WLS Emergency Plan, Section II.H.1 statement: “Implementing procedures make provisions for the relocation of the OSC as needed…” Additional information on the operation of the OSC and TSC can be found in AP1000 DCD Section 18.8.3.5, “Technical Support Center Mission and Major Tasks.” AP1000 DCD Tier 1, Section 3.1, “Emergency Response Facilities,” contains a description of the facility and its EP-ITAAC acceptance criteria. In RAI 25, Question 13.03-61(H), the staff requested that the applicant provide additional information related to the design of the OSC. In a December 23. 2008, response to RAI 25, Question 13.03-61(H), the applicant stated that site layout drawings are not included in emergency plans or implementing procedures. This information will be included in training and orientation of OSC personnel. The applicant also stated AP1000 DCD Figures 1.2-17 through 1.2-20 are designated as security-related information and properly withheld from public disclosure pursuant to NRC regulations and guidance. WLS COLA Figure 1.2-201, (which replaces AP1000 DCD Figure 1.2-18) is similarly withheld and included in WLS COLA Part 9, “Withheld.” The applicant also stated that this information is available for review through processes and procedures established by the NRC for such material. The applicant proposed EP-ITAAC Acceptance Criterion 5.1 to test that the applicant has established a TSC and OSC. The introductory information in WLS Emergency Plan, Section H, “Emergency Facilities and Equipment,” states that the OSCs are designed to meet the intent of Supplement 1 to NUREG-0737. The applicant proposed EP-ITAAC Acceptance Criterion 5.1.4 to confirm that the OSC is in a location separate from the CR.

WLS Emergency Plan, Section II.H.1 describes OSC resources for communicating with the CR and the TSC. Also, WLS Emergency Plan, Section II.F, “Emergency Communications,” describes a radio system as the back-up communication capability. The communication capabilities permit personnel reporting to the OSC to be assigned to duties in support of emergency operations. AP1000 DCD Tier 1, Revision 19, EP-ITAAC, Table 3.1-1, Item 4 ensures that communication equipment is installed, and voice transmission and reception are accomplished for the OSC. Communication systems are covered in WLS Emergency Plan, Sections II.E, “Notification Methods and Procedures,” and II.F, “Emergency Communications.” In RAI 25, Question 13.03-61(F), the staff requested that the applicant provide additional information regarding communication systems available in the OSC. In a December 23, 2008,response, the applicant stated that WLS Emergency Plan, Section II.H.1 describes functionality and habitability of the ERFs in compliance with NUREG-0696, Criteria 3.1 and 3.2. WLS Emergency Plan, Section II.F discusses the use of a wireless telephone system for communication between the facilities. The telephone-page and Private Branch Exchange (PBX) telephone communication systems serve as backups to this system.

The OSC is not designed to remain habitable under all projected emergency conditions. However, implementing procedures make provisions for relocating the OSC as needed, based on ongoing assessments of plant conditions and facility habitability. The Site Emergency Director directs relocation of the OSC if required.

WLS Emergency Plan, Section II.H.1 states that protective clothing and respirators are discussed in WLS Emergency Plan, Section II.J.6, “Protective Measures.” WLS Emergency Plan, Appendix 6, “Emergency Equipment and Supplies,” provides a list of supplies and equipment for all ERFs and states that detailed equipment and supplies will be addressed in implementing procedures.

The staff finds that the relocation of the units’ OSCs to the space formerly designated in the AP1000 DCD for the TSCs is acceptable because the OSCs provide an area that meet the applicable guidance in NUREG-0696 and NUREG-0737; Supplement 1 and, as such, will allow the OSC to adequately support its intended emergency response functions. From a support and functional standpoint, the staff finds the applicant’s proposed OSCs locations acceptable, subject to a demonstration of adequacy during the full participation exercise (addressed in ITAAC 8.1). Therefore, the staff concludes that WLS DEP 18.8-1 is acceptable.

Emergency Operations Facility

WLS Emergency Plan, Section II.H.2, "Off-site Emergency Response Facilities," states that the purpose of the EOF and associated EOF staff is to provide the facilities and staffing for evaluating, coordinating, and directing the overall activities involved in coping with a radiological emergency. During an emergency, the EOF Director and his staff review the response to the emergency by the Duke Energy and the appropriate State and local agencies to facilitate execution of an effective and cooperative effort. The EOF Director is responsible for providing the Duke Energy's recommended protective actions to the appropriate State and local officials. The EOF staff coordinates with other Duke Energy emergency centers to facilitate an effective Duke Energy effort in response to an emergency situation. The EOF staff also provides an accurate description of the emergency situation for Duke Energy management and public information. In addition, the EOF coordinates with offsite Federal agencies, such as the NRC and DOE, to provide availability of additional outside resources to Duke Energy.

Duke Energy has filed for an exception to have the EOF located in the Charlotte General Office in the Energy Center at 526 South Church Street, Charlotte, North Carolina. Justification of this exception can be found in WLS Emergency Plan, Appendix 9, "Justification for Common EOF." WLS Emergency Plan, Appendix 9 states that because the EOF is more than 16 km (10 mi) from any of the Duke Energy nuclear stations, no radiological monitoring equipment is required. The applicant has proposed that the Charlotte, North Carolina, EOF currently used for Duke Energy's existing nuclear stations at McGuire Nuclear Station (MNS), Catawba Nuclear Station (CNS), and Oconee Nuclear Station (ONS) be used for the WLS as well. The applicant stated that the centralized EOF has proven to be an effective facility for implementing the nuclear station emergency plans. MNS and CNS have used a common EOF since August 1987. The existing facility has been in use since October 2005 and was used in the Catawba 2006 biennial exercise. In 2006, the applicant received NRC approval to use the EOF for ONS. Communication systems, data links, and staffing have been incorporated and tested. Using the centralized EOF for WLS would allow the applicant to apply its corporate emergency response structure and experience to the WLS Emergency Plan. The applicant has discussed this proposal with South Carolina Emergency Management, North Carolina Emergency Management, South Carolina Department of Health and Environmental Control, North Carolina Department of Environmental Health and Natural Resources, Cherokee County, South Carolina Emergency Management, York County, South Carolina Emergency Management, and Cleveland County, North Carolina Emergency Management. North and South Carolina are familiar with the EOF because it is the current facility used for responding to an event at MNS, CNS, and ONS. Acknowledgement of their support for use of the EOF location is included in their respective letters certifying their agreement with the WLS Emergency Plan. The NRC will have access to plant data through the Duke Energy satellite display system (SDS) and ERDS. The NRC also has telephones on the ETS in Charlotte, North Carolina. Equipment exists in the EOF for the acquisition, display, and evaluation of radiological, meteorological, and plant system data used to determine offsite protective measures. Release information is provided by the field monitoring teams and is used to determine appropriate PARs. Plant and effluent data would be provided on as timely a basis at an EOF in Charlotte as it would be at a near-site location. Various plant parameters are available to the EOF staff via a connection through Duke's Wide Area Network (WAN). Data available at the EOF provides a snapshot of data from each unit's integrated set of plant data as described in AP1000 DCD Chapter 18, "Human Factors Engineering." Plant data can be displayed at the EOF. These data sets are sufficient to perform accident assessment and evaluate the potential onsite and offsite environmental consequences of an emergency at Lee Nuclear Station. The computers in the Dose Assessment Area are capable of running the dose projection computer programs (Raddose-V) and accessing SDS data. A Duke Energy staff meteorologist in the EOF provides meteorological information to the EOF staff, in support of offsite dose projections. In addition, the applicant proposed EP-ITAAC 6.4 to perform an inspection of the EOF to verify the availability of the meteorological data.

WLS Emergency Plan, Appendix 9 describes the building construction, radiological protection, installed equipment to support the response, security measures, up-to-date plant reference information, and communication systems. The EOF is constructed so that it is capable of withstanding wind loads and live loads equal to or greater than those specified in the current North Carolina State Building Code (2000 International Building Code). The current Building Code specifies a basic design wind speed of 145 kilometers per hour (90 miles per hour) (mph))-3 second gust and a total minimum floor live load of 342 kilograms per square meter (70 pounds per square foot (psf)). This evaluation is based on a review of original structural drawings and comparison to requirements of the current North Carolina State Building Code. The EOF includes space for South and North Carolina liaisons reporting to the EOF, as well as workspace for the NRC that is co-located with decision-makers, radiological assessment, and accident assessment personnel. The EOF draws its primary power from commercial sources. A loss of commercial power should not impact any of the voice or data communication equipment located in the EOF. Common Duke Energy telecommunication infrastructure that supports EOF functions, including, but not limited to, fiber optic transmission equipment, telephone switching equipment and data network routers, is configured to operate from at least one and usually multiple backup power sources in the event of a loss of commercial power. These backup sources include generator, dc battery, and UPS systems. EP-ITAAC Acceptance Criterion 5.2.1 will confirm that the EOF has at least 243 square meters (2625 square feet).

Since the EOF is located outside of the 16-km (10-mi) EPZ, it is not required to have any additional radiation protection factor.

The communication systems at the EOF are designed to provide for the reliable, timely flow of information between all parties having an emergency response role. The single facility results in commonality of communication and interface with offsite officials and liaisons. The Selective Signaling System continues to be the primary means of communicating changes in event classification and PARs to the States and counties. The Selective Signaling System, as well as the Decision Line, operate on a combination of the Duke Energy Telecommunications network and leased circuits. The offsite communication network is used to communicate with Federal, State, and other supporting agencies. Access to these agencies is provided through several redundant, diverse routes. This diversity provides offsite routing through more than one type of facility. These facilities include, but are not limited to, commercial facilities such as central office trunks, tie-lines and digital services, plus privately owned and maintained microwave and fiber-optic systems. The offsite telecommunication networks are designed to facilitate traffic in the most fail-safe manner to the EROs. ENS, HPN and commercial telephones provide communication from each site TSC, CR, and the EOF to the NRC Headquarters and regional offices. These telephones are tested on a periodic basis consistent with the WLS Emergency Plan. A control station with a remote connection to the EOF allows the EOF to communicate with the WLS Field Monitoring Teams. Additional radio capability for communication with counties within the WLS plume exposure pathway EPZ will include South Carolina Local Government Radio for Cherokee and York Counties, and North Carolina Satellite Radio for

Cleveland County. Existing commercial telephone service will serve as the designated backup means of communication in the event of a Selective Signaling System or Decision Line failure. Duke Energy has telecommunication capabilities that can provide access to long distance networks without having to go through a local telephone company switch. Long distance calls from the EOF are routed through Duke Energy's corporate PBX in Charlotte, NC, directly to either a primary or backup long distance carrier. Telephones are provided for the respective Federal and State representatives, including lines for faxes and modems. Facsimile machines are available in the EOF to support the transmission of information between the ERFs and with State, local, and Federal authorities.

The applicant proposed ITAAC Acceptance Criteria 5.2.2 and 5.2.3, which will verify the EOF's capability to communicate with the TSC by voice and the use of the Selective Signaling Telephone system with the local and State Warning Points.

Hard copies of key reference materials are maintained in the Nuclear General Office facilities in Charlotte, and are brought to the EOF upon activation. In addition, station design documentation, plant drawings, WLS COL FSAR, procedures, etc., are available via local area network (LAN) connection from the Nuclear Electronic Document Library. The following information is available for WLS at the EOF: plant technical specifications, plant operating procedures, emergency operating procedures, WLS COL FSAR, up-to-date license, State and local emergency response plans, offsite population distribution data, and evacuation plans.

The EOF is provided with normal industrial security, and processes are already established to upgrade security during activation.

Section 18.2.1 of this report discusses the implementation and verification of applicable EOF displays in accordance with the AP1000 HFE program.

WLS Emergency Plan, Section II.E.2, "Notification and Mobilization of Licensee Response Organizations," describes staffing of the EOF. The EOF can be activated within about 75 minutes of the declaration of an "alert" or higher level emergency. The EOF staff has demonstrated their ability to staff the EOF within 75 minutes of emergency declaration during annual augmentation drills for McGuire and Catawba Nuclear Stations. The EOF staff will include personnel to manage overall licensee emergency response, coordinate radiological and environmental assessment, determine recommended public protective actions, and interface with offsite officials. WLS Emergency Plan, Section II.F, "Emergency Communications," provides a description of the communication capabilities provided in the EOF. The EOF is staffed and activated in accordance with EPIPs. WLS COL FSAR Table 13.4-201, "Operational Programs Required by NRC Regulations," states detailed implementing procedures will be submitted at least 180 days prior to fuel loading. The EOF is declared activated following an assessment of staffing levels, habitability, operability of installed systems, sufficiency of supplies and equipment, and communication interfaces. Alternate plans can be initiated in a time of adverse conditions. In RAI 25, Question 13.03-73(B), the staff requested that the applicant provide clarification between the concepts of "activation" and "staffing." In a December 23, 2008, response, the applicant stated that within the WLS Emergency Plan, the concept of "activation" as used in NUREG-1793 and the AP1000 DCD includes the activities of notifying the appropriate emergency response personnel, staffing the ERFs, establishing the required communication interfaces, and declaring the facility to be operational.

The staff focused its review on the extension of the existing centralized Duke Energy EOF to the WLS, and concluded that the EOF is applicable to the proposed reactor, the information provided in the application was adequate to support a combined use facility, and that WLS will be addressed in the EOF procedures as part of the implementation milestones and requirements related to emergency planning in WLS COL FSAR Table 13.4-301, “Operational Programs Required by NRC Regulations,” Item 14, In accordance with the Staff Requirements Memorandum (SRM) to SECY 10-0078, “Centralized Emergency Operations Facilities and Combined License Applications,” the Commission granted the authority for the staff to review any applicant proposals for centralized EOFs and to determine their respective acceptability as part of the 10 CFR Part 52 COL application process. Therefore, the staff finds that the WLS Emergency Plan adequately describes a combined EOF from which evaluation and coordination of all licensee activities related to an emergency is to be carried out. In addition, the EOF provides information to Federal, State and local authorities responding to radiological emergencies. The considers this acceptable because it meets the applicable guidance in NUREG-0737, Supplement 1 and the applicable regulatory requirements of 10 CFR 50.34(f)(2)(xxv).

In WLS COLA Part 10, the applicant proposed License Condition 4 “Emergency Planning Actions,” to demonstrate the integrated capability and functionality of the EOF. The staff refers to this as License Condition (13-7).

License Condition (13-7):

> Prior to fuel load, Duke Energy will demonstrate the integrated capability and functionality of the EOF for activation and operation of the facility to respond to emergency events at both the WLS and one additional nuclear facility that is supported by the EOF. Integrated communication and data capability and functionality will include the Technical Support Centers for WLS and one additional nuclear facility, and other Federal, State, and local coordination centers as appropriate.

The staff evaluated the proposed license condition and finds License Condition (13-7) acceptable because it is consistent with the Final Rule and NSIR/DPR-ISG-01.

Other Emergency Facilities and Equipment

WLS Emergency Plan, Section II.H.1 designates and describes two alternate facilities that would be used for an assembly and staging location if the onsite emergency facilities were not available, including during a hostile action event. The WLS Training building, which will be located within the owner-controlled area outside of the protected area, and the Kings Mountain Generation Support Facility, which is located approximately 25 kilometers (km) (15.5 mi) from the site support a rapid site response. Both locations have communication links with the EOF, control room and security; the capability to provide timely notification to offsite response organizations for changes in classification levels or protective action recommendations; the capability for engineering assessment, damage control planning and preparations; and computer links to the site’s plant data.

WLS Emergency Plan, Section II.H.5, “Onsite Monitoring Systems,” contains a description of the various monitoring systems necessary for initiating emergency measures and performing accident assessment. This includes monitoring systems for geophysical phenomena, radiological conditions, plant procedures, and fire hazards as follows:

- Personnel monitoring equipment is described in the AP1000 DCD, Revision 19, and the WLS COL FSAR corresponding section.

- Geophysical phenomena are described in AP1000 DCD, Revision 18, Section 3.7.4, “Supporting Media for Seismic Category I Structures” and the WLS COL FSAR corresponding section.

- Radiological monitoring systems can be found in AP1000 DCD, Revision 19, Sections 11.5, “Radiation Monitoring,” and 12.3, “Radiation Protection Design Features,” and the WLS COL FSAR corresponding sections. A supply of portable radiation monitoring and sampling equipment and emergency response equipment is addressed in WLS Emergency Plan, Section II.H, “Emergency Facilities and Equipment,” and WLS Emergency Plan, Appendix 6, “Emergency Equipment and Supplies.”

- Plant process monitoring systems are described in AP1000 DCD, Revision 19, Section 11.5 and the WLS COL FSAR corresponding section.

- Plant fire monitoring systems are described AP1000 DCD, Revision 19, Section 9.5.1, “Fire Protection Systems” and the WLS COL FSAR corresponding section.

- An emergency plan implementing procedure will describe the bases for the selection of the designated instruments as indicators of emergency conditions.

WLS Emergency Plan, Section II.H.7, “Off-site Radiological Monitoring Equipment,” states that WLS provides offsite radiological monitoring equipment suitable for assessment of offsite radiological consequences of facility incidents. WLS Emergency Plan, Appendix 6, “Emergency Equipment and Supplies,” lists the general types of equipment that would be available for offsite measurements. This equipment includes: radiation survey instruments; surface contamination control and survey supplies; air sampling equipment and media; and scalers or other appropriate radio analytical counting instruments. Further, WLS Emergency Plan, Section II.I.9, “Measuring Radioiodine Concentrations,” states that the field equipment is capable of detecting radioiodine concentrations of 10E-7 microcuries per milliliter under field conditions. Furthermore, WLS has an Offsite Dose Calculation Manual (ODCM) that describes the monitoring systems. The plant also has equipment and radiological laboratory facilities available on site. Environmental monitoring equipment includes multiple radioiodine and particulate monitors and thermo luminescent dosimeters (TLDs). The TLDs are posted and collected in accordance with Branch Technical Position (BTP), Revision 1, Table 1 included with GL 79-65, “Environmental Monitoring for Direct Radiation.” Locations of TLDs and air sampler postings are listed in the ODCM. In RAI 25, Question 13.03-61(E), the staff requested that the applicant provide additional information on monitoring systems and the locations of dosimeters and air samplers that is available in the ODCM. In a December 23, 2008, response, the applicant stated that the ODCM is discussed in Environmental Report (ER), Section 6.2, “Radiological Monitoring.” WLS COL FSAR Section 11.5.7, “Combined License Information,” states that a description of the ODCM program will be finalized prior to fuel load. Milestones for implementation of the ODCM program are provided in WLS COL FSAR Table 13.4-201. Station monitoring and sampling locations are identified in WLS COL FSAR Table 6.2-2, and WLS COL

FSAR Figures 6.2-1, “Near Field Radiological Sampling and Monitoring Locations,” and 6.2-2, “Far Field Radiological Sampling and Monitoring Locations.” The program is based on guidance in BTP Revision 1 included with GL 79-65. The applicant identified a license condition for implementing ODCM and Radiological Environmental Monitoring Program, which is addressed in WLS COLA Part 10, “Proposed Licensed Conditions (including EP-ITAAC).” The staff notes this is consistent with 10 CFR Part 50, Appendix E, Section V and the allowances provided in SECY-05-0197, “Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria.” Arrangements for backup support and analysis are described in WLS Emergency Plan, Section II.A, “Assignment of Responsibility (Organizational Control),” and arrangements with other organizations are documented with certification letters in WLS Emergency Plan, Appendix 7, “Certification Letters.” Descriptions of laboratory facilities both fixed and mobile are in WLS Emergency Plan, Section II.C.3, “Radiological Laboratories.”

WLS Emergency Plan, Section II.H.8, “Meteorological Instrumentation and Procedures,” discusses the onsite meteorological data collection system. The meteorological data is acquired from an onsite meteorological tower. The tower measures wind speeds, ambient temperatures, atmospheric stability, dew point, and precipitation. The meteorological monitoring program and climatology are described in WLS COL FSAR Section 2.3, “Meteorology.” All measured data from onsite meteorological tower is available to the plant and ERF display systems. Meteorological data can also be obtained from the CNS and the NWS in Greer, South Carolina. In RAI 25, Question 13.03-61(E)(1), the staff requested that the applicant provide additional information regarding the procedures related to meteorological data. In a December 23, 2008, response, the applicant stated that alternate meteorological data sources are located within 80 km (50 mi) of the WLS site and have been found to be representative of the WLS location. The applicant’s meteorologist is responsible to interpret data received and to determine representativeness of the data when onsite meteorological systems cannot be used. The applicant also provided Duke Energy's corporate procedure for obtaining data from an alternate source as Attachment 1 to the response to RAI 25, Question 13.03-62. Furthermore, flooding data is available from NOAA Hydro Meteorological Reports, and the backup seismic data is available from the U.S. Geological Survey (USGS). All of the data is shared with local, State, and Federal organizations. (See WLS Emergency Plan, Section II.F, “Emergency Communications,” for a description.)

WLS Emergency Plan, Section II.H.10, “Emergency Equipment and Supplies,” n states that the applicant performs inspections and operational test of emergency equipment once each calendar quarter. Reserves are maintained to replace instruments removed for calibration or repair. The scope and responsibilities for performing these tests are provided in administrative procedures. A description of the equipment is in WLS Emergency Plan, Appendix 6, “Emergency Equipment and Supplies.” In RAI 25, Question 13.03-61(B), the staff requested that the applicant provide additional information on the procedures to inspect and test dedicated emergency equipment. In a December 23, 2008, response, the applicant stated that the procedure for verifying availability and readiness of emergency response equipment will be similar to that in use at other Duke Energy nuclear plants. The applicant provided CNS's Procedure and a Duke Energy corporate procedure as examples of this process. The staff notes that a license condition has been proposed in WLS COLA Part 10 addressing the submittal schedule for operational programs, including EPIPs, which is consistent with 10 CFR Part 50, Appendix E, Section V and the allowances provided in SECY-05-0197.

WLS Emergency Plan, Appendix 6, “Emergency Equipment and Supplies,” states there will be emergency equipment and provided a general list of its contents. In RAI 25, Question 13.03-61(G), the staff requested that the applicant provide additional information on the contents of the emergency kits. In a December 23, 2008, response the applicant stated that information regarding emergency kits will be similar to that in use at other Duke Energy nuclear plants. The applicant provided a CNS's procedure and a Duke Energy corporate procedure as examples of this process. A license condition addressing the submittal schedule for implementation of EPIPs, is addressed in WLS COLA Part 10, “Proposed Licensed Conditions (including EP-ITAAC),” which is consistent with 10 CFR Part 50, Appendix E, Section V and the allowances provided in SECY-05-0197.

WLS Emergency Plan, Section II.H.12, “Receipt of Field Monitoring Data,” states that Radiological Assessment personnel in the EOF are the central point for the receipt of offsite monitoring data results and sample media analysis. The Radiological Assessment personnel will evaluate the information and make recommendations. The equipment in the lab can be used to determine activity of samples. Instruments are routinely calibrated to ensure availability. Field monitoring equipment is maintained at Lee Nuclear Station.

Onsite first aid capability is discussed in WLS Emergency Plan, Section II.L.2, “On-site First Aid Capability,” and a generic list of supplies can be found in WLS Emergency Plan, Appendix 6, “Emergency Equipment and Supplies.”

Procedures to review, audit, and update the emergency plan are covered in WLS Emergency Plan, Section II.P.4, “Plan Reviews and Updates.” The WLS Emergency Plan is to be reviewed and updated on an annual basis. Implementing procedures are discussed in WLS Emergency Plan, Section II.P.7, “Implementing Procedures,” and WLS Emergency Plan, Appendix 5, “Implementing Procedures.” WLS Emergency Plan, Section II.H.10, “Emergency Equipment and Supplies,” states that the applicant performs inspections and operational tests of emergency equipment once each calendar quarter.

Emergency Response Data System (ERDS)

WLS Emergency Plan, Section II.F.1.g, “Description of Communications Links,” states that the ERDS is activated within 1 hour after declaring an “Alert,” or higher emergency classification. WLS Emergency Plan, Section II.N.2.a states that testing of the communication systems between the WLS and the NRC Headquarters and the Regional Operations Center will be performed monthly. Additional information is provided in WLS COL FSAR Section 9.5.2.2.3.2.1, “NRC Communication Interfaces.” AP1000 DCD Tier 2, Section 7.7, “Control and Instrumentation Systems,” discusses system parameters. Meteorological data parameters are discussed in WLS COL FSAR Chapter 2, “Site Characteristics,” and WLS COL FSAR Section 2.3.3, “Onsite Meteorological Measurement Programs.” and WLS Emergency Plan, Section II.H.8, “Meteorological Instrumentation and Procedures.” Radiation monitoring is discussed in the AP1000 DCD Tier 2, Section 11.5, “Radiation Monitoring,” and AP1000 DCD Section 11.1.2, “Plant Monitoring Systems.” Containment parameter monitoring is discussed in AP1000 DCD Chapter 7, “Instrumentation and Controls.” In RAI 25, Question 13.03-61(C), the staff requested that the applicant provide the following information regarding the data points transmitted for selected plant conditions: (1) verify that data points can be transmitted for reactor core and coolant system conditions; reactor containment conditions; radioactivity release rates; and plant meteorological tower data; (2) verify that a separate data feed will be provided for each reactor unit; (3) if the ERDS is to communicate with a safety system, verify that appropriate isolation devices will exist at these interfaces. The staff also requested additional information regarding the ERDS in RAI 25, Questions 13.03-61(D)(1 through 4). In a December 23, 2008,response, the applicant stated that data points for reactor and core coolant system conditions; reactor containment conditions; radioactivity release rates; and plant meteorological tower data will be available for transmittal, and a separate data feed for each reactor unit is to be provided. Data transmission design will include isolation devices as part of the Cyber Security Program being developed. The process and hardware used to transmit data has not been identified but will be specific to AP1000 design features and based on regulatory guidance. The applicant also stated that the ERDS for WLS will be developed on a schedule in compliance with the milestones provided in WLS COLA, Part 10.

In RAI 25, Question 13.03-61(D)(1), the staff requested that the applicant verify that the system is capable of transmitting ERDS parameters in not more than 60 seconds or no less than 15 seconds. In a December 23, 2008, response, the applicant stated that ERDS parameters can be transmitted in no more than 60 seconds or no less than 15 seconds.

In RAI 25, Question 13.03-61(D)(2), the staff requested that the applicant verify that the link control and data transmission is established in a compatible format with NRC receiving equipment. In a December 23, 2008, response, the applicant stated that link control and data transmission is in a compatible format with the NRC receiving equipment.

In RAI 25, Question 13.03-61(D)(3), the staff requested that the applicant verify that any hardware or software changes that affect the transmitted data points identified in the ERDS Data Point Library will be submitted to the NRC within 30 days after the changes are completed. The staff also requested that the applicant verify that hardware and software changes that could affect the transmission format and computer communication protocol to the ERDS will be provided to the NRC at least 30 days prior to the modification. In a December 23, 2008, response, the applicant stated that hardware or software changes that affect the transmitted data points identified in the ERDS Data Point Library will be submitted to the NRC within 30 days after the changes are completed. The applicant also stated that hardware and software changes that could affect the transmission format and computer communication protocol to the ERDS will be provided to the NRC at least 30 days prior to the modification. The applicant proposed EP-ITAAC Acceptance Criterion 3.2.2 that confirms ERDS data was provided from the plant computer system to NRC Headquarters and Region II EOC.

In RAI 25, Question 13.03-61(D)(4), the staff requested that the applicant verify that an ERDS implementation program plan has or will be submitted to the NRC. In a December 23, 2008, response, the applicant stated that an ERDS implementation program plan will be submitted to the NRC. The applicant also stated that some of the details regarding this information are specific to the design features of the AP1000 and will be based on applicable regulatory guidance. Other details are applicable to the emergency planning program implementation. The ERDS and implementation procedures for WLS will be developed on a schedule in compliance with the milestones provided in WLS COLA, Part 10, Tier 1, Table 2.3-1, “ITAAC for Emergency Planning.”

In WLS COLA Part 10, the applicant proposed License Condition 6e to address the implementation of ERDS. This proposed License Condition is evaluated in Section 13.3.4.19 of this report and the staff refers to this as License Condition (13-9).

In RAI 94, Question 13.03-88(A), the staff requested that the applicant additional information to address alternate ERO facilities to be used during security based events. In an April 25, 2011, response, the applicant stated that in the event of hostile actions directed at WLS; the Security Training Area, Nuclear Station Training Building and the Visitor's Center are designated as alternate locations for the assembly of relocated, evacuated and responding personnel. In addition to the identified near site locations, two additional sites available for staging of personnel were discussed in the applicant's December 22, 2008, response to RAI 13.3-80. The York Operations Center is located approximately 24 km (15 mi) East South East of WLS and the new Duke Energy In-Processing Facility is located approximately 24 km (15 mi) North North East of WLS as described in WLS Emergency Plan, Section II.J.2. The York Operations Center and Duke Energy In-Processing Centers have telecommunication capabilities consistent with their emergency function as a designated relocation center. If necessary, technical support activities can be established in the Duke Energy EOF to support onsite and offsite communication and to coordinate the entry of damage control and engineering teams when safe conditions are established at the site.

The assessment of other nearby hazards that could potentially affect the safety of the Lee Nuclear Station was not addressed in the WLS Emergency Plan. In RAI 94, Question 13.03-88(B), the staff requested that the applicant provide additional information concerning other nearby hazards that could cause a security-based event. In an April 25, 2011, response, the applicant stated that no additional modification is needed for the WLS Emergency Plan based on the analysis of nearby hazards provided in WLS COL FSAR Section 2.2 identifying no hazards that pose a significant risk.

The staff finds the additional information and clarifications provided by the applicant in response to RAI 25, Questions 13.03-61, 13.03-62, 13.03-73(B); RAI 83, Question 13.03-80; and RAI 94, Question 13.03-88(A and B) acceptable because they conform to the guidance in NUREG-0654, NUREG-0696, or NUREG-0737.

In its Interim Finding Report for Reasonable Assurance, FEMA found that the offsite emergency plans are adequate for this planning standard and the associated evaluation criteria in NUREG-0654.

Subject to License Condition (13-7) and License Condition (13-9), the staff finds that the applicant has described, provided, and maintains adequate emergency facilities and equipment to support the emergency response, including a licensee onsite OSC and TSC, and an EOF from which effective direction can be given and effective control can be exercised during an emergency. This includes onsite and offsite radiological and meteorological monitoring systems. The applicant also described provisions to be employed to ensure that the emergency plan, its implementing procedures, and emergency equipment and supplies are kept up-to-date. In addition, the applicant provided for an ERDS data link between the onsite computer system and the NRC Operations Center. The potential effect has been determined on the plant, onsite staffing and augmentation, and onsite evacuation strategies from damage to nearby hazardous facilities, dams, and other nearby sites, in consideration of a security based event.

The staff further finds that WLS DEP 18.8-1, which addresses the new locations of the TSC and OSC, provides for adequate emergency facilities and equipment to support the emergency response. The new locations conform to the guidance in NUREG-0654 except for being within two minutes of the control room. The locations' installation of various and redundant communication systems supports the need for immediate communication between the control room and TSC and provides for adequate communication capability between the control room and TSC.

Conclusion

Subject to License Condition (13-7) and License Condition (13-9), the staff concludes that the information provided in the WLS COLA is consistent with the guidelines in NUREG-0654, Planning Standard H and the guidance in NUREG-0696, NUREG-0737, and NSIR/DPR-ISG-01. Therefore, the staff finds the information acceptable and meets the relevant requirements of 10 CFR 50.47(b)(8) and 10 CFR Part 50, Appendix E, Sections IV.E.8, IV.G, VI.1 and VI, insofar as the information describes the essential elements of advanced planning and the provisions made to cope with emergency situations.

13.3.4.9 Accident Assessment

As stated in NUREG-0654, Planning Standard I, “Accident Assessment,” 10 CFR 50.47(b)(9) requires the use of adequate methods, systems, and equipment for assessing and monitoring the actual or potential offsite consequences of a radiological emergency condition. In addition, 10 CFR Part 50, Appendix E, Section IV.A.4 requires the identification of persons within the licensee organization who will be responsible for making offsite dose projections, and a description of how these projections will be made and the results transmitted to State and local authorities, the NRC, and other appropriate governmental entities. 10 CFR Part 50, Appendix E, Section IV.B requires a description of the means to be used for determining the magnitude of, and for continually assessing the impact of, the release of radioactive materials. 10 CFR Part 50, Appendix E, Section IV.E.2 requires that adequate provisions shall be made and described for emergency facilities and equipment, including equipment for determining the magnitude of, and for continuously assessing the impact of, the release of radioactive materials to the environment.

In WLS Emergency Plan, Section II.I, “Accident Assessment,” the applicant described the methods, systems, and equipment available for assessing and monitoring actual or potential consequences of a radiological emergency. The staff reviewed this section, as well as other relevant portions of the application, to determine whether the application conforms to the applicable guidance and complies with the pertinent regulatory requirements. The staff’s primary focus was to evaluate the emergency plan against NUREG-0654, Planning Standard I, which provides the detailed evaluation criteria that the staff should consider to determine whether the emergency plan meets the applicable regulatory requirements in 10 CFR 50.47(b)(9).

WLS Emergency Plan, Section II.I, “Accident Assessment,” briefly describes measuring, monitoring, readout and continuous sampling systems. WLS COL FSAR Section 7.5, “Safety-Related Display Information,” states this section of the referenced AP1000 DCD is incorporated by reference with no departures or supplements. In RAI 25, Question 13.03-62(A), the staff requested that the applicant provide additional information regarding the emergency preparedness-related instrumentation found in the CR that is available for use in emergency classification and dose assessment. In a December 23, 2009, response, the applicant stated that the selection of monitored variables, based on guidance provided in RG 1.97, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants,” is discussed in AP1000 DCD

Tier 2, Section 7.5 and incorporated by reference in the WLS COL FSAR. Instrument design criteria are described in AP1000 DCD Sections 7.5.2, “Variable Classifications and Requirements,” and 7.5.3, “Description of Variables.” AP1000 DCD Section 7.5.4, “Processing and Display Equipment,” discusses the equipment that processes the safety-related display information and make it available to the operator. The applicant stated WLS Emergency Plan, Appendix 2, “Radiological Assessment and Monitoring,” provides information regarding atmospheric transport and diffusion assessment. Plant vent and turbine island vent effluent monitors are discussed in AP1000 DCD, Revision 19, Section 11.5.3, “Effluent Monitoring and Sampling.” The applicant proposed EP-ITAAC 6.1 to test that the means exists to provide initial and continuing radiological assessment throughout the course of an accident.

WLS Emergency Plan, Section II.I.1, “Parameters Indicative of Emergency Conditions,” states that an EPIP, "Emergency Classification," includes the various indications that correspond to the emergency initiating conditions. Plant procedures specify the instruments used to indicate emergency conditions. In RAI 83, Question 13.03-75, the staff requested that the applicant submit a revised EAL scheme based on an approved and endorsed NEI 07-01 document. In a June 12, 2009, response, the applicant stated that a license condition to provide a revised EAL scheme will be developed and submitted to the NRC for approval in accordance with the ITAAC and license condition. (See Section 13.3.4.4, “Emergency Classification System,” of this report for additional information.

WLS Emergency Plan, Section II.I.2, “Plant Monitoring Systems,” describes the methods of making initial and continuing assessments of plant conditions through the course of an event. This section incorporates AP1000 DCD Section 9.3.3, “Primary Sampling System,” dealing with the primary sampling system by reference. The primary sampling system includes a post-accident sampling capability, but it does not include a post-accident sampling system specifically. WLS Emergency Plan, Section II.I.2 also incorporates AP1000 DCD Tier 2, Section 11.5, “Radiation Monitoring.” WLS COL FSAR Section 11.5, “Radiation Monitoring,” supplements this information and lists departures from the AP1000 DCD.

WLS Emergency Plan, Section II.H.5, “On-site Monitoring Systems,” contains a description of the various monitoring systems necessary for initiating emergency measures and performing accident assessment. A supply of portable radiation monitoring and sampling equipment and emergency response equipment (WLS Emergency Plan, Section II.H, “Emergency Facilities and Equipment,” and Appendix 6, “Emergency Equipment and Supplies”) are available. Plant process monitoring systems are described in AP1000 DCD Section 11.5, “Radiation Monitoring,” and the corresponding section of the WLS COL FSAR. Additional information related to accident assessment can be found in WLS Emergency Plan Appendix 2, “Radiological Assessment and Monitoring.”

WLS Emergency Plan, Section II.I.3, “Determination of Source Term and Radiological Conditions,” refers to WLS Emergency Plan, Appendix 2, “Radiological Assessment and Monitoring,” for descriptions of the means for relating various measured parameters, including containment radiation monitor reading, to the source term available for release within plant systems and effluent monitor readings to the magnitude of the release of radioactive materials. WLS Emergency Plan, Appendix 2, Section 3.0, “Conceptual Design Description: Atmospheric Transport and Diffusion Assessment,” lists five basic release types. Four of the release types have fixed radionuclide composition; the user can specify the composition for the fifth types.

The fixed release types are reactor coolant, gap, core damage, and core melt. The source term may also be specified as noble gas or isotope Iodine-131 release rate. Tabulated release mixes are used if either of these options is used. The applicant has proposed EP-ITAAC 6.2 to evaluate the EPIPs to determine that the means exist to identify the source term of releases of radioactive material within plant systems and the magnitude of a release of radioactive materials using plant system parameters and effluent monitors.

In RAI 25, Question 13.03-62(D), the staff requested that the applicant provide additional information on the process used to estimate accident source terms. In a December 23, 2008, response, the applicant stated that WLS Emergency Plan, Appendix 2, "Radiological Assessment and Monitoring," provides a description of the Raddose-V dose assessment model, which is used to analyze offsite doses at Duke Energy facilities. This model provides results that are compatible and consistent with the NRC dose assessment models evaluated during successful emergency plan exercises. The code is maintained current with respect to the facility's physical and operational characteristics and the assumptions and criteria used in the dose consequence analysis performed as part of the regulatory required accident analyses described in WLS COL FSAR Chapter 15, "Accident Analysis." The applicant further stated that Raddose-V does not currently include modeling for WLS, but anticipate modifying the code to include data for WLS or using more advanced assessment capabilities that may be available. The applicant proposed EP-ITAAC 6.2 to analyze the emergency plan implementing procedures for a methodology to determine the source term of the releases within plant systems.

In RAI 25, Question 13.03-62(D)(1), the staff requested that the applicant provide a list of procedures that cover the estimation of accident source terms (radionuclides and activities) and describe the contents of each procedure. In a December 23, 2008, response, the applicant stated that instruction to dose assessors for determining source term and calculating the projected offsite dose to the public using Raddose-V and guidance for completion of Emergency Notification Forms is provided in Duke Energy's corporate procedure for making offsite dose projections.

In RAI 25, Question 13.03-62(D)(2), the staff requested that the applicant identify the person responsible for making source term estimates at various stages of the event. In a December 23, 2008, response, the applicant stated that Dose Assessors in the EOF, under the direction of the Radiological Assessment Manager, are responsible for evaluating source terms until the event is terminated.

In RAI 25, Question 13.03-62(D)(3), the staff requested that the applicant clarify assumptions related to the pathway from the reactor to the environment. In a December 23, 2008, response, the applicant stated that AP1000 DCD Section 15.6, "Decrease in Reactor Coolant Inventory," identifies the following pathways to the environment: (1) a steam generator tube rupture where the pathways may involve the unit vent and main steam isolation valves; (2) a loss of coolant accident inside containment where the pathway involves a loss of containment or design basis leakage with significant increase in reactor coolant activity (unit vent); (3) a loss of coolant accident outside of containment (unit vent); and (4) a fuel handling accident (unit vent).

In RAI 25, Question 13.03-62(D)(4), the staff requested that the applicant discuss whether the assumptions include reduction of the source term to account for filters, sprays, or other safety [features]. In a December 23, 2008, response, the applicant stated that the code used in Raddose-V includes provisions for features that provide for source term reduction specific to the as-built plant. The applicant further stated that features of the Lee Nuclear Station have not yet been added to the code as specified in response to RAI 25, Question 13.3-62(D).

In RAI 25, Question 13.03-62(D)(5), the staff requested that the applicant clarify whether the source term estimates will be modified during the course of the event to account for changes in the release pathway. In a December 23, 2008, response, the applicant stated that the source term available for release is modified within the Raddose-V program to account for processes that reduce or increase the release based on the pathway or release rates. The applicant further stated that features of the Lee Nuclear Station have not yet been added to the code as specified in response to RAI 25, Question 13.03-62(D).

In RAI 25, Question 13.03-62(D)(6), the staff requested that the applicant clarify how long it takes to obtain source term estimates. In a December 23, 2008, response, the applicant stated that 15-minute averages of effluent or accident monitors may be needed to obtain source term estimates for the model, which is currently used for other Duke Energy's operating facilities.

In RAI 25, Question 13.03-62(D)(7), the staff requested that the applicant explain how source term estimates are obtained in the event that the computer-based methods are not available. In a December 23, 2008, response, the applicant stated that laptop computers are available for onsite evaluations if the primary computers are not functional. The applicant further stated that the program can also be run at other Duke Energy facilities if necessary. Source term estimates can be obtained by inserting data provided by the affected site or using default values in the program code for the facility.

WLS Emergency Plan, Section II.I.4, "Relationship Between Effluent Monitor Reading and Exposure and Contamination Levels," introduces the dose assessment capability. WLS Emergency Plan, Appendix 2, Section 3.0, "Conceptual Design Description: Atmospheric Transport and Diffusion Assessment," describes the dose assessment programs. WLS Emergency Plan, Sections 3.3, "Data Acquisition"; 3.4, "Modeling"; and 3.5, "Data Output," of Emergency Plan, Appendix 2, "Radiological Assessment and Monitoring," describe the method of estimating offsite exposures and contamination from monitoring readings and meteorological data using the Raddose-V computer code. In RAI 25, Question 13.03-62(E)(1-6), the staff requested that the applicant provide additional information regarding the dose assessment program. In December 23, 2008, response, the applicant stated that WLS specific procedures have not yet been developed but they will be similar to those in use at CNS. Dispatch of onsite survey teams is discussed in procedure HP/0/B/1009/009, "Guidelines for Accident and Emergency Response," Enclosure 5.1. The dispatch of teams to monitor the particulate and iodine levels present during an emergency is discussed in procedure HP/0/B/1009/007, "In-Plant Particulate and Iodine Monitoring Under Accident Conditions." These procedures were provided as Attachments 2 and 3 to the applicant's response.

In RAI 25, Question 13.03-62(E)(2), the staff requested that the applicant identify the person responsible for making estimates of onsite exposures and contamination. In a December 23, 2008, response, the applicant stated that WLS specific procedures have not yet been developed but they will be similar to those in use at CNS. On-shift staff is responsible for initial emergency response actions as discussed in HP/0/B/1009/009 Section 4.1. This procedure is provided as Attachment 2 to the applicant's response.

In RAI 25, Question 13.03-62(E)(3), the staff requested that the applicant provide a list of procedures that cover the estimation [of] offsite exposures and contamination and summarize the contents of each procedure. In a December 23, 2008, response, the applicant stated that the WLS-specific procedures have not yet been developed but they will be similar to those in use at other Duke Energy facilities. The procedure contains guidance for utilizing the automatic mode for data input, which uses a number of defaults to speed the initial dose assessment process. Dose assessment is performed by the ERO dose assessors in the EOF. The applicant also stated that Raddose-V will be updated with actual plant data to improve the dose estimates. Duke Energy's corporate procedure for making offsite dose projections will be modified to include the Lee Nuclear Station. This procedure is provided as Attachment 1 to the applicant's response.

In RAI 25, Question 13.03-62(E)(4), the staff requested that the applicant identify the person responsible for making estimates of offsite exposures and contamination. In a December 23, 2008, response, the applicant stated that dose assessment will be provided by EOF Dose Assessment personnel reporting to the EOF Director.

In RAI 25, Question 13.03-62(E)(5), the staff requested that the applicant identify how exposure and contamination estimates would be made in the event that the computer method is unavailable. In a December 23, 2008, response, the applicant refers to information provided in response to RAI 25, Question 13.03-62(D)(7).

In RAI 25, Question 13.03-62(E)(6), the staff requested that the applicant describe how exposure and contamination estimated would be adjusted in the event that onsite meteorological data are not available. In a December 23, 2008, response, the applicant refers to information provided in response to RAI 25, Question 13.03-62(C) regarding meteorological data.

WLS Emergency Plan, Section II.D, "Emergency Classification System," discusses WLS standard emergency classification scheme, based on system and effluent parameters, on which affected State and local response organizations may rely for determining initial offsite response measures. WLS Emergency Plan, Section II.H, "Emergency Facilities and Equipment," describes the capability of WLS to assess the magnitude and consequences of releases.

The applicant proposed EP-ITAAC 6.3 to analyze the emergency plan implementing procedures to identify the relationship between effluent monitor readings and the onsite and offsite exposures using various meteorological conditions.

WLS Emergency Plan, Section II.H.6.a, "Access to Data from Monitoring Systems," WLS Emergency Plan, Section II.H.8, "Meteorological Instrumentation and Procedures," and WLS Emergency Plan, Appendix 2, "Radiological Assessment and Monitoring," of the WLS Emergency Plan briefly discuss meteorological data acquisition and evaluation. There is a more detailed discussion in WLS COL FSAR Section 2.3.3, "Onsite Meteorological Measurement Programs." In RAI 25, Question 13.03-62(F), the staff requested that the applicant provide additional information on the acquisition and distribution of the representative meteorological information. In a December 23, 2008, response, the applicant referred to information provided in the response to RAI 25, Questions 13.03-62(B) and 13.03-62(C) regarding distribution of meteorological information to the CR, TSC, and EOF and processes used in the event the primary meteorological data system is unavailable. The applicant proposed EP-ITAAC 6.4 to conduct an inspection to verify that 10-meter and 60-meter wind speeds, wind directions and temperatures are available in the control room, TSC and EOF. WLS Emergency Plan, Section II.I.6, “Determination of Release Rates and Projected Doses When Installed Instruments are Inoperable or Off-Scale,” states that plant implementing procedures establish processes for estimating release rates and doses when instrumentation used for assessments is not available. In addition, two considerations are mentioned by the applicant: field monitoring data and surrogate instrumentation as methods for estimating fuel damage. In RAI 25, Question 13.03-62(G), the staff requested that the applicant provide additional information on surrogate monitoring and estimating fuel damage.

In RAI 25, Question 13.03-62(G)(1), the staff requested that the applicant describe methods to determine release rates and doses when instrumentation used for assessments is inoperable or readings are off-scale, and summarize the contents of each procedure. In a December 23, 2008, response, the applicant stated that release rates can be estimated by using default source term inventories or back calculations from field data both provided in the Raddose-V model.

In RAI 25, Question 13.03-62(G)(2), the staff requested that the applicant identify the person who makes the decision to use alternative methods to estimate release rates and doses. In a December 23, 2008, response, the applicant stated that the Radiation Protection Manager in the TSC or the Radiological Assessment Manager in the EOF would make the decision to use alternative methods for estimating release rates and doses.

In RAI 25, Question 13.03-62(G)(3), the staff requested that the applicant identify the person who estimates release rates in these cases. In a December 23, 2008, response, the applicant stated that the ERO Dose Assessors under guidance of the Radiological Assessment Manager will estimate the release rates, in all cases.

In RAI 25, Question 13.03-62(G)(4), the staff requested that the applicant explain what compensatory measures are taken in the assessment. In a December 23, 2008, response, the applicant stated that necessary or appropriate compensatory measures not already considered in the existing dose assessment procedures and Raddose-V code that are specific to WLS operation will be addressed in the procedures implemented for or to include the WLS when developed.

In RAI 25, Question 13.03-62(G)(5), the staff requested that the applicant describe how release rates are estimated from field monitoring data. In a December 23, 2008, response, the applicant stated that the Raddose-V code uses field data, meteorology, and accident assumptions to back-calculate source term required to result in measured field dose. That source term could then be used to generate a complete dose projection.

In RAI 25, Question 13.03-62(G)(6), the staff requested that the applicant explain what assumptions are made in the process. In a December 23, 2008, response, the applicant stated that any assumptions beyond those provided in WLS COL FSAR Chapter 15 that are specific to the WLS will be determined during the modifications made to Raddose-V or within the development of alternative software.

In RAI 25, Question 13.03-62(G)(7), the staff requested that the applicant explain what the sensitivity of the release rate estimates is to the assumptions. In a December 23, 2008, response, the applicant stated that sensitivities of the release rates will be determined and evaluated based on the site-specific modification to the software determined for the WLS.

WLS Emergency Plan, Section II.I.7, “Field Monitoring Capability,” briefly describes the field monitoring capability. Implementing procedures provide guidance for field monitoring teams’ performance of monitoring activities. Instrumentation typically available for field deployment is listed in WLS Emergency Plan, Appendix 6, “Emergency Equipment and Supplies,” and WLS Emergency Plan, Section II.B, “Onsite Emergency Organization,” WLS Emergency Plan, Table II-2, “Plant Staff Emergency Functions,” indicates that four individuals comprise two teams. Each field monitoring team has a driver and a qualified radiation protection (RP) technician, and the teams should be available for offsite field monitoring within 75 minutes. Field monitoring teams are directed by RP personnel in the TSC.

WLS Emergency Plan, Section II.I.8, “Assessing Hazards Through Liquid or Gaseous Release Pathways,” states that actual or potential magnitude and locations of radiological hazards are assessed by field teams consistent with WLS Emergency Plan, Section II.I.7, “Field Monitoring Capability.” Implementing procedures provide guidance for field monitoring teams’ performance of monitoring activities. Notification and activation of field team personnel described in WLS Emergency Plan, Section II.E, “Notification Methods and Procedures.” Mobilization times are described in WLS Emergency Plan, Section II.B, “Onsite Emergency Organization.” The applicant proposed EP-ITAAC 6.5 to analyze the emergency plan implementing procedures to determine that the means exist to make rapid assessments of actual or potential magnitude of and locations of any radiological hazards through liquid or gaseous release pathways.

WLS Emergency Plan, Section II.I.9, “Measuring Radioiodine Concentrations,” states that equipment typically available to field teams is listed in WLS Emergency Plan, Appendix 6, “Emergency Equipment and Supplies.” This list includes air samplers, appropriate sample media, and analysis equipment, stated to be capable of detecting radioiodine concentrations at or below 10^{-7} microcuries per milliliter under field conditions. The applicant proposed EP-ITAAC 6.6 to verify the capability exists to detect and measure radioiodine concentrations in air in the plume exposure pathway EPZ, as low as 10E-7 μCi/cc under field conditions.

WLS Emergency Plan, Section II.I.10, “Relating Measured Parameters to Dose Rates,” states that WLS Emergency Plan, Appendix 2 describes the methods to relate the measured activity levels to dose rates for the key isotopes listed in NUREG-0654, Table 3 and the provisions to estimate the projected dose based on the actual dose rates. Radiation Protection personnel are responsible for directing implementation of these procedures under emergency conditions. Therefore, in RAI 25, Question 13.03-62(H), the staff requested that the applicant provide additional information on relating measured parameter to dose rates. In a December 23, 2008, response, the applicant provided Duke Energy's corporate procedure to activate the EOF, which provides instructions for preparing protective action recommendations (PARs) to appropriate State authorities. The procedure includes Offsite Protective Action Flowcharts used by Duke Energy at its operating nuclear plants. The flowcharts include radiological dose considerations. The applicant stated that the dose assessment procedures used for the WLS will be similar to those in use at other Duke Energy nuclear plants. The procedure is included as Attachment 1 in the response to RAI 252, Question 13.03-55. The applicant also stated that implementing procedures and programs will be modified to include WLS on a schedule that supports NRC inspection activities and execution of the emergency exercise required by 10 CFR Part 50, Appendix E, Section IV.F.2. The applicant proposed EP-ITAAC 6.7 to test that the means exist to estimate integrated dose and for comparing these estimates with the EPA protective action guides (PAGs).

The staff finds the additional information and clarifications provided by the applicant in response to RAI 25, Question 13.03-55 and all the questions in RAI 25, Question 13.03-62 acceptable because the responses conform to the guidance in NUREG-0654. Accordingly, the staff considers RAI 25, Question 13.03-5 resolved.

In its Interim Finding Report for Reasonable Assurance, FEMA found that the offsite emergency plans are adequate for this planning standard and the associated evaluation criteria in NUREG-0654.

The staff finds that the applicant has described and provided adequate facilities, systems, equipment, and means for assessing and monitoring the actual or potential offsite consequences of a radiological emergency condition, including determining the magnitude of, and continually assessing the impact of, the release of radioactive materials. The applicant also described the capability and resources for field monitoring within the 16-km (10-mi) plume exposure pathway EPZ, and has the methods, equipment, and expertise to rapidly assess actual or potential radiological hazards. This includes the capability to detect and measure radioiodine airborne concentrations within the plume exposure pathway EPZ as low as 1×10^{-7} µCi/cc under field conditions, and to relate the various measured parameters to dose rates for key isotopes and gross radioactivity measurements. In addition, the applicant identified, by position and function to be performed, persons within the licensee organization who will be responsible for making offsite dose projections, and has described how these projections will be made and the results transmitted to State and local authorities, the NRC, and other appropriate governmental entities.

Conclusion

The staff concludes that the information provided in the WLS COLA is consistent with the guidelines in NUREG-0654, Planning Standard I. Therefore, the staff finds the information acceptable and meets the relevant requirements of 10 CFR 50.47(b)(9) and 10 CFR Part 50, Appendix E, Sections IV.A.4, IV.B, and IV.E.2, insofar as the information describes the essential elements of advanced planning and the provisions made to cope with emergency situations.

13.3.4.10 Protective Response

As stated in NUREG-0654, Planning Standard J, "Protective Response," 10 CFR 50.47(b)(10) requires that a range of protective actions have been developed for the plume exposure pathway EPZ for emergency workers and the public. In developing this range of actions, the applicant considered evacuation, sheltering, and as a supplement to these, the prophylactic use of potassium iodide (KI). ETEs have been developed by the applicant. Guidelines for the choice of protective actions during an emergency are developed and in place, and protective actions for the ingestion exposure pathway EPZ appropriate to the locale have been developed. In addition, 10 CFR 50.47(c)(2) and 10 CFR Part 50, Appendix E, Section I require that the size and configuration of the EPZs be determined in relation to local emergency response needs and capabilities, as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries. 10 CFR Part 50, Appendix E, Section IV.I requires the development of a range of protective actions to protect onsite personnel during hostile action to ensure the continued ability of the licensee to safely shut down the reactor and perform the functions of the emergency plan.

In WLS Emergency Plan, Section II.J, “Protective Response,” the applicant described the protective response measures that have been developed to limit radiation exposure of plant personnel and the public following an accident at the WLS. The staff reviewed this section, as well as other relevant portions of the application, to determine whether the application conforms to the applicable guidance and complies with the pertinent regulatory requirements. The staff’s primary focus was to evaluate the emergency plan against NUREG-0654, Planning Standard J, which provides the detailed evaluation criteria that the staff should consider to determine whether the emergency plan meets the applicable regulatory requirements in 10 CFR 50.47(b)(10).

WLS Emergency Plan, Section II.J.1, “Onsite Notification,” describes that individuals within the protected area are notified by the plant public address system and audible warning systems. In high noise areas, other measures may be used. Individuals located outside of the protected area are notified by audible warnings, by Security Force actions, and, if needed, by local law enforcement personnel. Information on the warning systems and response actions are provided through plant training programs, visitor orientation, escort instructions, posted instructions, or within the audible messages. The applicant stated that it has the ability to notify all individuals within the owner controlled area within about 15 minutes of the declaration of an emergency requiring individual responses, such as evacuation and accountability. Additionally, the applicant states that there are methods to notify and alert onsite personnel in a timely manner during a hostile action. The actions are coordinated with Security and maintained in the EPIPs.

In RAI 25, Question 13.03-63(A), the staff requested that the applicant: (1) clarify other measures to be used for notification of individuals in high noise areas; and (2) provide information on timing to notify the people outside the protected area. In a December 23, 2008, response, the applicant stated audibility problems encountered on evacuation of personnel from high-noise areas for its fleet were addressed in previous Duke Energy responses to IE Bulletin No. 79-18, “Audibility Problems Encountered on Evacuation of Personnel from High-Noise Areas.” A consistent process will be used for the WLS, in that the plant alarm system will use the telephone page system amplifiers and speakers that will be assessed in the as-built plant to determine if additional measures or equipment is necessary. The accountability process has been proven at other Duke Energy facilities. The CNS’s procedure for site assembly and evacuation was provided as Attachment 1 to the response to RAI 25, Question 13.03-63(A). The applicant stated that the site alarm system along, with surveillances of the owner controlled area, will be adequate to assemble and evacuate nonessential personnel. The applicant proposed EP-ITAAC 7.1 to test that the means exist to warn and provide instructions to onsite individuals of an emergency, including those in areas controlled by the operator, including: (a) employees not having emergency assignments; (b) visitors;(c) contractor and construction personnel; and (d) other persons who may be in the public access areas, on or passing through the site, or within the owner controlled area using the plant public announcement system within the protected area or an audible warning system for those who may be outside the protected area and within the owner controlled area.

WLS Emergency Plan, Section II.J.2, “Evacuation Routes and Transportation,” states that evacuation routes are determined by the Shift Manager or Emergency Coordinator using available information on conditions. Provisions for evacuation of onsite individuals include evacuation by private automobile. WLS Emergency Plan, Section II.J.2, also states that the Security Forces will arrange transportation for those individuals without cars. The designated relocation site will have decontamination and contamination control capability and equipment. In adverse conditions, affected individuals will be directed to a safe onsite area (as determined by the Emergency Coordinator). Relocation centers were not identified in the WLS Emergency Plan. Therefore, in RAI 25, Question 13.03-63(B), the staff requested that the applicant: (1) explain why prearranged routes, coordinated with the State and local governments did not exist; and (2) provide information on the type of transportation the Security Force will have available to transport people without cars. In RAI 25, Question 13.03-80, the staff requested that the applicant identify where the relocation center will be established. Additionally, if the relocation center is not within the control of Duke Energy, when will the letters of agreement (LOAs) be available? In a December 23, 2008, response to RAI 25, Question 13.03-63(B), the applicant provided additional information regarding the role of the Security Force in site evacuation, stating that if an individual does not have access to personal transportation, either the affected individual or the Security Force will make arrangements for transportation (ride-share) with another affected individual. In a December 11, 2009, response to RAI 25, Question 13.03-80, the applicant identified the York County Operations Center and a planned Duke Energy In-Processing Facility as designated locations for relocated site personnel. Both locations will have personnel decontamination capabilities and both are controlled by Duke Energy. The locations of reception centers and shelter areas are not finalized.

In Part 10 of the WLS COL application, the applicant proposed License Condition 4 “Emergency Planning Actions,” which states that reception centers and relocation sites will be identified and LOAs will be obtained prior to the full participation exercise. The staff refers to this as License Condition (13-4). WLS Emergency Plan, Appendix 4 contains maps that include proposed reception centers, relocation site, and pre-selected radiological sampling and monitoring locations.

License Condition (13-4):

> Prior to the full participation exercise to be conducted in accordance with the requirements of 10 CFR Part 50, Appendix E, Duke Energy shall identify the specific locations of the reception centers and relocation sites and shall obtain Letters of Agreement for locations not under the Duke Energy’s control.

The staff evaluated the proposed license condition and finds it is acceptable because it is consistent with the guidance in NUREG-0654, Planning Standard J and the requirements in 10 CFR Part 50.47(b)(10) and 10 CFR Part 50, Section IV.I, Appendix E.

WLS Emergency Plan, Section II.J.3, “Personnel Monitoring and Decontamination,” states that the Emergency Coordinator directs contamination monitoring of personnel, vehicles, and personal property arriving at relocation sites. The procedures and criteria for monitoring are not addressed in the plan. Therefore, in RAI 25, Question 13.03-63(C), the staff requested that the applicant: (1) provide a summary of the decontamination capabilities and equipment; and (2) provide information to identify the criteria for monitoring. In a December 23, 2008, response, the applicant stated procedures for personnel and vehicle monitoring at relocation sites will be consistent with that in use at other Duke Energy facilities. The applicant provided CNS procedures for personnel/vehicle monitoring and equipment inspection and inventory as examples.

WLS Emergency Plan, Section II.J.4, “Non-Essential Personnel Evacuation and Decontamination,” states that non-essential personnel will be evacuated in accordance with WLS Emergency Plan, Section II.J.2, “Evacuation Routes and Transportation,” and that the appropriate equipment and supplies will be moved from WLS to the decontamination site. All public visitors will be evacuated whenever the possibility exists that a visitor may be exposed to levels exceeding 2 millirem per hour or 1 times the derived air concentration (DAC) for an unrestricted area.

WLS Emergency Plan, Section II.J.5, “Personnel Accountability,” states that the WLS maintains the capability for all individuals within the protected area to be accounted and missing individuals identified within 30 minutes following initiation of evacuation and accountability processes and to maintain continuous accountability for any individual within the protected area consistent with the requirements of the WLS Security Plan.

Furthermore, WLS Emergency Plan, Section II.J.5 addresses protective measures in the event of a hostile attack against the site. The section states that in the event of a hostile attack against the site, conditions may dictate initiation of protective measures other than personnel assembly, accountability, and evacuation. The Emergency Coordinator will make decisions regarding appropriate protective measures based on evaluation of site conditions, including input from the security force. If, based on the judgment of the Emergency Coordinator, personnel assembly, accountability, and evacuation may result in undue hazards to site personnel, the Emergency Coordinator may direct other protective measures, including:

- evacuation of personnel from areas and buildings perceived as high value targets
- site evacuation by opening, while continuing to defend, security gates
- dispersal of key personnel
- onsite sheltering
- staging of ERO personnel in alternate locations pending restoration of safe conditions
- implementation of accountability measures following restoration of safe conditions

WLS Emergency Plan, Section II.J.6, “Protective Measures,” describes the provisions for respiratory protection, ventilation systems, use of protective clothing, and individual thyroid protection. The plan states that measures are taken to minimize ingestion and or inhalation of radionuclides to minimize exposure below limits specified in 10 CFR Part 20, “Standards for Protection against Radiation,” WLS Emergency Plan, Appendix B, “Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage.” However, the protective measures used are not identified. WLS Emergency Plan, Section II.J.6, “Protective Measures,” states that self-contained breathing apparatus (SCBA) are used in locations where there is low oxygen or fires. Other respiratory protection is available and issued by Radiation Protection personnel or Safety and Health staff. The plan does not address training for use of SCBAs or other respiratory protection equipment, the number of respirators available, or the maintenance of the equipment. The criteria for use of protective clothing are given; however, the location of the equipment and inventory is not addressed to ensure that the protective clothing is available when needed. The use of radioprotective drugs (potassium iodide [KI]) is also mentioned, and there are no criteria for issuance, how and where it is stored and inventoried, and who makes the decision on issuance. Therefore, in RAI 25, Question 13.03-63(D), the staff requested that the applicant provide additional information related to: (1) measures used to minimize ingestion and inhalation of radionuclides; (2) training in the use of respiratory equipment as well as the inventory and maintenance of the respiratory equipment; (3) storage and inventory of the protective clothing; and (4) storage and use of radioprotective drugs. In a December 23, 2008, response, the applicant stated that radiation protection personnel will be responsible for monitoring the safety of personnel during a site assembly or site evacuation, which includes contamination monitoring at site exits, and monitoring of work locations for personnel remaining on site. A description of their monitoring process was provided. Respiratory protection will be prescribed for workers that are trained, qualified, and fit tested in accordance with the respiratory protection program discussed in WLS COL FSAR Chapter 12, “Radiation Protection.” Details related to procedures and quantity or locations of respiratory equipment are not available. The applicant expects procedures will be similar to CNS's procedure for inspection and inventory of emergency equipment provided in response to RAI 25, Question 13.03-61. Procedures will be completed in accordance with 10 CFR Part 50, Appendix E, Section IV.F.2. With regard to protective clothing, the applicant stated that they are maintained in the change rooms inside the Radiation Control Area (RCA) and inventoried each quarter. A discussion related to the issuance of protective clothing was included. Additional information related to protective clothing was also provided in the response to RAI 25, Question 13.03-61. With regard to the distribution of KI, the applicant stated Duke Energy’s corporate procedure for distribution of KI provides information related to distribution, storage, and supply of KI tablets. The WLS Radiation Protection Manager shall evaluate the distribution of KI. The KI is distributed only to prevent a significant uptake that would result in a committed dose equivalent (CDE) of 5 rem or more to the thyroid.

WLS Emergency Plan, Section II.J.7, “Protective Action Recommendations and Bases,” describes the process in which WLS develops protective action recommendations (PARs) and issues them to the affected State and local governments. General public PARs are based on plant conditions (EALs), dose projection results or both. Plant and dose based PARs are developed from the guidance contained in NUREG-0654, Supplement 3, “Criteria for Protective Action Recommendations for Severe Accidents.” The process includes EALs corresponding to projected dose to the general public and with the recommendations set forth in EPA-520/1-75-001, “Manual of Protective Action Guides and Protective Actions for Nuclear Incidents.” The dose projection results are compared to PAGs shown in WLS Emergency Plan, Table II-3. The PAGs are derived from EPA-400-R-92-001, “Manual of Protective Action Guides and Protective Actions for Nuclear Incidents.” Based on the comparison, PARs are developed by the Radiological Assessment Manager. If the recommendations suggest sheltering or evacuation of the public around the plant, The Radiological Assessment Manager informs the Emergency Coordinator or EOF Director so that he or she notifies the affected States and counties. The Emergency Coordinator or EOF Director is responsible for recommending offsite protective actions to the affected States and counties. WLS is required to issue PARs within 15 minutes of declaring a General Emergency to the affected States and local governments. The State and local governments are responsible for notification of the public and implementation of protective measures. (Emergency Action Levels are discussed in Section 13.3.4.3 of this report. Emergency Notification is discussed in Section 13.3.4.5 of this report, and Accident Assessment is discussed in Section 13.3.4.9 of this report.)

WLS Emergency Plan, Section II.J.8, “Evacuation Time Estimates,” of the WLS Emergency Plan states that the ETE Report is included in the COLA as supplemental information to the WLS Emergency Plan, and the updated population distribution and ETEs are summarized in WLS Emergency Plan Appendix 4, which includes the updated ETE’s Executive Summary. ETEs are a factor considered in the development of offsite PARs, and are provided to the States and local governmental authorities for use in developing offsite protective action strategies. The ETE Report provides maps of the plume exposure pathway EPZ, which illustrate the population distribution around the WLS, evacuation areas and routes, and locations of assembly areas. A summary of the staff’s detailed review of the ETE Report is included in Section 13.3.4.17 of this report.

WLS Emergency Plan, Section II.J.10.a, “Protective Measure Implementation,” states that maps of evacuation routes, evacuation areas, and general locations of shelter areas and relocation sites are provided in WLS Emergency Plan, Appendix 4, “Evacuation Time Estimate.” In RAI 25, Question 13.03-63(E), the staff requested that the applicant provide the specific location of the shelter areas and relocation sites and the pre-identified monitoring locations or provide an EP-ITAAC for when those locations would be identified. In a December 23, 2008, response to RAI 83, Question 13.03-80, the applicant identified the York County Operations Center and a planned Duke Energy In-Processing Facility as designated locations for relocated site personnel and provided wording for WLS Emergency Plan, Section II.J.2.a that will be included in a future revision of the WLS Emergency Plan. Additional information related to reception centers can be found in the evaluation of Section II.J.2, above.

WLS Emergency Plan, Section II.J.10.b, “Protective Measures Implementation,” states that maps of the EPZ population distribution around the facility by evacuation area and in a sector format can be found in WLS Emergency Plan, Appendix 4.

WLS Emergency Plan, Section II.J.10.c. “Protective Measures Implementation,” states that the Alert and Notification System will be used to warn the public within the 16-km (10-mi) EPZ and that this is the responsibility of State and local officials. The Alert and Notification System is described in Section 13.3.4.5 of this report.

WLS Emergency Plan, Section II.J.10.c. “Protective Measures Implementation,” n states that recommended protective actions are based on the guidance provided in NUREG-0654, Supplement 3 4, WLS Emergency Plan, Section II.J.8, and WLS Emergency Plan, Appendix 4.

The staff finds the additional information and clarifications provided by the applicant in response to RAI 25, Questions 13.03-61 and 13.03-63 and RAI 83, Question 13.03-80 acceptable because they conform to the guidance in NUREG-0654. Accordingly, the staff considers RAI 25, Questions 13.03-61 and 13.03-63 and RAI 83, Question 13.03-80 resolved.

In its Interim Finding Report for Reasonable Assurance, FEMA found that the offsite emergency plans are adequate for this planning standard and the associated evaluation criteria in NUREG-0654.

Subject to License Condition (13-4), the staff finds that the applicant developed a range of protective actions for the 16-km (10-mi) plume exposure pathway EPZ for emergency workers and the public, including consideration of evacuation, sheltering, and the prophylactic use of KI. The staff finds that the applicant has developed guidelines for the choice of protective actions during an emergency that are consistent with Federal guidance, including protective actions for the 80-km (50-mi) ingestion exposure pathway EPZ that are appropriate to the locale. The size and configuration of the EPZs have been determined in relation to local emergency response needs and capabilities, as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries. In addition, the staff finds that the applicant has developed a range of protective actions to protect onsite personnel during hostile action. Development of ETEs is addressed in Section 13.3.4.17 of this report.

Conclusion

Subject to License Condition (13-4), the staff concludes that the information provided in the WLS COLA is consistent with the guidelines in NUREG-0654, Planning Standard J. Therefore, the staff finds the information acceptable and meets the relevant requirements of 10 CFR 50.47(b)(10), 10 CFR 50.47(c)(2), and 10 CFR Part 50, Appendix E, Sections I and IV.I, insofar as the information describes the essential elements of advanced planning and the provisions made to cope with emergency situations.

13.3.4.11 Radiological Exposure Control

As stated in NUREG-0654, Planning Standard K, "Radiological Exposure Control," 10 CFR 50.47(b)(11) requires that the means for controlling radiological exposures in an emergency be established for emergency workers. The means for controlling radiological exposures shall include exposure guidelines consistent with EPA "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," EPA 400-R-92-001, May 1992. In addition, 10 CFR Part 50, Appendix E, Section IV.E.3 requires that adequate provisions shall be made and described for emergency facilities and equipment, including facilities and supplies at the site for decontamination of onsite individuals.

In WLS Emergency Plan, Section II.K, "Radiological Exposure Control," the applicant described the emergency exposure limits for emergency workers, including decisions and efforts made to minimize exposures. The staff reviewed this section, as well as other relevant portions of the application, to determine whether the application conforms to the applicable guidance and complies with the pertinent regulatory requirements. The staff's primary focus was to evaluate the emergency plan against NUREG-0654, Planning Standard K, which provides the detailed evaluation criteria that the staff should consider to determine whether the emergency plan meets the applicable regulatory requirements in 10 CFR 50.47(b)(11).

WLS Emergency Plan, Section II.K.1, "On-site Exposure Guidelines and Authorizations," discusses implementation of guidelines from EPA-400-R-92-001, Table 2.2, "Guidance on Dose Limits for Workers Performing Emergency Services," and are listed in Table II-4, "Emergency Worker Exposure Guidelines." In consultation with senior radiological protection personnel, the Emergency Coordinator is responsible for authorizing onsite emergency exposures that would result in doses in excess of occupational dose limits in 10 CFR Part 20. Exposures in excess of 10 CFR Part 20 limits are limited to individuals who are properly trained and knowledgeable of the tasks to be performed and the risks associated with the exposures. Selection criteria for volunteer emergency workers are outlined. In the absence of extenuating circumstances listed in WLS Emergency Plan, Table II-4, routine dose limits are applied to activities including those listed above.

WLS Emergency Plan, Section II.K.2, “Radiation Protection Program (RPP),” of the WLS Emergency Plan refers to WLS COL FSAR Chapter 12, “Radiation Protection,” for a description of the WLS Radiation Protection Program (RPP), which is stated to be consistent with 10 CFR Part 20. WLS Emergency Plan, Section II.K.1, “On-site Exposure Guidelines and Authorizations,” describes the provisions made for implementation of emergency exposure guidelines. The RPP in the WLS COL FSAR incorporates by reference material from the AP1000 DCD and NEI 07-08, “Generic FSAR Template Guidance for Ensuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA),” to support ALARA principles for exposure criteria, and NEI 07-03, “Generic FSAR Template Guidance for Radiation Protection Program Description,” Appendix 12AA, to develop RPP. In RAI 25, Questions 13.03-64(A) and 13.03-64(B), the staff requested that the applicant provide a summary of the occupational RPPs outlined in the WLS COL FSAR, the AP1000 DCD, NEI 07-08, and NEI 07-03. In response to RAI 13.03-64(A), the applicant provided a description of their procedure for requesting exposures in excess of occupational dose limits. The applicant also provided CNS’s procedure for emergency worker dose extension as an example. The applicant expects that a similar process will be established for the WLS via implementing procedures that are to be developed on a schedule that supports NRC inspection activities and execution of the emergency exercise required by 10 CFR Part 50, Appendix E, Section IV.F.2. In a December 23, 2008, response to RAI 25, Question 13.03-64(B), the applicant stated that a summary of the WLS RPP is provided in WLS COL FSAR Appendix 12AA, “Radiation Protection Program Description.” Milestones for the development of the RPP are provided in WLS COL FSAR Table 13.4-201. Procedures are discussed in WLS COL FSAR Section 13.5.2.2.1, “Plant Radiation Protection Procedures.” The processes for authorizing and implementing emergency dose constraints consistent with EPA guidance are discussed in WLS Emergency Plan, Section II.K. The applicant also stated that compliance with the RPP is maintained under emergency conditions. Procedures are discussed in more detail in the response to RAI 25, Question 13.03-64(A). The applicant further stated that variations from routine radiation protection practices may be implemented on a case-by-case basis, consistent with ERO management direction and the provisions of 10 CFR 20.1001(b).

WLS Emergency Plan, Section II.K.3, “Dosimetry and Dose Assessment,” states that self-reading and cumulative type dosimeters are provided to all personnel involved in emergency onsite response. Dose records are maintained and checked throughout the emergency. A personnel radiation dosimetry program with capability to determine both external and internal doses consistent 10 CFR Part 20 is maintained. The external dosimetry program includes provisions and requirements for use of both permanent record and self-reading dosimeters. EPIPs establish requirements for distributing dosimeters to emergency responders, including individuals from offsite locations. Internal doses are estimated with whole body counting and/or in-vitro sampling and analysis routines. Dose assessment capabilities are available on a 24-hour per day basis. Procedures related to external and internal dosimetry are mentioned. WLS Emergency Plan, Section II.H.5, “On-site Monitoring Systems,” states that an adequate supply of portable radiation monitoring equipment is maintained at the site including dedicated emergency response equipment. A generic description of this equipment is provided in WLS Emergency Plan, Appendix 6, “Emergency Equipment and Supplies.”

Furthermore, of the WLS Emergency Plan, Section II.K.3, “Dosimetry and Dose Assessment,” states, “Station procedures establish guidance for wearers to periodically read their self-reading dosimeters….” and, “Duke Energy maintains individual dose records in accordance with the requirements of 10 CFR Part 20 and the radiation protection program and its supporting procedures." In RAI 25, Question 13.03-64(C), the staff requested that the applicant provide a list and summary of applicable implementing procedures. In a December 23, 2008, response, the applicant stated that WLS provides and distributes self-reading and cumulative type dosimeters to personnel involved in emergency onsite response regardless of their affiliation. Dosimetry is available at the single point access in the operating facilities. Distribution of dosimetry to TSC and OSC personnel is discussed in the facility activation procedures. Dosimetry is also available for NRC personnel if needed. Radiation protection personnel are assigned to locations to assist and support this effort. Requirements for determining internal and external doses are established by the RPP. When instrument failure or an inadvertent contamination event occurs that requires dose analysis, support can be provided by unaffected Duke Energy facilities. Duke currently maintains procedure SH/0/B/2001/001, which determines dose received while working at a Duke Energy facility that will also be used at WLS. In RAI 25, Question 13.03-63(D), the staff requested that the applicant provide a description or summary of contingency plans for dosimetry services (including recordkeeping), loss of power, instrument failure, inadvertent contamination, etc. In a December 23, 2008, response to RAI 13.03-64(D), the applicant stated that the Dose Records Coordinator Supervisor in the OSC is responsible for maintaining the emergency dose records in accordance with the OSC Activation Procedure. The applicant included applicable portions of the CNS procedure as Attachment 2 to the response. The applicant anticipates that a similar procedure will be developed for the WLS. The applicant further stated that immediate approximations of external dose may be derived from self-indicating dosimeters during an emergency. Records of dosimeter readings may be maintained on log sheets or other record form. Individual dose records are maintained on plant computer systems. If the records are not available during an emergency, the OSC activation procedures requires that copies of the Daily Dose Report be gathered for the TSC and OSC upon activation. The WLS COL FSAR addresses radiation protection procedures as discussed in the response to RAI 25, Question 13.03-64(A).

WLS Emergency Plan, Section II.K.5.a, "Decontamination Action Levels," states that the applicant implements procedures and has supplies. WLS Emergency Plan, Section II.K.5.a does not state what the decontamination levels are, who decides how and when to decontaminate, etc. Therefore, in RAI 25, Question 13.03-64(C), the staff requested that the applicant provide a list and summary of applicable implementing procedures. The WLS Emergency Plan does not reference the RPP in this area or describe any procedures related to decontamination. Therefore, in RAI 25, Question 13.03-64(A), the staff requested that the applicant provide a summary of the occupational RPPs outlined in the WLS COL FSAR, the AP1000 DCD, NEI 07-08, and NEI 07-03. Additional information received in a December 23, 2008, response to RAI 25, Questions 13.03-64(A) and 13.03-64(C) and are summarized in the evaluations of Section K.2 and Section K.3.b, of this report.

WLS Emergency Plan, Section II.K.5, "Decontamination Action Levels," states that the applicant implements procedures to decontaminate onsite emergency personnel wounds, etc., and refers to the general list of decontamination supplies found in WLS Emergency Plan, Appendix 6, "Emergency Equipment and Supplies." In RAI 25, Question 13.03-64(E), the staff requested that the applicant provide a list and summary of applicable implementing procedures related to responsibilities for maintenance, inventories, waste disposal, and locations of decontamination supplies and decontamination of wounds. In a December 23, 2008, response, the applicant provided a definition for contaminated based on their procedures and also procedures for surveying equipment or items and personnel, decontamination of equipment and the return of equipment to normal use. The applicant also stated that the applicant follows Electric Power

Research Institute (EPRI), “Guidelines for Industry Response to Personnel Contaminants.” Levels used at WLS will be consistent with those used at other Duke Energy nuclear stations. Decontamination methods are established in radiation protection procedures and are implemented under the direction of trained radiation protection personnel. The WLS COL FSAR addresses the RPP and procedures with respect to decontamination as discussed in the response to RAI 25, Question 13.03-64(B). In RAI 25, Question 13.03-64(F), the staff requested that the applicant provide a summary of the occupational RPPs outlined in the WLS COL FSAR, the AP1000 DCD, NEI 07-08, and NEI 07-03. In a December 23, 2008, response, the applicant stated that the WLS COL FSAR addresses the RPP as discussed in the response to RAI 25, Question 13.03-64(B). A description of personnel and equipment decontamination facilities and the means for handling radioactive waste is provided in AP1000 DCD Section 1.2, “Definitions.” The applicant expects that the bulk of the emergency equipment and supplies will be stored in the established ERFs. Additional supplies may be stored at locations convenient for use by emergency response personnel, for example, within the radiological control area, access areas, and decontamination areas. The applicant will determine initial storage locations based on an assessment of plant layout and their experience operating nuclear power plants. Locations may be changed based on assessments of plant emergency operations, drills, and exercises. WLS Emergency Plan, Section II.K.5.b, “Decontamination Action Levels,” states that WLS implements procedures for decontamination of onsite emergency personnel. In a December 23, 2008, response to RAI 25, Question 13.03-63, the applicant stated the hot machine shop (Room 40358) will contain decontamination equipment.

WLS Emergency Plan, Section II.K.6.a, “Contamination Control Measures,” discusses access control in the event of an emergency by stating that requirement for site access control is established in the WLS COL FSAR and Security Plan. State and local agencies will control access to the owner-controlled area consistent with State and local plans. In addition, the Station Security Force will control entry to the protected area in the event of an emergency. In RAI 25, Question 13.03-64(G), the staff requested that the applicant provide a list and summary of applicable implementing procedures. In a December 23, 2008, response, the applicant stated that access to the protected area is maintained by the Security Force. The security plans and associated procedures are discussed in WLS COLA Part 8, “Safeguards.” Milestones associated with the implementation of the security program are presented in WLS COL FSAR Table 13.4-201. WLS COL FSAR Chapter 12, “Radiation Protection,” describes the RPP, applicable to contamination control measures, consistent with the requirements of 10 CFR Part 20. WLS COL FSAR Appendix 12AA provides a summary of the WLS RPP; WLS COL FSAR Table 13.4-201 addresses milestones associated with the development of the RPP; and WLS COL FSAR Section 13.5.2.2.1 provides a discussion of radiation protection procedures. In RAI 83, Question 13.03-81, the staff requested that the applicant address the control of access to contaminated areas in the WLS Emergency Plan. In a December 23, 2008, response, the applicant stated that the OSC radiation protection staff, when activated, controls access to RCAs, assesses onsite radiological conditions, and initiates contamination control for the protected area. In addition, the OSC monitors radiological status and provides for radiological monitoring inside the protected area; and, based on survey results identifies travel routes and assigns personnel protective and monitoring equipment that limits or controls access to contaminated areas. The applicant updated the WLS Emergency Plan in the response to RAI 83, Question 13.03-81.

WLS Emergency Plan, Section II.K.6.b, “Contamination Control Measures,” states that Nuclear Supply Chain Personnel will make arrangements for transport of non-contaminated offsite supplies in event of contamination. In RAI 25, Question 13.03-64(H), the staff requested that the applicant provide a list and summary of applicable implementing procedures. In a December 23, 2008, response, the applicant stated that procurement support located in the EOF is responsible for ensuring adequate supplies of food and water are available to the ERO. Food and water would be made available onsite through acquisition of supplies under the applicant's commercial arrangements and subsequent transportation of supplies to the site, using either vendor or the applicant-supplied transport. The applicant expects that distribution of food and water under emergency conditions would be made on an ad-hoc basis. The applicant also stated that procedures are likely to be limited to the existing Duke Energy corporate procedure, assigning responsibility to the procurement support assigned to the EOF. The applicant will modify Duke Energy's corporate procedure describing EOF procurement support to incorporate WLS on a schedule that supports execution of the emergency exercise required by 10 CFR Part 50, Appendix E, Section IV.F.2.

WLS Emergency Plan, Section II.K.6.c, "Contamination Control Measures," states that areas and items are permitted to return to normal use following conduct of appropriate surveys and verification that the contamination levels meet criteria specified in the RPP or its supporting procedures. In RAI 25, Question 13.03-64(I), the staff requested that the applicant provide additional information related to radiological surveys and to summarize the RPP criteria for decontamination. In a December 23, 2008, response, the applicant stated that contamination levels and decontamination are discussed in the response to RAI 25, Question 13.03-64(E). The applicant also stated that the WLS will use the same radiological guidance followed at all existing Duke Energy nuclear stations. The list of procedures in use at Duke Energy's existing nuclear stations to address decontamination and the release of previously contaminated areas and items to normal was provided. The applicant expects that similar procedures will be developed or corporate procedures expanded to account for WLS.

WLS Emergency Plan, Section II.K.7, "Decontamination of Relocated Lee Nuclear Station Personnel," states that WLS makes provisions for protective clothing, contamination monitoring, and decontamination at the designated relocation site. A general description of the equipment and supplies that are typically available is included in WLS Emergency Plan, Appendix 6. In RAI 25, 13.03-64(J), the staff requested that the applicant provide additional information requesting how WLS would use decontamination equipment and facilities, personnel and vehicle monitoring. In a December 23, 2008, response, the applicant stated that the WLS would use procedures similar to those in use at other Duke Energy facilities. Procedures for CNS were provided as examples. Procedures will be available in accordance with 10 CFR Part 50, Appendix E, Section IV.F.2. The applicant proposed License Condition (13-4), which states that reception centers and relocation sites will be identified and LOAs will be obtained prior to the full participation exercise. The license condition is evaluated in Section 13.3.4.10 of this report,

The staff finds the additional information and clarifications provided by the applicant in response to RAI 25, Question 13.03-64 and RAI 83, Question 13.03-81 acceptable because they conform to the guidance in NUREG-0654. Accordingly, the staff considers RAI 25, Question 13.03-64 and RAI 83, Question 13.03-81 resolved.

In its Interim Finding Report for Reasonable Assurance, FEMA found that the offsite emergency plans are adequate for this planning standard and the associated evaluation criteria in NUREG-0654.

The staff finds that the applicant has established the means to control radiological exposures for emergency workers in a way consistent with the exposure guidelines in EPA 400-R-92-001. In addition, the applicant made and described adequate provisions for emergency facilities and equipment, including facilities and supplies for monitoring and decontamination of onsite and relocated personnel, vehicles, and other affected materials, and has established appropriate contamination control measures.

Conclusion

The staff concludes that the information provided in the WLS COLA is consistent with the guidelines in NUREG-0654, Planning Standard K. Therefore, the staff finds the information acceptable and meets the relevant requirements of 10 CFR 50.47(b)(11) and 10 CFR Part 50, Appendix E, Section IV.E.3, insofar as the information describes the essential elements of advanced planning and the provisions made to cope with emergency situations.

13.3.4.12 Medical and Public Health Support

As reflected in NUREG-0654, Planning Standard L, "Medical and Public Health Support," 10 CFR 50.47(b)(12) requires that arrangements be made for medical services for contaminated injured individuals. In addition, 10 CFR Part 50, Appendix E, Section IV.E requires facilities and medical supplies at the site for appropriate emergency first aid treatment, and arrangements for medical service providers qualified to handle radiation emergencies onsite. Arrangements are also required for transportation of contaminated injured individuals from the site to specifically identified treatment facilities outside the site boundary.

In WLS Emergency Plan, Section II.L, "Medical and Public Health Support," the applicant described the arrangements for medical services for contaminated injured personnel at the WLS. The staff reviewed this section, as well as other relevant portions of the WLS COLA to determine whether the application conforms to the applicable guidance and complies with the pertinent regulatory requirements. In this evaluation, the staff's primary focus was to evaluate the emergency plan against NUREG-0654, Planning Standard L. Planning Standard L provides the detailed evaluation criteria that the staff should consider to determine whether the emergency plan meets the applicable regulatory requirements in 10 CFR 50.47(b)(12).

WLS Emergency Plan, Section II.L.1, "Hospital and Medical Support," states that an agreement has been established with Piedmont Medical Center (PMC) to provide medical services for injured personnel. Radiation monitoring equipment, dosimeters, and protective clothing are available at PMC. The PMC has the capability to evaluate and handle contaminated victims due to training courses supported by Duke Energy. Radiation protection personnel may accompany victims to support the radiological aspects of the treatment. Periodic drills, exercises, and material support are provided consistent with agreements developed with medical support providers addressed in Section II.N, "Exercise and Drills," and certification letters in WLS Emergency Plan, Appendix 7, "Certification Letters." In RAI 25, Question 13.03-65, the staff requested that the applicant discuss when the agreements will be finalized between Duke Energy and the medical support providers. In a December 23, 2008, response, the applicant proposed License Condition (13-4) and stated LOAs will be established and incorporated into the WLS Emergency Plan prior to receipt of nuclear fuel at the site. The license condition is evaluated in Section 13.3.4.10 of this report

WLS Emergency Plan, Section II.L.2, “On-site First Aid Capability,” states that a trained Medical Emergency Response Team (MERT) is maintained at the site to provide 24-hour first aid support. The MERT personnel are Department of Transportation (DOT) first responder trained. Medical services are also available from Upstate Carolina Medical Center (ambulance) and Draytonville-McKown Mountain-Wilkinsville Volunteer Fire Department. Duke Energy provides for First Aid Team readiness through training consistent with WLS Emergency Plan, Section II.O, “Radiological Emergency Response Training,” and drills and exercises consistent with WLS Emergency Plan, Section II.N. WLS Emergency Plan, Appendix 6, “Emergency Equipment and Supplies,” provides a brief description of first aid supplies and equipment.

WLS Emergency Plan, Section II.L.4, “Medical Emergency Transportation,” states that initial offsite support for medical emergencies is provided by the Draytonville-McKown Mountain-Wilkinsville Volunteer Fire Department, and Upstate Carolina Center Emergency Medical Services (UCCEMS) provides transport for non-contaminated injured personnel. However, WLS Emergency Plan, Appendix 7 contains a letter of agreement with UCCEMS to suggest that that they will provide ambulance services to transport contaminated injured personnel.

WLS Emergency Plan, Section II.L.4 also states that PMC provides ambulance services for transport of contaminated personnel to PMC. Contaminated injured personnel are suitably clothed or prepared to prevent the spread of contamination in the transporting vehicle. Communication can be maintained from the station to the site ambulance or to the ambulance through the dispatching station. Response team members receive training concerning transportation of contaminated injured individuals. The approximate time to transport a patient to PMC is 60 minutes. The estimated time for local rescue squads to arrive at the station is 30 minutes.

The staff finds the additional information and clarifications provided by the applicant in the response to RAI 25, Question 13.03-65 acceptable because the information and clarifications conform to the guidance in NUREG-0654. Accordingly, the staff considers RAI 25, Question 13.03-65 resolved.

In its Interim Finding Report for Reasonable Assurance, FEMA found that the offsite emergency plans are adequate for this planning standard and the associated evaluation criteria in NUREG-0654.

The staff reviewed the certification letter for the medical service providers described above and the additional information provided in WLS Emergency Plan, Section II.L. The staff finds that the applicant has made arrangements for hospital and medical service providers that have the capability to evaluate radiation exposure and uptake, and persons providing these services are adequately prepared to handle contaminated individuals. In addition, the applicant provided for appropriate emergency first aid treatment at the site, including qualified medical personnel to handle radiation emergencies, and arrangements for transporting victims of radiological accidents (i.e., contaminated injured individuals) to offsite medical support facilities.

Conclusion

The staff concludes that the information provided in the WLS COLA is consistent with the guidelines in NUREG-0654, Planning Standard L. Therefore, the staff finds the information acceptable and meets the relevant requirements of 10 CFR 50.47(b)(12) and 10 CFR Part 50, Appendix E, Section IV.E, insofar as the information describes the essential elements of advanced planning and the provisions made to cope with emergency situations.

13.3.4.13 Recovery and Reentry Planning and Post-accident Operations

As reflected in NUREG-0654, Planning Standard M, "Recovery and Reentry Planning and Post-Accident Operations," 10 CFR 50.47(b)(13), requires that general plans for recovery and reentry be developed. In addition, 10 CFR Part 50, Appendix E, Section IV.H requires a description of criteria to be used to determine when, following an accident, reentry of the facility would be appropriate or when operation could be resumed.

In WLS Emergency Plan, Section II.M, "Recovery and Re-entry," the applicant provides a general framework for the contents of recovery plans and procedures to address a range of recovery and re-entry activities including recovery organization and its concepts of operation. The staff reviewed this section, as well as other relevant portions, to determine whether the application conforms to the applicable guidance and complies with the pertinent regulatory requirements. The staff's primary focus was to evaluate the emergency plan against NUREG-0654, Planning Standard M. NUREG-0654, Planning Standard M provides the detailed evaluation criteria that the staff should consider to determine whether the emergency plan meets the applicable regulatory requirements in 10 CFR 50.47(b)(13).

WLS Emergency Plan, Section II.M states that the applicant implements plans and procedures for recovery and re-entry that provide guidance on: the recovery/re-entry organization; responsibilities for decision-making activities, including decisions for relaxing protective measures based on existing or potential hazardous conditions; the means for informing ERO members about organizational structure changes and the start of recovery operations; and the methods for periodically updating estimates of total population exposure. WLS Emergency Plan, Appendix 5 includes an EPIP titled, "Recovery and Re-entry," which implements WLS Emergency Plan, Section II.M.2.

WLS Emergency Plan, Section II.M.2, "Recovery Organization," states that the applicant establishes a recovery organization consistent with existing conditions and continuing organizational needs that may be modified to address the given situation. The applicant does not expect this to be necessary following a Notification of Unusual Event or an Alert emergency classification level. Primary positions for the Recovery Organization are identified by title and responsibilities, including the Emergency Coordinator who acts as site liaison with the Recovery Organization and the EOF Director who assumes control and direction of the recovery operation with the authority and responsibilities as described in the EPIPs, including the coordination with Federal, State, and local governments. Other key Recovery Organization personnel include the Work Control, Radiological Assessment, Engineering Support, Public Information, and EOF Services managers who support the EOF Director and site recovery and re-entry efforts. The Recovery Organization may perform its activities from one or more designated ERFs or from other locations as specified by the responsible recovery organization managers. The applicant's response to RAI 25, Question 13.03-66(A) provided additional insight into the applicant's intent to model the WLS recovery organization to align with Catawba, Maguire, and Oconee nuclear stations. The applicant's corporate procedure for recovery and re-entry was provided as an example of this structure. Although no revision to the WLS Emergency Plan was proposed in response to this RAI, the staff considers the additional clarification provided by the applicant regarding its Recovery Organization structure acceptable since it conformed to the guidance in NUREG-0654.

WLS Emergency Plan, Section II.M.2 also includes provisions for relaxing protective measures when reentry of the facility would be appropriate or when operation could be resumed: station parameters no longer indicate a potential or actual emergency exists; the release of radioactivity is controllable, does not exceed permissible levels, and does not present a credible danger to the public; and the station is capable of sustaining itself in a long term shutdown condition.

WLS Emergency Plan, Section II.M.3, "Changes in Organizational Structure," states that the recovery process is implemented when the ERO managers, with concurrence from State and Federal agencies, have determined the station is in a stable and controlled condition. Upon this determination, the EOF Director notifies the NRC Operations Center, and the State and local EOCs, to inform them that the emergency condition has been terminated and any required recovery has commenced.

WLS Emergency Plan, Section II.M.4, "Updating Total Population Exposure During Recovery Operations," states that the Radiological Assessment Manager will work with South Carolina and North Carolina officials to periodically update estimates of total population exposure using population distribution data. In a December 23, 2008, response to RAI 25, Question 13.03-66(C), the applicant provided additional information to indicate that the Radiological Assessment Manager will communicate with South Carolina Department of Health and Environmental Control and the North Carolina Department of Environment and Natural Resources/Radiation Protection section via liaison personnel that are assigned within the EOF to periodically update estimates of total population exposure using population distribution data. Although no revision to the WLS Emergency Plan was proposed in response to this RAI, the staff finds the additional clarification provided by the applicant acceptable since it conformed to the guidance in NUREG-0654. Accordingly, the staff considers this question resolved.

In its Interim Finding Report for Reasonable Assurance, FEMA found that the offsite emergency plans are adequate for this planning standard and the associated evaluation criteria in NUREG-0654.

The staff finds that the applicant has developed general plans for recovery and reentry, including describing criteria to be used to determine when, following an accident, reentry of the facility is appropriate or operation can be resumed. In addition, the applicant designated the individuals who will fill key positions in the facility recovery organization. The staff finds that the plans adequately specify the means for informing members of the response organizations that a recovery operation is to be initiated, describe how decisions to relax protective measures are made, and include a method for periodically estimating total population exposure.

Conclusion

The staff concludes that the information provided in the WLS COLA is consistent with the guidelines in NUREG-0654, Planning Standard M. Therefore, the staff finds the information acceptable and meets the relevant requirements of 10 CFR 50.47(b)(13) and 10 CFR Part 50, Appendix E, Section IV.H, insofar as the information describes the essential elements of advanced planning and the provisions made to cope with emergency situations.

13.3.4.14 Exercises and Drills

As stated in NUREG-0654, Planning Standard N, "Exercises and Drills," 10 CFR 50.47(b)(14) requires that periodic exercises be conducted to evaluate major portions of emergency response capabilities, periodic drills be conducted to develop and maintain key skills, and deficiencies identified as a result of exercises or drills be corrected. In addition, 10 CFR Part 50, Appendix E, Section IV.F requires a description of the program that provides for training of employees, exercising by periodic drills, and participation by other assisting persons. The exercises, including hostile action exercises of the onsite and offsite emergency plans, shall test the adequacy of timing and content of implementing procedures and methods, test emergency equipment and communication networks, test the public alert and notification system, and ensure that emergency organization personnel are familiar with their duties. 10 CFR Part 50, Appendix E, Section IV.F describes the full participation exercise, including timing of the exercise, correction of any deficiencies identified during the exercise, the use of remedial exercises, developing exercise scenarios, and an eight-year exercise cycle.

In WLS Emergency Plan, Section II.N, "Exercises and Drills," the applicant described the conduct and frequency of emergency exercises and drills, including coordination between WLS and offsite organizations. The staff reviewed this section, as well as other relevant portions of the application, to determine whether the application conforms to the applicable guidance and complies with the pertinent regulatory requirements. The staff's primary focus was to evaluate the emergency plan against NUREG-0654, Planning Standard N, which provides the detailed evaluation criteria that the staff should consider to determine whether the emergency plan meets the applicable regulatory requirements in 10 CFR 50.47(b)(14).

WLS Emergency Plan, Section II.N.1, "Exercises and Drills," states that exercises are conducted on a biennial basis in a manner that tests the major elements of the plans and emergency response capabilities. Exercises test adequacy of timing and content of implementing procedures; test emergency equipment and communication networks, and public notification systems; evaluate emergency organization personnel's familiarity with their duties; and identify deficiencies.

WLS Emergency Plan, Section II.N.2, "Drills," states that upon request, the applicant allows affected State and local governments located within the plume exposure pathway EPZ to participate in drills. Drills are conducted between biennial exercises to maintain adequate emergency response capabilities, and the drills are controlled and observed by individuals qualified to conduct and evaluate the drill. The drills are used to implement accident management strategies, provide supervised instruction, allow the operating staff to resolve problems and focus on internal training objectives. One or more drills may be included as portions of an exercise. WLS Emergency Plan, Section II.A.1, "Emergency Organization," identifies participating organizations. The section describes communication, fire, medical emergency and radiation protection drills.

As described in WLS Emergency Plan, Section II.N.2.a, "Communication Drills," communication drills are performed monthly to test the notification capabilities with State and local governments within the plume exposure pathway EPZ. The capability to notify NRC Headquarters and the Regional Operations Center is also tested monthly. The capability to notify the NRC region and Federal EROs from the EOF is tested quarterly along with the functionality of computer and communication equipment. All communication systems discussed in WLS Emergency Plan, Section II.F, “Emergency Communications,” are tested annually. The drills include provisions to ensure that all participants are able to understand the content of the messages. Communication between the nuclear facility, State and local EOCs, and field assessment teams will be tested annually.

WLS Emergency Plan, Section II.N.2.b, “Fire Drills,” describes that fire drills are conducted as required by WLS COL FSAR Section 9.5.1, “Other Auxiliary Systems.” WLS COL FSAR Section 9.5.1, “Fire Brigade Training,” states that training is conducted by qualified individuals and consists of classroom instruction supplemented with periodic classroom retraining, practice in firefighting, and fire drills. WLS COL FSAR Table 13.4-201 (Sheet 2 of 7), “Operational Programs Required by NRC Regulations,” identifies the Fire Protection Program implementation milestones to be completed prior to receipt of fuel onsite and prior to initial fuel load.

WLS Emergency Plan, Section II.N.2.c, “Medical Emergency Drills,” states that medical emergency drills that include a simulated contaminated injured individual, transportation to offsite facilities, and participation by the local medical support agencies are performed annually. The WLS Emergency Plan also states that the offsite portions of the medical drill may be performed as part of the required biennial exercise.

WLS Emergency Plan, Section II.N.2.d, “Radiological Monitoring Drills,” states that radiological monitoring drills, involving both onsite and offsite radiological monitoring activities are conducted at least once each calendar year. Radiological monitoring drills include the use of appropriate procedures for collecting and analyzing samples and recording results; collection and analysis of the sample media for which the facility is responsible; communication with monitoring teams and recordkeeping activities. Drills may be coordinated with State and local organizations or conducted separately.

WLS Emergency Plan, Section II.N.2.e, “Radiation Protection Drills,” states that onsite radiation protection drills that include response to, and analysis of, simulated elevated airborne and liquid activity levels and elevated area radiation levels in the environment are conducted at least semi-annually.

WLS Emergency Plan, Section II.N.3 “Conduct of Drills and Exercises,” states that basic performance objectives, the participants, observers, coordination of offsite resources, casualties, simulated events, a timeline, a narrative summary of the events and plant conditions, and evaluation criteria are included in scenario materials. The WLS Emergency Plan states that exercises and drills will be carried out to allow free play for decision-making and to meet the drill objectives.

WLS Emergency Plan, Section II.N.4, “Exercise and Drill Evaluation,” states that one or more qualified instructors or evaluators supervises and evaluates drills and exercises. A qualified individual must have been evaluated by an Emergency Planning Manager. Areas to be observed by the evaluators are defined in a critique sheet.

WLS Emergency Plan, Section II.N.5, “Drill and Exercise Critiques,” states that the applicant records input from the critique participants, evaluates the need for changes to the plan, procedures, equipment, facilities, and other components of the program and develops an action plan to address the identified substantive issues. Identified substantive issues are written up as corrective actions and are tracked to completion following the corrective action program. WLS

Emergency Plan, Section II.O.4, “Emergency Response Training and Qualification,” states that training programs may include practical drills consistent with WLS Emergency Plan, Section II.N, “Exercises and Drills.” Instructors or evaluators immediately correct any erroneous action and if appropriate, demonstrate performance consistent with procedures.

WLS Emergency Plan, Section II.N, “Exercises and Drills,” does not contain a statement related to conducting remedial exercises if the emergency plan is not satisfactorily tested during the biennial exercise. Therefore, in RAI 25, Question 13.03-67, the staff requested that the applicant provide additional information related to remedial exercises. In a December 18, 2008, response, the applicant revised WLS Emergency Plan, Section II.N.1.c, “Remedial Exercises,” to include a discussion on remedial exercises. The staff finds the additional information and clarifications provided by the applicant in response to RAI 25, Question 13.03-67 acceptable because they conform to the guidance in NUREG-0654. Accordingly, the staff considers RAI 25, Question 13.03-67 resolved.

The applicant proposed EP-ITAAC 8.1 to test that the licensee conducts a full participation exercise within the specified time periods of 10 CFR Part 50, Appendix E prior to fuel load to evaluate major portions of emergency response capabilities, including participation by each State and local agency within the plume exposure pathway EPZ, and each State within the ingestion pathway EPZ.

In its Interim Finding Report for Reasonable Assurance, FEMA found that the offsite emergency plans are adequate for this planning standard and the associated evaluation criteria in NUREG-0654. In addition, FEMA stated that the adequacy of the WLS Emergency Plan review is also dependent on satisfactory demonstration of plan implementation during a joint exercise with the licensee and State and local governments, and utilizing WLS facilities. EP-ITAAC Acceptance Criterion 8.1.3.1 establishes the standards that the exercise is completed per 10 CFR Part 50, Appendix E, the offsite objectives are met, and there is no uncorrected offsite deficiencies or a license condition requiring that offsite deficiencies to be corrected prior to any operation above 5 percent rated thermal power.

The staff finds that the applicant has described provisions for conducting periodic exercises and drills to evaluate major portions of emergency response capabilities and to develop and maintain key skills. The exercises will test the adequacy of implementing procedures, emergency equipment and communication networks, and the public notification system, and will ensure that the ERO personnel are familiar with their duties. In addition, the applicant described the full participation exercise, participation by offsite authorities, and how exercise and drill deficiencies will be identified and corrected. The staff notes that EP-ITAAC Acceptance Criterion 8.1.3.1 conforms to EP-ITAAC Acceptance Criterion 14.1.3 from NUREG-0800, Chapter 14.3.10, Table 14.3.10-1. The staff finds the WLS Emergency Plan adequately describes the security based drill and exercise program.

Conclusion

The staff concludes that the information provided in the WLS COLA is consistent with the guidelines in NUREG-0654, Planning Standard N. Therefore, the staff finds the information acceptable and meets the relevant requirements of 10 CFR 50.47(b)(14) and 10 CFR Part 50, Appendix E, Section IV.F, insofar as the information describes the essential elements of advanced planning and the provisions made to cope with emergency situations.

13.3.4.15 Radiological Emergency Training

As stated in NUREG-0654, Planning Standard O, "Radiological Emergency Response Training," 10 CFR 50.47(b)(15) requires that radiological emergency response training be provided to those who may be called on to assist in an emergency. In addition, 10 CFR Part 50, Appendix E, Section IV.F.1 requires a description of the program that provides for training of employees, exercising by periodic drills, and participation by other assisting persons.

In WLS Emergency Plan, Section II.O, "Radiological Emergency Response Training," the applicant described the training that will be conducted for both onsite and offsite response organizations in support of an emergency at the WLS. The staff reviewed this section, as well as other relevant portions of the application, to determine whether the application conforms to the applicable guidance and complies with the pertinent regulatory requirements. The staff's primary focus was to evaluate the emergency plan against NUREG-0654, Planning Standard O, which provides the detailed evaluation criteria that the staff should consider to determine whether the emergency plan meets the applicable regulatory requirements in 10 CFR 50.47(b)(15).

WLS Emergency Plan, Section II.O.1, "General," states that the WLS training program provides for initial training and retraining for individuals who have been assigned emergency response duties. WLS Emergency Plan, Sections II.O.1.a, "Off-site Emergency Response Training," and "Mutual Aid Agreements," describe training of offsite personnel likely to provide assistance during an emergency. The training addresses: scope of the WLS Emergency Plan; emergency classification, notification methods, basic radiation protection, individuals in response organizations who direct onsite activities, definition of support roles, and station access procedures. In RAI 25, Question 13.03-68, the staff requested that the applicant provide additional information on training of media representatives. In a December 17, 2008, response, the applicant stated this information is provided in WLS Emergency Plan, Section II.G.5, "News Media Training," and states that, "Annually, Duke Energy provides to affected media organizations information regarding the emergency plans, information regarding radiation hazards, and points of contact for release of public information during an emergency." Training is performed prior to assignment to a position, which includes practical drills consistent with WLS Emergency Plan, Section II.N, "Exercises and Drills."

WLS Emergency Plan, Section II.O.2, "On-site Emergency Response Training," states that the emergency response training program includes Duke Energy personnel who may be called upon to respond to an emergency. The training program includes classroom training and practical drills in which each individual demonstrates ability to perform his/her assigned emergency function. Training is complete prior to assignment to a position in the ERO. The training program includes practical drills addressed in WLS Emergency Plan, Section II.N, "Exercises and Drills," during which each individual demonstrates the ability to discharge the assigned emergency response function. Any erroneous performance is immediately noted during these practical drills, and proper performance may be demonstrated consistent with procedures and standards by an instructor or evaluator. Training is provided to the following categories of responders:

a. Directors or coordinators of the plant emergency organization

b. Personnel responsible for accident assessment, including CR shift personnel

c. Radiological monitoring teams

d. Fire control teams (fire brigades)

e. Repair and damage control teams

f. First aid and rescue teams

g. Medical support personnel

h. Licensee's headquarters support personnel

i. Security personnel

WLS Emergency Plan, Section II.O.4, "Emergency Response Training and Qualification," states that the applicant implements a program to provide position-specific training for positions covered in WLS Emergency Plan, Sections II.O.4.a through II.O.4.j, including offsite local support personnel. Content of the training program is appropriate for the duties and responsibilities of the assigned position.

WLS Emergency Plan, Section II.O.5, "Retraining," states that failure to successfully complete training in a timely manner as specified in plant training program requirements results in the individual's removal from the ERO pending completion of the required training.

The staff finds the additional information and clarifications provided by the applicant in response to RAI 25, Question 13.03-68 acceptable because they conform to the guidance in NUREG-0654. Accordingly, the staff considers RAI 25, Question 13.03-68 resolved.

In its Interim Finding Report for Reasonable Assurance, FEMA found that the offsite emergency plans are adequate for this planning standard and the associated evaluation criteria in NUREG-0654.

The staff finds that the applicant has provided for radiological emergency response training to those who may be called on to assist in an emergency. In addition, the applicant described the program that trains employees to ensure they are familiar with their specific emergency response duties, including exercising with periodic drills. The applicant also described the participation in training and drills by other persons whose assistance might be needed, including specialized initial training and periodic retraining.

Conclusion

The staff concludes that the information provided in the WLS COLA is consistent with the guidelines in NUREG-0654, Planning Standard O. Therefore, the staff finds the information is acceptable and meets the relevant requirements of 10 CFR 50.47(b)(15) and 10 CFR Part 50, Appendix E, Section IV.F.1, insofar as the information describes the essential elements of advanced planning and the provisions made to cope with emergency situations.

13.3.4.16 Responsibility for the Planning Effort

As reflected in NUREG-0654, Planning Standard P, “Responsibility for the Planning Effort – Development, Periodic Review, and Distribution of Emergency Plans,” 10 CFR 50.47(b)(16), as reflected in the Planning Standard P, requires that responsibilities for plan development and review and for distribution of emergency plans are established and that planners are properly trained. In addition, 10 CFR Part 50, Appendix E, Section IV.G requires a description of provisions to be employed to ensure that the emergency plan, its implementing procedures, and emergency equipment and supplies are maintained up to date.

In WLS Emergency Plan, Section II.P, “Responsibility for the Planning Effort,” the applicant described the responsibilities associated with maintaining the emergency preparedness program, including the development, review, and distribution of the emergency plan. The staff reviewed this section, as well as other relevant portions of the application, to determine whether the application conforms to the applicable guidance and complies with the pertinent regulatory requirements. The staff’s primary focus was to evaluate the emergency plan compared to NUREG-0654, Planning Standard P, “Responsibility for the Planning Effort: Development, Periodic Review and Distribution of Emergency Plans.” Planning Standard P provides the detailed evaluation criteria that the staff should consider to determine whether the emergency plan meets the applicable regulatory requirements in 10 CFR 50.47(b)(16).

WLS Emergency Plan, Section II.P.1, “Training,” describes the process used to provide training for the Emergency Planning Manager and support staff to facilitate effective implementation of the emergency planning effort, consistent with applicable regulatory requirements and guidance, license conditions, other commitments, and accepted good practices. Training may include formal education, professional seminars, plant-specific training, industry meetings, and other activities and forums that provide for an exchange of pertinent information.

WLS Emergency Plan, Section II.P.2, “Responsibility for Radiological Emergency Response Planning,” describes the responsibility of plan development. The WLS Emergency Plan states that the WLS Site Vice President is the overall authority for ensuing that there is an adequate level of emergency preparedness maintained at the site. The responsibility for the planning effort is delegated to the Emergency Planning Manager.

WLS Emergency Plan, Section II.P.3, “Emergency Planning Manager,” describes the Emergency Planning Manager position. The incumbent is responsible for developing and updating the Emergency Plan and coordinating that plan with other response organizations. The Duke Energy corporate staff may augment the WLS onsite emergency planning efforts, as needed.

WLS Emergency Plan, Section II.P.4, “Plan Reviews and Updates,” states that the WLS Emergency Plan is reviewed, updated as needed, and certified to be current on an annual basis. Changes identified by drills and exercises are incorporated into the WLS Emergency Plan. On an annual basis, the Emergency Planning Manager reviews with each affected State and local organization Lee Nuclear Station procedures for emergency classification.

WLS Emergency Plan, Section II.P.5, “Distribution of Revised Plans,” covers the distribution of the revised plans. The Emergency Planning Manager or designee makes needed changes to the WLS Emergency Plan. The pages that are changed are marked and dated to indicate the change. The WLS Site Vice President reviews and approves the changes. The approved revised plans are distributed through the WLS document control organization to organizations and individuals that have responsibilities associated with implementing the Plan.

WLS Emergency Plan, Section II.P.6, “Supporting Plans,” provides a list of supporting plans, including the following:

- South Carolina Operational Radiological Emergency Response Plan, Appendix 2 South Carolina Emergency Operation Plan
- North Carolina Emergency Operations Plan
- North Carolina Radiological Emergency Response Plan for Nuclear Power Facilities
- Cherokee County, SC, Emergency Operations Plan
- York County, SC, Emergency Operations Plan
- NRC Region II Incident Response Plan
- Interagency Radiological Assistance Plan - Region 3 - U.S. Department of Energy
- INPO Emergency Response Plan

WLS Emergency Plan, Section II.P.7, “Implementing Procedures” references the topical listing of EPIPs supporting the plan as being contained in WLS Emergency Plan, Appendix 5, “Implementing Procedures.”

The WLS Emergency Plan contains a Table of Contents. In addition, WLS Emergency Plan, Section II.P.8 “Table of Contents,” states that the format for this Emergency Plan directly follows the format of NUREG-0654. Appendix 8, “Cross-References to Regulations, Guidance, and State and Local Plans,” provides a cross reference for regulatory requirements (includes Appendix E) and NUREG-0654.

WLS Emergency Plan, Section II.P.9, “Emergency Plan Audits,” describes the applicant’s Nuclear Performance Assessment organization’s independent audit of the WLS Emergency Preparedness Program. The organization performs or oversees the performance of periodic independent audits of the emergency preparedness program consistent with 10 CFR 50.54(t). Frequency of the periodic audits is based on an assessment of performance, but all elements of the emergency preparedness program must be reviewed at least once every 24 months. An audit is performed after a change occurs in personnel, procedures, equipment, or facilities that potentially could adversely affect emergency preparedness, but no longer than 12 months after the change. Audit results are documented and improvement recommendations sent to WLS and Duke Energy management. Duke Energy’s Records Management shall file and maintain records of this and a description of any corrective actions for five years.

WLS Emergency Plan, Section II.P.10, “Emergency Telephone Numbers,” states that the Emergency Planning Manager (or designee) is responsible for performing a quarterly review of the telephone numbers in the emergency response procedures and for ensuring required revisions are completed.

In WLS COLA Part 10, the applicant proposed License Condition 4 “Emergency Planning Actions” to address NRC inspection of Letters of Agreement with entities specific to emergency planning responsibilities. The staff refers to this as License Condition (13-6).

Proposed License Condition (13-6):

> Prior to the full-participation exercise to be conducted in accordance with the requirements of Appendix E to 10 CFR Part 50, Duke Energy will have available for NRC inspection Letters of Agreement with the entities listed in Appendix 7 of the Lee Nuclear Station COLA Part 5, Emergency Plan. These Letters of Agreement will detail each entity’s specific emergency planning responsibilities, including response to hostile action affecting the plant site, and certify the entity’s concurrence with their responsibilities.

In addition, as discussed in Section 13.3.4.3 of this report, the general nature of the existing letters of agreement is such that the scope of expected support could include expected assistance associated with hostile action at the site, which is required by 10 CFR Part 50, Appendix E, Section IV.A.7 to be identified and described in the WLS Emergency Plan. However, this requirement was not effective until June 23, 2014, which occurred after WLS COLA submission. The WLS Units 1 and 2 letters of agreement supporting the WLS COLA did not specifically address hostile actions, and were not required to when the WLS COLA was initially submitted on December 12, 2007. To clarify that the expected assistance from offsite agencies includes hostile action, the staff has included in the License Condition (below) the requirement for the updated letters of agreement to reflect expected assistance associated with hostile actions at WLS, as required by 10 CFR Part 50, Appendix E, Section IV.A.7.

For the reasons discussed above, the staff finds that delaying the updating of the letters of agreement, and revising the WLS Emergency Plan to include the letters after execution until prior to WLS fuel load is consistent with the requirements in 10 CFR Part 50, Appendix E, Section IV.G, and guidance in NUREG-0654 Evaluation Criterion II.P.4. The staff reviewed the License Condition, and with the exception of the timeframe for submission of the updated letters of agreement, finds that it acceptable for the reasons as discussed above. The staff includes a similar submission timeframe for the updated letters of agreement based on the date scheduled for initial fuel load set forth in the notification submitted in accordance with 10 CFR 52.103(a). Therefore, consistent with the applicant’s proposed License Condition the staff identified License Condition (13-6).

License Condition (13-6):

> No later than 180 days before the date schedule for initial fuel load set forth in the notification submitted in accordance with 10 CFR 52.103(a), Duke Energy shall submit to the Director of the Office of New Reactors (NRO), or the Director’s designee, in writing, updated WLS Units 1 and 2, Letters of Agreement with the following entities, or their successors. These updated Letters of Agreement shall identify the specific nature of arrangements in support of emergency preparedness for WLS, and reflect expected assistance associated with hostile action at the WLS, as defined in 10 CFR Part 50, Appendix E, Section IV.A.7. The WLS Emergency Plan shall have been revised to include these updated Letters of Agreement after they have been executed.
>
> 1. South Carolina Emergency Management Division

2. Piedmont Medical Center

3. Upstate Medical Center, Emergency Medical Services

4. Draytonville-McKown Mountain-Wilkinsville Volunteer Fire Department

5. Cherokee County Emergency Management

6. Cleveland County Emergency Management and Fire Marshall's Office

7. North Carolina Emergency Management

8. South Carolina Department of Health and Environmental Control

9. York County Emergency Management

In its Interim Finding Report for Reasonable Assurance, FEMA found that the offsite emergency plans are adequate for this planning standard and the associated evaluation criteria in NUREG-0654.

Subject to License Condition (13-6), the staff finds that the applicant has established the responsibilities for plan development and review, including distribution of the emergency plans to all appropriate organizations. In addition, the applicant established provisions to properly train the planners (i.e., the individuals responsible for the emergency planning effort) and described the provisions to be employed to ensure that the emergency plan, its implementing procedures, and emergency equipment and supplies are maintained up-to-date.

Conclusion

Subject to License Condition (13-6), the staff concludes that the information provided in the WLS COLA is consistent with the guidelines in NUREG-0654, Planning Standard P. Therefore, the staff finds the information acceptable and meets the relevant requirements of 10 CFR 50.47(b)(16) and 10 CFR Part 50, Appendix E, Section IV.G, insofar as the information describes the essential elements of advanced planning and the provisions made to cope with emergency situations.

13.3.4.17 Evacuation Time Estimate (ETE) Analysis

The WLS Emergency Plan includes an analysis of the time required to evacuate the plume exposure pathway EPZ 16-km (10-mile EPZ). In a March 4, 2010, letter, the applicant filed the WLS Units 1 and 2 Emergency Plan, Revision 2. This submission included the report, "William S. Lee Nuclear Station Development of Evacuations Time Estimates," Revision 2, February 2010, referred to from this point forward as the ETE Report. The ETE Report is a non-public document and the results of the ETE Report, Revision 2, are summarized in WLS Emergency Plan, Appendix 4 ""Evacuation Times Estimate. The ETE Report provides the basis for the following discussion and analyses. At the direction of the staff, technical experts from Sandia National Laboratories reviewed the ETE Report and prepared a technical evaluation report containing the results of their review. The staff and its contractors reviewed the applicant's ETE Report analysis for content and conformity to NUREG-0654, Appendix 4 and to guidance provided in NUREG/CR-7002 and NUREG/CR-6863. The staff considered the contractor's assessment in findings of acceptability and applicability in determining its conclusions of adequacy and compliance with the regulatory guidance.

Introductory Materials Related to the ETE Report

The proposed WLS is located west of the Broad River about 11 km (7 mi) southeast of Gaffney, South Carolina. A description, including a map in the ETE Report (Figure 1-1, "Lee Nuclear Station Site Location"), of the 16-km (10-mi) EPZ and surrounding area was provided. Additional information concerning small communities and topographical features in the WLS vicinity was requested in RAI 25, Questions 13.03-4 and 13.03-5, respectively. In a November 7, 2008, response, the applicant provided an updated Figure 1.1, which identified small unincorporated areas The WLS EPZ includes 14 emergency response planning areas (ERPAs), as illustrated in ETE Report Figure 6-1, "Lee Nuclear Station EPZ ERPA." The ERPAs are described in ETE Report, Appendix L, "ERPA Boundaries," and are typically bounded by State highways, rivers, creeks and town boundaries.

The ETE study includes many assumptions, most of which are identified in Section 2.2, "Study Methodological Assumptions," and Section 2.3, "Study Assumptions." Methodological Assumption Number 3 explains evacuation movements are assumed to be outbound relative to the plant site. Methodological Assumption Number 5 describes assumptions related to voluntary evacuations. Section 2.3 also describes assumptions for the planning basis, school evacuations, mobilization of the general population, percent of households with commuters, and staffing of traffic access and control locations. Study Assumption Number 5 states that vehicles entering the 16-km (10-mi) EPZ along Interstate 85 will not be diverted. A fundamental assumption for the ETE is that advisory to evacuate is coincident with siren notification, which is consistent with guidance in NUREG/CR-7002, "Criteria for Development of Evacuation Time Estimate Studies" (referred to as NUREG/CR-7002)[10]. The assumptions related to commuters are based on the results of a site-specific telephone survey, which is included in the ETE study as Appendix F, "Telephone Survey." Additional assumptions related to factors that influence roadway capacities are described in Section 4, "Estimation of Highway Capacity."

ETE Report Section 3, "Demand Estimation," describes the assumptions for each population segment. For instance, it is assumed 100 percent of the public will evacuate and that no residents are on vacation. Population estimates at special facilities are based on available data from county emergency management offices. Population mobilization times are based on a statistical analysis of data acquired from a telephone survey, as is the relationship between resident population and evacuating vehicles (vehicle occupancy factors). Those without access to private vehicles will be transported to reception centers by bus with an assumption that 50 percent share rides with family, neighbors, and friends. This assumption is consistent with current guidance in NUREG/CR-7002.

The method for analyzing evacuation times included gathering demographic information, performing a field survey of the 16-km (10-mi) EPZ, estimating trip generation times, defining evacuation regions, applying the procedures specified in the 2000 Highway Capacity Manual

[10] 10 CFR 52.17(a)(xii) allows the use of existing guidance in effect 6 months or more from the date of the application submittal to be the basis of the evaluation. The staff used NUREG/CR-7002, which was not issued prior to the application submittal or included in the acceptance criteria in the SRP, to determine if the current guidance would be met for added prudence.

(HCM), and modeling the site using the IDYNEV system. The IDYNEV system includes the PC-DYNEV macroscopic simulation model and the Traffic Assignment Model (TRAD). In the ETE Report, the TRAD model is described in Appendix B, “Traffic Assignment Model,” which provides a description of the model and application of links and nodes. The traffic simulation model is described in the ETE Report, Appendix C, “Traffic Simulation Model: PC-DYNEV,” which describes the model, discusses inputs, outputs, and measures of effectiveness. In the ETE Report, Appendix D, “Detailed Description of Study Procedure,” describes the step-by-step process for integrating the data and models to produce ETEs. In RAI 25, Question 13.03-39, the staff requested that the applicant provide additional information concerning the use of data from field surveys with the default capacity rates of the HCM. The applicant’s December 9, 2008, response clarified that highway characteristics (posted speed, number of lanes, shoulder conditions, free flow speed, terrain, traffic control devices, etc.) observed during the survey were documented in Geographic Information System shapefiles, which were used in the development of the link node analysis network. The capacity of each link was estimated using the procedures outlined in the HCM and the data from the shapefiles. Shadow evacuation is assumed to occur at population percentages that diminish with distance out to 24 km (15 mi). In RAI 25, Question 13.03-7(A) the staff requested that the applicant provide clarification of the assumptions regarding shadow evacuation for partial evacuations and in RAI 25, Question 13.03-7(B) for full 16-km (10-mi) EPZ evacuations. In a November 11, 2008, response, the applicant clarified that 35 percent of individuals in areas within the 16-km (10-mi) EPZ, but not advised to evacuate, may do so voluntarily, and that 30 percent of the population in the shadow region beyond the 16-km (10-mi) EPZ that extends a distance of 24 km (15 mi) from WLS will also elect to evacuate. In RAI 25, Question 13.03-14, the staff requested that the applicant provide a definition of the basis of future projections of numbers of vehicles involved in shadow evacuation. In a November 20, 2008, response, the applicant provided revised information for the year 2011 and proposed textual revisions and updated tables. In RAI 25, Question 13.03-42, the staff requested that the applicant provide additional information related to shadow evacuation. The applicant’s responses to RAI 25, Questions 13.03-42(A) and 13.03-42(C), describes how trip generation times for the shadow evacuation were developed. The applicant’s response to RAI 25, Question 13.03-42(B) describes the population values for the percent shadow evacuation used in the sensitivity analyses of the ETE Report, Appendix I. The response to RAI 25, Question 13.03-42(D) provides the basis for the population used to calculate the shadow evacuation vehicles. In the response, the applicant assumed that the demographics in the shadow region are similar to those in the 16-km (10-mi) EPZ. Results of a sensitivity analysis that varied the shadow population from 15 percent to 60 percent show that the ETE is not sensitive to the number of shadow evacuees. The applicant’s response to RAI 25, Question 13.03-44, clarified that “volunteer” evacuation and “shadow” evacuation have the same meaning.

The ETE Report includes a map showing the proposed site and plume exposure pathway EPZ, as well as transportation networks, topographical features, and political boundaries. The boundaries of the 16-km (10-mi) EPZ, in addition to the evacuation subareas within the 16-km (10-mi) EPZ, are based on factors such as current and projected demography, topography, land characteristics, access routes, and jurisdictional boundaries. A general description of the evacuation model was provided, including the assumptions used in the evacuation time estimate analysis.

The staff finds the clarifications, additional information and textual revisions provided by the applicant in response to RAI 25 Questions 13.03-4, 13.03-5, 13.03-7(A) and 13.03-7(B), 13.03-14, 13.03-39, 13.03-42, and 13.03-44 acceptable because they conform to the guidance in NUREG-0654. The ETE Report describes the method of analyzing the evacuation times. A general description of the evacuation model was provided including the assumptions used in the evacuation time estimate analysis.

Demand Estimation

The ETE study includes an analysis of permanent residents and transient populations, transit-dependent permanent residents (including ambulatory and non-ambulatory), special facility residents, and schools. As described in the ETE Report, Section 2.1, "Data Estimates," the general population is based on the 2000 census data and is projected to 2007.

The population values include those with and without vehicles. Site-specific telephone survey results, provided in the ETE Report, Appendix F, "Telephone Survey," were used to establish demographic characteristics and automobile occupancy information. The ETE Report, Section 3, "Demand Estimation," provides an estimate of the number of people needing to be evacuated (the "demand estimation"), and explains the values for the average household size of 2.62 persons per household and 1.44 vehicles per household were adapted from the survey results. The ETE Report, Table 3-2, "Permanent Resident Population and Vehicles by ERPA," presents the number of residents by evacuation area showing a total 2007 population of 48,249 requiring 26,520 vehicles for a ratio of 1.82 permanent residents per vehicle. The ETE Report, Figure 3-2, "Permanent Residents by Sector," illustrates the residents distributed within radial sectors of the 16-km (10-mi) EPZ. Consistent with guidance in NUREG/CR-6863, "Development of Evacuation Time Estimate Studies for Nuclear Power Plants," demand estimates were adjusted for evening and daytime scenarios and are presented in the ETE Report, Table 6-3, "Percent of Population Groups for Various Scenarios." The ETE Report states that census data showed that the local population increased 7.4 percent between 2000 and 2007. In Revision 2 of the ETE Report, estimates of employees who commute into the 10-mile EPZ to work are based upon the state "Journey to Work Database for 2000," and projected to 2007 using the U.S. Department of Labor job growth rates.

Population estimates are provided for permanent residents, transients, and employees of local businesses. Employment was decreasing by 1.6 percent per year in Cherokee County as shown in the ETE Report, Table 3-3, "Cherokee County Employment," and this was used to extrapolate the 2000 employment numbers to 2007. According to the county website, employment continues to decrease. A vehicle occupancy factor of 1.03 employees per vehicle, obtained from the telephone survey, was used to estimate the number of evacuating vehicles for employees.

In RAI 25, Question 13.03-1, the staff requested that the applicant provide additional information regarding the assumptions related to the population estimates during plant construction, and their consistency with projections in other COL documents. In a November 7, 2008, response, the applicant clarified the methods used to project populations. In RAI 25, Question 13.03-22, the staff requested that the applicant provide clarification regarding the plant construction population itself. The applicant's November 20, 2008, response stated that while the number of employees peak at 1,000, only about 750 are assumed to be onsite at any one time.

In Revision 2 of the ETE Report, the special event evaluated in the analysis was new plant construction. ETE Report Table 6-4, "Vehicle Estimates By Scenario," identifies the peak construction year as 2016 and explains that the permanent resident and shadow populations were escalated to 2016 to determine the ETE. During the peak construction period the workforce estimate is 4,398 construction workers. Based on shift work and operations work, the peak onsite worker estimate is 3,079. In RAI 123, Question 13.03-98, the staff requested that the applicant provide information regarding the peak construction year and the associated demand estimation. In a November 5, 2014, response, the applicant explained that peak construction is currently projected to occur in 2020, and the projected number of evacuating vehicles at that time is 51,255. A sensitivity study that considered 54,381 evacuating vehicles is provided in the ETE Report and shows there is no change to the ETE values with this increase in vehicles. This occurs because of a combination of excess roadway capacity within the 16-km (10-mi) EPZ, and because the ETE is influenced largely by the mobilization times for this site.

A separate analysis for people without personal vehicles is provided in ETE Report Section 8, "Transit Dependent and Special Facility Evacuation Time Estimates." Other transient groups include visitors to local recreational areas, boat launches, and parks. In RAI 25, Question 13.03-23, the staff requested that the applicant provide additional information regarding how transients and employees are factored into the need for additional transit services. The applicants response explained that because there is no mass transit to bring employees and transients into the area, the report assumes that employees and transients will evacuate via the same transportation method used to enter the evacuation area and do not require transit resources for evacuation. In RAI 25, 13.03-29, the staff requested the applicant provide clarification of information regarding special facility transit demand. In a March 4, 2010, response, the applicant explained that contributions for one of the Medical Centers were not included in the total and a revised ETE Report Table 8-4 was provided.

ETE Report, Appendix E, "Special Facility Data," includes data for schools, daycare centers, medical facilities and nursing homes, correctional facilities, hotels, motels and major retail areas. The medical facility patients are identified as ambulatory, wheelchair, and bedridden to quantify specialized vehicles needed to support the evacuation. Revision 2 of the ETE Report lists the EPZ school populations for Cherokee County, York County, and Cleveland County with a combined enrollment of over 10,800 students and 700 staff. Daycare centers in Cherokee County and York County have an enrollment of over 750 children and about 120 employees. In Revision 2 of the ETE Report, medical facilities and nursing homes with inpatient services are listed in Table 8-4, "Special Facilities Transit Demand," showing a capacity of over 500 residents. In RAI 25, Question 13.03-52, the staff requested that the applicant provide additional information regarding the definition of staff for these facilities. In a December 17, 2008, response, the applicant clarified that "staff" includes faculty, but does not include administrative, custodial, food service, and adult volunteers. In RAI 25, Questions 13.03-28(A) and 13.03-28(B), the staff requested that the applicant provide additional information regarding the adequacy of numbers of school buses for a single wave of evacuation. In a March 4, 2010, response, the applicant stated that there were an insufficient number of school buses in Cherokee County for a single wave of relocation. Evacuation of schools and transit-dependent individuals with the existing inventory of school buses available in Cherokee County would require additional trips resulting in an associated increase in the ETE. The applicant provided additional information stating that in accordance with a statewide mutual aid agreement, that in the event of an emergency at the WLS, additional bus transportation resources could be requested from other, non-affected counties or school districts. Also, additional analysis on the impact of solely using Cherokee County school busses for a second and third wave evacuation was provided. In RAI 83, Question 13.03-76, the staffed requested that the applicant provide an analysis to quantify the effect of these multiple trips on the ETE and to identify additional resources or implementation of alternate methods (e.g., mutual aid agreements) that will allow for single wave evacuation. In a December 11, 2009, response, the applicant stated that in accordance with a statewide mutual aid agreement, that in the event of an emergency at WLS, additional bus transportation resources could be requested from other, non-affected counties or school districts. Also, additional analysis on the impact of solely using Cherokee County school busses for a second and third wave evacuation was provided. The applicant updated the WLS Emergency Plan and the ETE Report and incorporated the necessary changes to resolve this question. The staff finds the applicant's responses to RAI 25, Question 13.3-28 and RAI 83, Question 13.03-76 acceptable. Accordingly, the staff considers these questions resolved.

In RAI 25, Questions 13.03-25(A) and 13.03-25(C), the staff requested that the applicant provide additional information regarding evacuation of daycare centers. . In a December 17, 2008, response, the applicant explained that to evacuate children not picked up by parents during the 90-minute mobilization period, daycare centers can transport children to the nearest public elementary school where they can evacuate with the school children. Daycare centers requiring transportation support may also contact the county emergency management agency who would dispatch buses to the daycare center when they become available following evacuation of the school children.

The ETE Report states that the 16-km (10-mi) EPZ has a number of areas that attract transient populations including Prime Outlets at Gaffney, Kings Mountain National Military Park, Kings Mountain State Park and hotels and motels. An estimate of the transient population is provided in ETE Report Section 3, "Demand Estimation," with the cumulative total population presented in ETE Report Figure 3-4, "Transient Population by Sector." The total transient population is listed as 6,678 requiring 2,790 vehicles. The method used to estimate the transient population within the 16-km (10-mi) EPZ included obtaining hotel information from the Cherokee County Office of Emergency Management (OEM), estimating parking spots using aerial imagery for the outlets stores, and obtaining data for the National Military Park and State Park from York County OEM. The staff notes this approach is consistent with the guidance in NUREG/CR-6863.
In RAI 25, Question 13.03-20, the staff requested that the applicant provide additional information concerning transient populations in small parks, recreation areas, and campgrounds not listed in the ETE Report. In a December 20, 2008, response, the applicant indicated that these small facilities are assumed to be used by local residents who are already accounted for in the general population estimates. There is one correctional facility within the 16-km (10-mi) EPZ. In RAI 25, Questions 13.03-25(A) and 13.03-25(B), the staff requested that the applicant provide additional information regarding evacuation of correctional center inmates. In a December 17, 2009, response, the applicant stated that the Cherokee County Detention Center maintains emergency plans that cover facility evacuation and any such evacuation would require only four buses, which would not impact the ETE.

The ETE study states that based on data provided by Limestone College, there are 740 students, of which 370 commute, and it is estimated that 20 percent of the students commute from beyond the 16-km (10-mi) EPZ. In RAI 25, Question 13.03-25(D), the staff requested that the applicant provide additional information about evacuation of Limestone College students. In a December 17, 2008, response, the applicant explained that half of the students are campus residents included in the general population and 80 percent of the remainder are local residents; therefore, only about 74 students are in the same category as "employee commuters." This accounting resulted in a minor change to the ETE Report.

Therefore, in RAI 123, Question 13.03-97, the staff requested that the applicant explain why college students are included as permanent population if they are evacuating in their own vehicles. In a November 6, 2014, response, the applicant explained that the on-campus student population is 1,030 and there is no need to include the full enrollment because this would consider online students and extended campus students who would not be part of an evacuation. The staff finds the applicant's response to RAI 123, Question 13.03-97 acceptable. Accordingly, the staff considers this question resolved.

Evacuation routes are described and times are estimated for transit-dependent and special facilities in ETE Report Section 8, "Transit dependent and Special Facility Evacuation Time Estimates." Additional information provided by the applicant in a November 25, 2009, response to RAI 25, Question 13.03-27, which clarifies that medical facility residents are all assumed to be evacuated by bus. Revision 2 of the ETE Report states that based on experience at other plants, the estimated average mobilization time for buses is 90 minutes. The method for developing the ETEs considered evacuation of the inner areas and outer areas in each direction around the 16-km (10-mi) EPZ. The 3.2-km (2-mi) zone is assumed to be evacuated simultaneously with the downwind sectors and ERPAs following the keyhole configuration. Vehicles traveling through the 16-km (10-mi) EPZ (external trips) at the time of an accident are assumed to continue to enter the 16-km (10-mi) EPZ during the first 90 minutes. Thereafter, none are assumed to enter and those remaining also evacuate with the residents and other transients. In RAI 25, Question 13.03-41, the staff requested that the applicant provide clarification of the number of vehicles passing through the 16-km (10-mi) EPZ. In response, the applicant provided additional information about individual highway segments. In a December 9, 2008, response to RAI 25, Question 13.03-41(B), the applicant clarified that Floyd Baker Boulevard (State Route 11) is also called Chesnee Highway. In the response to RAI 25, Question 13.03-41(C), the applicant provided a revision to ETE Report Table 6-4 (which is also discussed in the response to RAI 25, Question 13.03-14).

In RAI 25, Questions 13.03-6, 13.03-26, and 13.03-30, the staff requested that the applicant provide additional information regarding possible additional special populations. In a March 4, 2010, response to RAI 25, Question 13.03-6, the applicant explained that up to seven additional medical facilities could be operating within the 16-km (10-mi(EPZ and that contact would be made with these facilities to verify that they are operational and to obtain population data. These contacts are described in the responses to RAI 25, Questions 13.03-26 and 13.03-30. In a March 4, 2010, response to RAI 25, Questions 13.03-26 and 13.03-30, the applicant explained that minor revisions will be made in the ETE Report to include these facilities but that each has its own transportation, resulting in no impact to the ETE. In RAI 25, Question 13.03-32, the staff requested that the applicant provide maps showing locations of these facilities. The applicant's response provided additional maps for the ETE Report.

In RAI 25, Question 13.03-15, the staff requested that the applicant provide additional information regarding special needs populations. In a March 4, 2010, response, the applicant provided additional information regarding estimates of homebound disabled individuals who are transit-dependent and proposed new text for the ETE Report. In RAI 25, Question 13.03-21(A), the staff requested that the applicant provide additional information regarding transient populations at special events routinely held in the region. In a November 20, 2008, response, the applicant provided a revision to Section 3 of the ETE Report describing the construction peak scenario and a sensitivity study related to inclusion of the Ed Brown Rodeo in the ETE estimates. In RAI 25, Question 13.03-21(B), the staff requested that the applicant provide additional information regarding peak tourist populations. In a November 20, 2008, response, the applicant clarified that peak tourist populations are included in the recreational areas, shopping, and lodging estimates.

The potential for double counting is discussed in ETE Report Section 3 and subsequent sections quantify how double counting is considered. This includes an approach that avoids double counting by treating non-EPZ residents as commuters and 16-km (10-mi) EPZ residents who are employed within the 16-km (10-mi) EPZ as residents. The mapping clarity is consistent with the guidance in NUREG/CR-6863, "Development of Evacuation Time Estimate Studies for Nuclear Power Plants," with resolution that supports a detailed review of the roadway network and geographic features.

In RAI 25, Question 13.03-13, the staff requested that the applicant provide clarification of the "3 Miles to EPZ" notation in ETE Report Figure 3-3, "Permanent Resident Vehicles by Sector" and ETE Report Figure 3-4, "Transient Population by Sector." In a November 11, 2008, response, the applicant defined the notation as a distance 5 km (3 mi) from the plant to the 16-km (10-mi) EPZ boundary. In RAI 25, Question 13.03-11, the staff requested that the applicant provide additional information regarding number of vehicles estimated per scenario. In a November 11, 2008, response, the applicant clarified the relationship of the scenarios to the vehicle estimates in these figures.

The ETE Report provides an estimate of the number of people who may need to evacuate. Three population segments are considered including permanent residents, transients, and persons in special facilities. The permanent population is adjusted for growth, and the population data is translated into two groups including those with automobiles and those without automobiles. The number of vehicles used by permanent residents is estimated using an appropriate automobile occupancy factor. Evacuation time estimates for simultaneous evacuation of the entire plume exposure pathway 10-mile EPZ were provided.

Estimates of transient populations are developed using local data including peak tourist volumes and employment data. Estimates for special facility populations are provided and schools are included in this segment.

The zones for which evacuation time estimates were determined encompass the entire area within the plume exposure pathway 10-mile EPZ. The maps are generally adequate for the purpose, and the level of detail is approximately the same as USGS quadrant maps. The assumptions on evacuation are based on simultaneous evacuation of inner and outer sectors.

The staff finds the clarifications, additional information, and textual revision submitted by the applicant in response to RAI 25, Questions 13.03-1, 13.03-6, 13.03-11, 130.3-13, 13.03-15, 13.03-20, 13.03-21(A and B), 13.03-22, 13.03-23, 13.03-25(A, B, C, and D), 13.03-26, 13.03-27, 13.03-28(A and B), 13.03-29, 13.03-30, 13.03-32, 13.03-41(A and B), and 13.03-52, acceptable because they conform to the guidance in NUREG-0654. The staff confirmed Revision 2 of the WLS Emergency Plan incorporated the information and textual changes provided in the response to RAI 25, Questions 13.03-6, 13.03-15, 13.03-21(A), 13.03-25(D), 13.03-26, 13.03-28(A and B), 13.03-29, and 13.03-30. The staff finds the ETE Report adequately addresses the estimate of the number of people who may need to be evacuated. This is acceptable because it conforms to the guidance in NUREG-0654, Appendix A, Section II.

Traffic Capacity

As described in ETE Report Section 1.1, "Overview of the ETE Process," a detailed field survey of the highway system and traffic conditions within the 16-km (10-mi) EPZ and shadow evacuation region was conducted. Major evacuation routes are shown in figures within ETE Report Section 10, "Evacuation Routes." The evacuation network used in the analysis is illustrated on mapping provided in ETE Report Appendix K, "Evacuation Roadway Network Characteristics," and the types and capacities of each roadway segment are listed by unique link numbers in ETE Report Table K-1, "Evacuation Roadway Network Characteristics." These unique link numbers identify the upstream and downstream node numbers, length, number of lanes, saturation flow rate, and free flow speed. The high quality of mapping is such that all evacuation routes are illustrated together with residential streets. Separate segments were established for areas where the roadway segment characteristics change or where the roadway is narrowed. In RAI 25, Question 13.03-39, the staff requested that the applicant provide additional information, describing the road network used for the evacuation routes, specifically, information regarding highway lane widths. In a December 9, 2008, response to RAI 25, Question 13.03-39, the applicant clarified the assumptions on lane widths.

The ETE Report included assumptions for determining the number of vehicles needed, as well as the methodology, for determining the transport dependent population. The applicant also analyzed travel times and potential locations for serious congestion along the evacuation routes. Since the maps provided in the ETE Report illustrated areas for which congestion was predicted, but not the duration of that congestion, additional information regarding duration of congestion was requested by the staff in RAI 25, Question 13.03-48. In a March 4, 2010, response, the applicant included updated figures with congested links identified that can be cross referenced to a new table containing related information. In Revision 2 of the ETE Report, traffic queuing and congestion areas are presented in Figure 7-3, "Areas of Traffic Congestion 1 Hour after the Advisory to Evacuate," through Figure 7-5, "Areas of Traffic Congestion 3 Hours after the Advisory to Evacuate." A Level of Service F, which indicates heavy congestion, is observed in and around Gaffney, South Carolina from 30 minutes until more than 2 hours after the advisory to evacuate. After about 3 hours, heavy congestion within the 16-km (10-mi) EPZ is no longer present.

In RAI 25, Question 13.03-12, the staff requested that the applicant provide clarification on several traffic capacity questions. In a March 4, 2010, response to RAI 25, Question 13.03-12(A), the applicant provided clarification that the ETE developed for school in session considers that the same buses will be used to evacuate transit-dependent individuals. In a March 4, 2010, response to RAI 25, Question 13.03-12(B) concerning the effect that this "second wave" had on the transit-dependent individual ETE, the applicant stated that after dropping off school children at the reception centers, the buses return to the 16-km (10-mi) EPZ to perform a "second wave" evacuation of transit-dependent persons. A minor adjustment to the ETE was also made to account for the response to RAI 25, Question 13.03-31, which adjusted assumed bus speeds. The applicant's response to RAI 25, Question 13.03-12(C) indicates that the bus routes pass schools and that some transit-dependent individuals will walk to the bus route and be picked up as the buses traverse these routes; others will walk to a school to await the arrival of a bus. In its response to RAI 25, Question 13.03-12(D) concerning the use of school buses on weekends and in summer, the applicant stated that some buses were assumed to be in use during these periods, but that the ETE calculations were not altered by this assumption. In RAI 25, Question 13.03-12E, the staff also requested that the applicant provide an explanation of inbound travel speed and time. In a March 4, 2010, response, the applicant stated that transit bus speeds would be 72 kph (45 mph) in good weather and 64 kph (40 mph) in adverse weather, conditional on the assumption that traffic control points would not hinder the movement of inbound buses. In a March 4, 2010, response to RAI 25, Question 13.03-12(F), the applicant provided the basis for the estimate for pickup time that each bus will, on average, contain 30 passengers, each picked up individually, with a delay associated with each stop of 1 minute.

The ETE Report Section 4, “Estimation of Highway Capacity,” describes the methods used to estimate highway capacity. Revision 2 of the ETE Report explains the methods used are generally taken from HCM 2000 published by the Transportation Research Board of the National Research Council. In RAI 13.03-40, the staff requested that the applicant provide additional information regarding use of empirical modifiers to the HCM queue discharge flow (QDF) rates. In a November 24, 2008, response, the applicant defended a conservative view in estimating the capacity at bottlenecks when congestion develops, so a QDF factor of 0.85, when flow breaks down as determined by the simulation model, was applied.

The modeling described in ETE Report Section 4 relies upon the simulation model PC DYNEV. In RAI 25, Question 13.03-10(A), the staff requested that the applicant provide clarification regarding the modeling of traffic through intersections. In a November 11, 2008, response, the applicant stated that application of traffic control points was not considered and the modeling used the equations presented in ETE Report Section 4. The staff notes this information was repeated in the December 9, 2008, response to the similar question in RAI 25, Question 13.03-43. In RAI 25, Question 13.03-10(B), the staff requested that the applicant clarify the use of field observations to determine allocation of characteristics to the modeled highway segments. In a November 11, 2008, response, the applicant supplied additional information regarding the use of data characteristics including number and estimated width of lanes, shoulder type and estimated width, intersection configuration, lane channelization, roadway geometrics; posted speed; actual free speed; abutting land use; traffic control devices; street parking; and signage. In RAI 25, Question 13.03-10(C), the staff requested that the applicant provide clarification of the use of several factors from the HCM in estimating flow rates of vehicles turning through intersections. In a November 11, 2008, response, the applicant provided additional information regarding the definition of these parameters.

ETE Report Section 9, “Traffic Management Strategy,” presents a traffic control and management strategy that is designed to expedite the movement of evacuating traffic. The traffic management strategy is based on a field survey of critical locations and consultation with emergency management and enforcement personnel. In RAI 25, Question 13.03-3, the staff requested that the applicant provide clarification on whether local officials concurred with the selection of traffic control points. In a November 7, 2008, response to RAI 25, Question 13.03-3, the applicant explained the iterative process through which the traffic management plan had been arrived at with county and State authorities. In RAI 25, Questions 13.03-34 and 13.03-35, the staff requested that the applicant provide clarification of how the traffic management strategy was integrated into the ETE calculations. The applicant’s responses states that the calculations do not rely upon any of the traffic control measures described and that their use would improve the ETE.

ETE Report Section 10,” Evacuation Routes,” discusses the emergency evacuation routes. In RAI 25, Question 13.03-38, the staff also requested details regarding the link node map presented in ETE Report Appendix K, “Evacuation Roadway Network Characteristics.” In a March 4, 2010, response to RAI 25, Question 13.03-38, the applicant included a disc that contained a revised copy of ETE Report Figure 1-2, “Lee Link Node Analysis Network.” In RAI 25, Question 13.03-36, the staff requested that the applicant provide clarification between the link node map and the evacuation routes of ETE Report Figures 10-2 through 10-5. In a March 4, 2010, response, the applicant clarified that there is no implication that evacuees are restricted to the major evacuation routes shown. The evacuation network includes many other minor roads that are capable of servicing evacuating traffic. In RAI 25, Question 13.03-37, the staff requested that the applicant provide Information regarding funneling of traffic into the reception centers outside of the 16-km (10-mi) EPZ. In a March 4, 2010, response, the applicant indicated that the Reception Centers will be located several miles beyond the 16-km (10-mi) EPZ boundary and that congestion in the vicinity of the relocation/reception centers is unlikely to impact the ETE.

The ETE Report provides a complete review of the evacuation road network. Analyses are made of travel times and potential locations for congestion. All evacuation route segments and their characteristics, including capacity, are described.

A traffic control and management strategy that is designed to expedite the movement of evacuating traffic is described. The traffic management strategy is based on a field survey of critical locations and consultation with emergency management and enforcement personnel. The applicant also analyzed travel times and potential locations for serious congestion along the evacuation routes.

The staff finds the clarifications, additional information and textual revision submitted by the applicant in response to RAI 25 Questions 13.03-3, 13.03-10(A, B, and C), 13.03-12(A, B, C, D, E, and F), 13.03-34, 13.03-35, 13.03-36, 13.03-37, 13.03-38, 13.03-39, 13.03-40, 13.03-43, and 13.03-48 are acceptable because they conform to the guidance in NUREG-0654. The staff confirmed that Revision 2 of the WLS Emergency Plan incorporated the information and textual changes provided in the response to RAI 25, Question 13.03-38. The staff finds the ETE Report adequately describes the highway capacity estimates. The staff considers this acceptable because it conforms to the guidance in NUREG-0654, Appendix 4, Section III.

Analysis of Evacuation Times

The ETE study includes an analysis of permanent residents and transient populations, transit-dependent permanent residents (including ambulatory and non-ambulatory), special facility residents, and schools. There are 12 scenarios described in ETE Report Table 6-2, “Evacuation Scenario Definitions.” These scenarios cover different times of day, days of the week, weather conditions, and a special event. The ETEs for the permanent residents and transients were developed for the 12 evacuation scenarios for each of the 22 evacuation regions presenting a total of 264 unique ETEs.

The assumptions regarding the population allocation for each evacuation scenario are described in ETE Report Table 6-3, “Percent of Population Groups for Various Scenarios.” The percentage of each group of the public for each of the scenarios includes households with returning commuters, households without returning commuters, employees, transients, shadow, and special events. ETE Report Section 5, “Estimation of Trip Generation Time,” describes the process used to develop distributions of elapsed times associated with mobilization activities for each population subgroup for each scenario. A telephone survey of residents of the 16-km (10-mi) EPZ was conducted to gather data for trip generation time elements. In RAIA 25, Question 13.03-24, the staff requested that the applicant provide additional information regarding trip generation times for transients. In a November 20, 2008, response, the applicant clarified that the 2-hour mobilization time for transients is adequate for those transients who return to their lodging facilities before evacuating. In RAI 25, Questions 13.03-45(A) through 13.03-45(F), the staff requested that the applicant provide additional information regarding use of truncated distributions of mobilization times. For each question (A) through (F), in the applicant's December 17, 2008, response the applicant explained that distributions were shortened because the objective was to evaluate the evacuation times under circumstances of greatest highway loading, given the uncertainty about those who are reluctant to prepare to leave or unwilling to evacuate. The applicant stated that although a telephone survey had indicated that some people would require as much as 6 hours preparing for an evacuation, it had used a 4-hour preparation time in its ETE calculations.

The ETEs range from one hour thirty minutes to three hours twenty minutes for the ninetieth percentile normal weather general population. The maximum ETE for the one hundredth percentile is four hours twenty minutes for evacuation during special event, which is the construction of the new plant. Separate ETEs were developed for the transit-dependent and special facility populations. In RAI 25, Question 13.03-46, the staff requested that the applicant provide an explanation for the absence of a "prepare to leave activity" and "travel home" sequences for scenarios on weekends. ETE Report Figure 5-1, "Events and Activities Preceding the Evacuation Trip," was revised to show these activities. The applicant provided data and an analysis example, which explains that due to the small number of people in the affected region working on weekends, the times for those sequences are negligible. Trip generation times were based on results from a telephone survey of the region. In RAI 25, Question 13.03-47, the staff requested that the applicant provide additional information about normalization of the "Don't Know" response to the telephone survey. In a November 24, 2008, response, the clarified that, in effect, the "Don't Know" responses are ignored.

ETE Report Section 6, "Demand Estimation for Evacuation Scenarios," defines the various evacuation cases for which time estimates were made, where a case is a combination of a scenario and a region. A scenario is defined to be a combination of circumstances, including time of day, day of week, season, and weather conditions. Scenarios define the number of people in each of the affected population groups and their respective mobilization time distributions. A region is defined to be a grouping of contiguous evacuation zones, which forms either a "keyhole" sector based area, or a circular area within the plume exposure pathway EPZ, that is evacuated in response to a radiological emergency. The WLS plume exposure pathway EPZ has been defined to contain 14 ERPAs, with boundaries along major roads or rivers. The boundaries do not bisect any population centers.

A summary of the ETEs are provided in ETE Report Section 7, "General Population Evacuation Time Estimates (ETE)." The evacuation times are presented for all 22 evacuation regions and 12 scenarios in ETE Report Appendix J, "Evacuation Time Estimates for All Evacuation Regions and Evacuation Time Graphs for Region 3 (R3), for All Scenarios." Results are presented for 50 percent, 90 percent, 95 percent, and 100 percent of vehicles. Results are also provided for good and adverse (rainy) weather conditions. Evacuation times are reported separately for the general population, schools, and transit-dependent population. ETE Report Figures J-1 through J-12, "Evacuation Time Estimates – Scenarios 1 through 12 for Region 3 (Entire EPZ)," describe the time distribution of evacuating vehicles. In RAI 25, Question 13.03-2, the staff requested that the applicant provide clarification regarding whether the results presented in ETE Report Section 7 included schools, transit dependents, and special facilities. In a November 7, 2008, response, the applicant clarified that the ETE Report Section 7 results include only the general population and that the school, transient, and special facilities populations are reported separately. In RAI 25, Question 13.03-17, the staff requested that the applicant provide clarification regarding the use of rain as adverse conditions rather than icy conditions. In a November 11, 2008, response, the applicant stated that the counties considered icy conditions to be a low probability event that was not seriously under represented by rainy conditions. The apparent lack of impact of adverse weather on the calculated ETE values was questioned by the staff in RAI 25, Question 13.03-49. In a November 24, 2008, response, the applicant clarified that rain reduces the free travel speed somewhat, which is generally not sufficient by itself to increase the ETE, due to the relatively short trip lengths. In RAI 25, Question 13.03-33, the staff requested that the applicant provide clarification regarding the "long tail" of the traffic flow rates as a function of time. In a November 25, 2008, response, the applicant explained that the "long tail" of the curve in the Figure 7-6 of the ETE Report is where the slope of the curve has decreased to the point of being nearly horizontal. The staff finds the applicant's explanation acceptable.

The IDYNEV System, which includes the PC-DYNEV macroscopic simulation model and the Traffic Assignment Model (TRAD), was used in the analysis. Inputs, outputs, and measures of effectiveness for the traffic simulation model are discussed throughout the ETE Report, and a step by step process for integrating the data and models to produce ETEs is provided.

The combined time dependent mobilization distribution, which represents traffic loading, is presented in ETE Report Figure 5-3, "Evacuation Trip Generation for Various Population Groups." The simulation model analyzes when and where roadways are nearing capacity, and calculates the travel delays, queuing lengths, and travel times throughout the network. Traffic queuing and congestion areas are presented in ETE Report Figures 7-3 through 7-5, showing a Level of Service F, which indicates heavy congestion, is observed in and around Gaffney, SC from 30 minutes until more than 2 hours after the advisory to evacuate. After about 3 hours, heavy congestion within the 16-km (10-mi) EPZ is no longer present.

The ETE study describes the evacuation of ambulatory persons from special facilities as requiring 54 wheelchair bus runs. The resources required for each facility are identified in ETE Report Table 8-4, "Special Facility Transit Demand." ETE Report, Revision 2, Section 8.4, "Evacuation Time Estimates for Transit Dependent People," explains that wheelchair buses and vans are often scarce, but that regular buses can be used to transport wheelchair patients.
In RAI 123, Question 13.03-98, the staff requested that the applicant provide information regarding the resources required to evacuate the special facilities. In a November 6, 2014, response, the applicant clarified that the resources identified in ETE Report Table 8-4 do not include regular buses. The applicant explained that in an earlier response on December 11, 2009, to RAI 83, Question 13.03-76, limitations on school buses and the impact to the ETE were discussed. The applicant explained that due to these limitations, the ETE for a multiple wave response was developed resulting in an 8-hour ETE for good weather and 9 hours 15 minutes for rain for schools and the transit-dependent population. However, the maximum ETE for schools shown in the ETE Report, Revision 2, Table 8-5E, "Third Wave School Evacuation Time Estimate – Good Weather," is 5 hours 45 minutes, and the maximum ETE for rain for the third wave is 6 hours 35 minutes. In RAI 123, Question 13.03-98, the staff requested that the

applicant provide information regarding the impact to the ETE due to time to acquire regular buses and time for buses to complete other activities prior to supporting special facility evacuations. In a November 6, 2014, response, the applicant described the times needed to evacuate schools and the transit-dependent population are 8 hours in good weather and 9 hours 15 minutes in rain. The times were given in response to RAI 83, Question 13.03-76. The assumption that the applicant held is that there would be three waves of school evacuations before a single evacuation wave for the transit-dependent population. That is, the school evacuations have priority.

The times given were the sum of the average elapsed time to complete the three waves of school evacuations and the average of the single wave of transit-dependent evacuations. In each case, the applicant limited the evacuation to use only the 60 buses of Cherokee County. With both of the assumptions, the evacuation times are as follows:

Evacuation Population	Evacuation Times (hr:min)	
	Good Weather	Rain
Schools	5:30 (ETE Report Table 8-5E)	6:15 (ETE Report Table 8-5F)
Transit-dependent	2:30 (ETE Report Table 8-7A)	2:55 (ETE Report Table 8-7B)
Total	8:00	9:15 (sum rounded up)

The results prompted Cherokee County to find and use other resources, which are available through the South Carolina Statewide Mutual Aid Agreement for Catastrophic Disaster Response and Recovery. Using the mutual aid agreement will ensure sufficient transportation resources are available for timely evacuation of school children and transit-dependent people and allow the evacuations of both populations to occur simultaneously. The use of additional resources through the mutual aid agreement will reduce the amount of evacuation times. The staff finds the applicant’s responses to RAI 25, Question 13.03-76 and RAI 123, Question 13.03-98 acceptable.

A series of sensitivity tests are documented in Appendix I, “Evacuation Sensitivity Studies,” including sensitivity of the results to trip generation time (directly related to time-dependent traffic loading) and to the amount of shadow evacuation.

The ETE Report includes separate calculations for special populations of school children and transit-dependent individuals in ETE Report Section 8, “Transit Dependent and Special Facility Evacuation Time Estimates.” In RAI 25, Question 13.03-8, the staff requested that the applicant provide clarification of assumptions regarding mobilization of school buses. In a November 11, 2008, response, the applicant stated that the county authorities had suggested 90 minutes for Cherokee County schools, but that the schools in York and Cleveland Counties required only 30 minutes because the buses for those schools remained at the schools. Telephone survey results are used to estimate the portion of the population requiring transit service, including persons in households that do not have a vehicle available and persons in households that have vehicles, but these vehicles would not be available at the time the evacuation is ordered. ETE Report Table 8-1, “Transit Dependent Population Estimates,” shows 2,539 transit-dependent people, of which 1,270 of these would require public transport. ETE Report Table 8-6, “Summary of Transit Dependent Bus Routes for the Lee Nuclear Station,” lists 42 bus runs on 11 routes to evacuate this population. In the ETE Report, Revision 2, Section 8.4, “Evacuation Time Estimates for Transit Dependent People,” the applicant explained that the dispatch of buses should consider the time for transit-dependent evacuees to complete their mobilization activities and be in position to board buses when they arrive. ETE Report Section 8.4 states that bus resources are assigned to evacuating schoolchildren as the first priority. In RAI 123, Question 13.03-99, the staff requested that the applicant provide information regarding the resources needed to evacuate the transit-dependent population. In a November 6, 2014,response, the applicant explained that the times in ETE Report Tables 8-7A and 8-7B do not begin at the end of the three wave school evacuation because it is assumed that additional buses from neighboring counties will be available pursuant to mutual aid agreements. The staff finds the applicant’s response to RAI 123, Question 13.03-99 acceptable.

Proposed routes for transit-dependent and special facility populations are shown in ETE Report Figure 8-2, “Proposed Transit Dependent Bus Routes.” Assumed general population reception centers are shown in ETE Report Figure 10-1, “Assumed General Population Reception Centers.” Clarification of bus routes was requested by the staff in RAI 25, Question 13.03-16. In a November 20, 2008, response, the applicant included a revised ETE Report Figure 8-2 outlining the assumed routing for buses. In RAI 25, Question 13.03-18, the staff requested that the applicant provide clarification of travel times for bus service through congested areas of the City of Gaffney, South Carolina. In a November 20, 2008, response, the applicant stated that average speeds in congested areas include periods of higher speed outside of the congested zones. In RAI 25, Question 13.03-19, the staff requested that the applicant explain the assumptions regarding bus return times for “second wave” evacuation. In a November 20, 2008, response, the applicant clarified that the travel times back to the 16-km (10-mi) EPZ for those buses performing a second wave evacuation of transit dependents are the average travel times based on assumed bus speeds of 72 kph and 64 kpm (45 mph and 40 mph) for good weather and rain, respectively. The bus routes identified in ETE Report Figure 8-2 illustrate the primary evacuation routes and are described in ETE Report Table 8-6, “Summary of Transit Dependent Bus Routes for the Lee Nuclear Station,” which identifies buses travelling major roadways. The Cherokee County Emergency Operating Procedure states that residents should walk to the nearest public school if it is within 0.8 km (0.5 mi) or contact the Cherokee County Emergency Management Agency for assistance. In RAI 123, Question 13.03-100, the staff requested that the applicant provide a description of how the ETE calculation considers the time for buses to travel residential routes to pick up those residents living more than one-half mile from schools. In response, the applicant referenced ETE Report Figure 5-3, “Comparison of Trip Generation Distributions,” show that 85 percent of these evacuees will be ready in 90 minutes, and 100 percent would be ready in 180 minutes. The applicant explained that ETE Report Tables 8-7A and B show 90 minutes for mobilization in good weather and 100 minutes for mobilization in the rain, respectively. The ETE Report shows an additional 30 minutes for pickup time. When considering pickup times, route travel times, and return to the EPZ, the second wave of evacuation takes more than 180 minutes to begin, showing that there is time for all residents to mobilize. In RAI 123, Question 13.03-101, the staff requested that the applicant provide information regarding the ETE values for the transit-dependent population. In a

November 6, 2014, response, the applicant explained that ETE Report, Tables 8-7A and B include time for all buses to complete the evacuation route because the buses travel the routes simultaneously. With this process, there is no impact to the ETE because all buses are considered in the analysis. The staff finds the applicant's responses to RAI 123, Questions 13.03-100 and 13.03-101 acceptable.

The quantity and type of specialized vehicles required to support evacuation of special facilities and special needs populations are identified. The ETE Report, Revision 2 explains that bus mobilization time for special facilities is estimated to be 90 minutes based on experience at other plants. In RAI 123, Question 13.03-102, the staff requested that the applicant provide additional information regarding bus mobilization time. In a November 6, 2014, response, the applicant explained that although previous experience for mobilization times was initially based on other plants, the counties either confirmed or corrected the mobilization times, and county specific times were used in the analysis. In ETE Report Section 8.5, "Special Needs Population," the ETE Report, Revision 2 explains that 24 ambulance runs are needed to evacuate the special facility bed-ridden population and 10 ambulance runs are needed to evacuate the homebound special needs residents. The ETE study describes the time to evacuate the special facility residents first, followed by the homebound population. The study explains that ambulances will be provided from within the 16-km (10-mi) EPZ and additional ambulances will be provided by neighboring cities. In RAI 123, Question 13.03-103, the staff requested that the applicant provide additional information regarding ambulance resources. In a November 6, 2014, response, the applicant clarified that there are 17 ambulances available within 30 minutes, and 7 of these are based in the EPZ. An additional 30 ambulances are available within 90 minutes of notification. The applicant explained that the ETE Report includes the time for ambulances to make return trips to pick up the next set of residents. Using the ambulances that are available within 90 minutes of notification would reduce the ETE value. The maximum ETE for this population segment is 4 hours and 25 minutes. The staff finds the applicant's responses to RAI 123, Questions 13.03-102 and 13.03-103 acceptable.

A total of 264 evacuation time estimates are computed for the evacuation of the general public. Each evacuation time estimate quantifies the aggregate evacuation time estimated for the population within one of the 22 Evacuation Regions to completely evacuate from that Region, under the circumstances defined for one of 12 evacuation scenarios (22 x 12 = 264). Separate evacuation time estimates are calculated for transit-dependent evacuees, including school children. An acceptable variant of the NUREG-0654 format is used for the presentation of the evacuation times.

Distribution functions for notification of the various categories of evacuees were developed. The distribution functions for the action stages after notification predict what fraction of the population will complete a particular action within a given span of time. There are separate distributions for auto-owning households, school population, and transit-dependent populations. These times are combined to form the trip generation distributions.

On-road travel and delay times are calculated. An estimate of the time required to evacuate a particular segment of the non-auto-owning population dependent on public transportation is developed, in a manner similar to that used for the auto-owning population.

The staff finds the clarifications, additional information, and textual revision submitted by the applicant in response to RAI 25, Questions 13.03-2, 13.03-8, 13.03-16, 13.03-17, 13.03-18, 13.03-19, 13.03-24, 13.03-33, 13.03-45(A through F), 13.03-46, 13.03-47, and 13.3-49 acceptable because they conform to the guidance in NUREG-0654. The staff also confirmed that Revision 2 of the WLS Emergency Plan incorporated the information and textual changes provided in the response to RAI 25. Questions 13.03-16 and 13.03-46.

The staff finds that the ETE Report adequately addresses the descriptions of the methods used to estimate the evacuation times. The staff finds this acceptable because it conforms to the guidance in NUREG-0654, Appendix 4, Section IV.

Other Requirements

ETE Report Section 11, "Surveillance of Evacuation Operations," addresses monitoring of the evacuation by use of staff at traffic control points, ground and aerial surveillance and citizen reports via cellular telephones. Surveillance of the evacuation will be coordinated and executed by local authorities. ETE Report Section 12, "Confirmation Time," describes the necessity to confirm the evacuation process and explains this is a county level responsibility. In RAI 13.03-50, the staff requested that the applicant provide clarification regarding assumptions regarding telephone surveys and county agreements on methods of confirming evacuation. In a November 24, 2008, response, the applicant identified an advantage to the telephone based approach and suggested the approach could be reinforced with ground vehicles if decided at a later date. The use of a telephone survey is one approach suggested in NUREG-0654.

Intelligent Transportation Systems (Dynamic Message Signs, Highway Advisory Radio, Automated Traveler Information Systems, etc.) are discussed in ETE Report Section 9. In RAI 25, Question 13.03-51, the staff requested that the applicant provide additional information regarding the use of such systems in the ETE analysis. In a November 24, 2008, response, the applicant clarified that the various intelligent transportation systems were not credited in the development of the ETE, and the results of the ETE are not dependent on their use.

The development of the ETE Report was coordinated with emergency planners from the State of South Carolina, and from Cherokee and York Counties, which are involved in emergency response for the site. In RAI 25, Question 13.03-3, the staff requested that the applicant provide information regarding the review of the ETE Report by State and local organizations involved with emergency response. In a November 7, 2008, response, the applicant described collaboration with State and local emergency management officials and law enforcement personnel in developing the ETE analysis and resolving their comments on the draft ETE Report. In addition, RAI 25, Question 13.03-3, the staff requested that the applicant provide clarification regarding whether State and local organizations provided any comments and that any comments and their resolution be provided as additional information. In a November 7, 2008, response to RAI 25, Question 13.03-3, the applicant described its collaboration with State and local emergency management officials and law enforcement personnel in developing the ETE analysis and resolving their comments on the draft ETE Report. Also, in RAI 25, Question 13.03-3, the staff requested that the applicant provide specific clarification regarding whether the State and local emergency response agencies had concurred with traffic control point and access control point selection and arrangements. In a November 7, 2008, response, the applicant stated that State and local emergency response agencies had concurred with the traffic control points during the development of the Emergency Plan and ETE, and had signed letters of commitment to support the Emergency Plan and ETE. In a November 11, 2008, response to RAI 25, Question 13.03-9, in which the staff requested that the applicant provide clarification of the use of traffic control points in the ETE estimate, the applicant explained that the functions of the traffic control points were not assumed in the ETE estimation and would act to reduce the time if employed. The staff noted in RAI 25, Question 13.03-53 that ETE Report Table G-1, "Lee Nuclear Station Traffic Control Points," that summarized the traffic control points was referred to but omitted from the ETE Report. The applicant's response rectified this omission.

In the March 4, 2010, submission of Revision 2 of the ETE Report, the applicant explained that ETE Report, Revision 2 was provided to State and county emergency management agencies for review and comment. In a March 4, 2010, letter, the applicant provided a list of state and county agencies that had been provided a copy of Revision 2 of the ETE Report. The applicant also provided copies of signed letters from these agencies which indicate they have reviewed the document.

The applicant estimated the time required for confirmation of evacuation. In addition, the development of the ETE Report was coordinated with emergency planners from the state and counties who are involved in emergency response for the site.

The staff finds the clarifications, additional information, and textual revision submitted by the applicant in response to RAI 25, Questions 13.03-3, 13.03-9, 13.03-50, 13.03-51, and 13.03-53 acceptable because they conform to the guidance in NUREG-0654. The staff confirmed that Revision 2 of the WLS Emergency Plan incorporated the information and textual changes provided in the response to RAI 25, Question 13.03-53. The staff finds the ETE Report adequately addresses the description of the procedure to confirm that the evacuation process is effective. This is acceptable because it conforms to the guidance in NUREG-0654, Appendix 4, Section V.

Conclusions

The ETE Report describes the method of analyzing the evacuation times. A general description of the evacuation model was provided including the assumptions used in the evacuation time estimate analysis.

The staff finds the ETE Report adequately addresses the estimate of the number of people who may need to be evacuated. This is acceptable because it conforms to the guidance in NUREG-0654, Appendix 4, Section II.

The ETE Report provides a complete review of the evacuation road network. Analyses are made of travel times and potential locations for congestion. All evacuation route segments and their characteristics, including capacity, are described.

A traffic control and management strategy that is designed to expedite the movement of evacuating traffic is described. The traffic management strategy is based on a field survey of critical locations and consultation with emergency management and enforcement personnel. The applicant also analyzed travel times and potential locations for serious congestion along the evacuation routes.

The time required for confirmation of evacuation was estimated. In addition, the development of the ETE Report was coordinated with emergency planners from the state and counties who are involved in emergency response for the site.

On the basis of its review of the analysis of the ETE Report as described above, the staff concluded that the information provided in the ETE Report is consistent with the guidance in Appendix 4 to NUREG-0654, Appendix 4; NUREG-6368; and portions of NUREG/CR--002 as described above. Therefore, the staff considers the ETE Report acceptable and meets the applicable requirements of 10 CFR Part 50, Appendix E.IV.

13.3.4.18 AP1000 COL Information Items

WLS COLA FSAR Table 1.8-202, "COL Item Tabulation," identifies two COL information items from AP1000 DCD Tier 2, Section 13.3.1, relating to EP. These consist of STD COL 13.3-1 and STD COL 13.3-2, which correspond to COL Action Items 13.3-1 and 13.3.3.3.5-1, respectively, in NUREG-1793, Section 13.3. The following addresses the resolution of the two COL information items.

- STD COL 13.3-1

STD COL Information Item 13.3-1 requires that COL applicants referencing the AP1000 certified design will address EP, including post-72 hour actions and its communication interface. In consideration of WLS COL 13.3-1, the applicant addressed STD COL 13.3-1 in WLS COL FSAR Section 13.3 by stating the following:

> The emergency planning information is submitted to the Nuclear Regulatory Commission as a separate licensing document (WLS COL 13-3.1).
>
> Post-72 hour support actions, as discussed in DCD Subsections 1.9.5.4 and 6.3.4, are addressed in DCD Subsections 6.2.2, 8.3, and 9.1.3. Provisions for establishing post-72 hour ventilation for the main control room, instrumentation and control rooms, and dc [direct current] equipment rooms are established in operating procedures.

The staff's evaluation of communication interfaces is addressed above in Section 13.3.4.6, "Emergency Communications," of this report. With regard to post-72 hour actions associated with the AP1000 DCD, the applicant referenced operating procedures and various AP1000 DCD sections listed above that address post-72 hour support actions. The staff identified additional AP1000 DCD Tier 2 sections that address post-72 hour support actions, which include AP1000 DCD Sections 6.4, "Habitability Systems"; 9.4, "Air-Conditioning, Heating, Cooling, and Ventilation System"; and 9.5, "Other Auxiliary Systems," for example, plant lighting systems described in AP1000 DCD, Section 9.5.3.

As discussed in AP1000 DCD Section 1.9.5.4, post-72 hour support actions relate to an extended loss of the non-safety-related systems for both offsite and onsite ac power sources for more than 72 hours. For purposes of the staff's review of EP information in the WLS COLA, and in the context of COL Action Item 13.3-1, the reference to post-72 hour support actions is limited and indirectly related to the habitability and functionality of the TSC. Specifically, it is limited to the reliability of the electrical power supply (post-72 hours) to the TSC ventilation system and communication equipment. The evaluation of the reliability of the electrical power supplies, including the power supplies to the TSC, is addressed in the AP1000 DCD sections referenced above. The habitability and functionality of the TSC is addressed in Section 13.3.4.8 of this report.

The staff finds that the applicant has addressed EP communication interfaces in support of WLS Units 1 and 2 in the WLS Emergency Plan. In addition, the applicant has addressed post-72 hour actions through reference to the AP1000 DCD sections that specifically address an extended loss of the non-safety-related systems for both offsite and onsite ac power sources for more than 72 hours. The staff's evaluation of those systems and power sources, including the establishment of associated operating procedures, are addressed in their respective sections of this report. Therefore, the staff finds that the COL applicant has adequately addressed STD COL 13.3-1.

- STD COL 13.3-2

STD COL 13.3-2 requires that COL applicants referencing the AP1000 certified design will address the activation of the EOF, consistent with current operating practice and NUREG-0654. In WLS COL FSAR Section 13.3, the applicant addressed STD COL 13.3-2 by stating that the emergency plan describes the plans for coping with emergency situations, including communication interfaces and staffing of the EOF.

Activation and staffing of the EOF is described in the WLS Emergency Plan, and the staff's evaluation of this information is addressed in Section 13.3.4.2, "Onsite Emergency Organization," Section 13.3.4.3, "Emergency Response Support and Resources," Section 13.3.4.5, "Notification Methods and Procedures," and Section 13.3.4.8, "Emergency Facilities and Equipment," of this report. Communication interfaces are addressed in Section 13.3.4.6, "Emergency Communications," of this report. Therefore, the staff finds that the COL applicant has adequately addressed STD COL 13.3-2.

- WLS COL 9.5-9 and WLS COL 9.5-10

WLS COL 9.5-9 requires that COL applicants referencing the AP1000 certified design will address interfaces to required offsite locations, including the recommendations of BL-80-15 regarding loss of the emergency notification system due to a loss of offsite power. In addition, WLS COL 9.5-10 requires that COL applicants referencing the AP1000 certified design will address the emergency offsite communication system, including the crisis management radio system. In WLS COL FSAR Section 9.5.2.2.3, the applicant addressed WLS COL 9.5-9 and WLS COL 9.5-10 together by stating that offsite interfaces and emergency offsite communication are described in the emergency plan (See also, WLS COL FSAR Table 1.8-202).

The applicant described the emergency notification systems, including the ENS, in WLS Emergency Plan, Section II.E, and the emergency communication systems in WLS Emergency Plan, Section II.F. The staff's evaluation of offsite emergency notification and communication systems is addressed in Sections 13.3.4.5 and 13.3.4.6, respectively of this report. Therefore, the staff finds that the COL applicant has adequately addressed WLS COL 9.5-9 and WLS COL 9.5-10, with regard to emergency planning for WLS Units 1 and 2. Offsite interfaces and emergency offsite communication are discussed further in Section 9.5.2, "Communication System," of this report.

- WLS COL 18.2-2

WLS COL 18.2-2 requires that COL applicants referencing the AP1000 certified design will provide specific information regarding EOF and TSC communication and human factors attributes. WLS COL FSAR Table 1.8-202 identifies WLS COL FSAR Section 18.2.1.3 as the location where WLS COL 18.2-2 is addressed. In WLS COL FSAR Section 18.2.1.3, the applicant addressed WLS COL 18.2-2 by stating that the EOF and TSC communication strategies, as well as the EOF and TSC human factors attributes, are described in the emergency plan.

The applicant described EOF and TSC communication and human factors attributes in WLS Emergency Plan, Sections II.E, II.F, and II.H. The staff's evaluation is addressed in Sections 13.3.4.5, 13.3.4.6, and 13.3.4.8, respectively, of this report. Therefore, the staff finds that the COL applicant has adequately addressed WLS COL 18.2-2, with regard to emergency planning for WLS Units 1 and 2. WLS COL 18.2-2 is discussed further in Section 18.2, "Human Factors Engineering Program Management," of this report.

13.3.4.19 Supplemental Information

- STD SUP 13.3-1

Activities that the COL holder shall perform after the COL is issued, that are applicable to emergency planning, consist of the implementation milestones and license conditions. The applicant provided supplemental information in STD SUP 13.3-1, which states that WLS COL FSAR Table 13.4-201, "Operational Programs Required by NRC Regulations," provides milestones for emergency planning implementation. WLS COL FSAR Table 13.4-201 identifies the emergency planning program as operational program Item No. 14, and includes the four associated implementation milestones listed below. (See also, Table 13.3-1 of this report, ITAAC 8.1.) The staff reviewed WLS COL FSAR Table 13.4-201, and finds that the identified implementation milestones associated with the emergency planning program are acceptable because they are consistent with the relevant guidance and acceptance criteria in NUREG-0800; therefore, the milestones meet the respective requirements in 10 CFR Part 50, Appendix E. (See also, Sections 1.5.5[11] and 13.4 of this report.)

Implementation Milestones

- A full participation exercise conducted within 2 years of the scheduled date for initial loading of fuel, as required by 10 CFR Part 50, Appendix E, Section IV.F.2(a)(ii)
- Onsite exercise conducted within 1 year before the scheduled date for initial loading of fuel, as required by 10 CFR Part 50, Appendix E, Section IV.F.2(a)(ii)

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[11] Section 1.5.5, "Receipt, Possession, and Use of Source, Byproduct and Special Nuclear Material Authorized by 10 CFR Part 52 Combined Licenses," of this report addresses implementation milestones for the various operational programs (including emergency planning) relating to byproduct, source, and special nuclear material (pursuant to 10 CFR Part 30, "Rules of general applicability to domestic licensing of byproduct material"; 10 CFR Part 40, "Domestic licensing of source material"; and 10 CFR Part 70, "Domestic licensing of special nuclear material").

- Licensee's detailed implementing procedures for its emergency plan submitted at least 180 days prior to the scheduled date for initial loading of fuel, as required by 10 CFR Part 50, Appendix E, Section V

- EPZ population change review conducted at least 365 days before scheduled date for initial loading of fuel, as required by 10 CFR Part 50, Appendix E, Section IV.7

License Condition 6

In WLS COLA Part 10, the applicant proposed License Condition 6 to provide a schedule to support NRC inspection of operational programs, including (a) the EPIPs, and (e) an ERDS implementation program plan. The applicant proposed the following, emergency preparedness parts extracted:

> The licensee shall submit to the appropriate Director of the NRC, a schedule, no later than 12 months after issuance of the COL, that supports planning for and conduct of NRC inspections of operational programs listed in the operational program FSAR Table 13.4-201. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until either the operational programs in the FSAR table have been fully implemented or the plant has been placed in commercial service, whichever comes first. This schedule shall address:
>
> a. the emergency planning implementation procedures to the NRC consistent with 10 CFR Part 50, Appendix E, Section V
>
> e. an emergency response data system (ERDS) implementation program plan consistent with 10 CFR Part 50, Appendix E, Section VI

The schedule for submission of the EPIPs to the NRC is addressed in WLS COL FSAR Table 13.4-201. (See implementation milestone, discussed above.) The ERDS program, including implementation, is addressed above in Sections 13.3.4.5, 13.3.4.6, 13.3.4.8 and 13.3.4.14 of this report and in EP-ITAAC 3.2. The staff reviewed License Condition 6 against the recommendations in SECY-05-0197, as endorsed by the related February 22, 2006, SRM. The staff concludes that License Condition 6 conforms to the guidance in SECY-05-0197, with regard to implementation of the emergency planning program (including EPIPs and an ERDS), and is therefore acceptable. Consistent with the applicant's proposed License Condition 6, the staff identified License Condition (13-8) and License Condition (13-9), which include minor revisions to reflect the emergency planning operational program:

License Conditions (13-8) and (13-9)

> No later than 12 months after issuance of the COL, Duke Energy shall submit to the NRC a schedule that supports planning for and conduct of NRC inspection of the emergency planning program implementation as identified in FSAR Table 13.4-201. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until the emergency planning operational program has been fully implemented. This schedule shall address the following two items from FSAR Table 13.4-201:

a. the emergency planning implementation procedures submitted to the NRC consistent with 10 CFR Part 50, Appendix E, Section V. (License Condition (13-8))

e. an emergency response data system (ERDS) implementation program plan consistent with 10 CFR Part 50, Appendix E, Section VI. (License Condition (13-9))

(See also, Section 13.4, “Operational Programs,” of this report.)

ITAAC

- WLS SUP 14.3-1

As stated in Section 13.3.2 of this report, the applicant proposed a license condition to incorporate EP ITAAC into the COL, which are identified in Table 3.8-1 of Appendix B to Part 10 of the WLS COL application. WLS COLA Part 10, Appendix B includes Table 3.8-1 (EP-ITAAC) and incorporates by reference AP1000 DCD Tier 1, Table 3.1-1 (ITAAC). AP1000 DCD Table 3.1-1 also addresses the AP1000 locations of the OSC and TSC, which are changed by WLS COLA Departure WLS DEP 18.8-1 and evaluated in Section 13.3.4.8 of this report. In addition, in WLS COL FSAR Section 14.3.2.3.1, the applicant provided supplemental information in WLS SUP 14.3-1, which states:

> Emergency Planning ITAAC (EP-ITAAC) have been developed to address implementation of elements of the Emergency Plan. Site-specific EP-ITAAC are based on the generic ITAAC provided in Appendix C.II.1-B of Regulatory Guide 1.206. These ITAAC have been tailored to the specific reactor design and emergency planning program requirements.

The staff reviewed the complete set of EP-ITAAC for WLS, which consist of the EP-ITAAC in WLS COLA Part 10, Table 3.8-1 plus the EP ITAAC in AP1000 DCD Tier 1, Table 3.3-1. The staff identified that proposed EP-ITAAC Acceptance Criterion 3.1.4 omitted communication links between the EOF and the radiological monitoring teams as compared to the EP-ITAAC in NUREG-0800, Table 14.3.10-1. Therefore, the staff revised the proposed EP-ITAAC Acceptance Criterion 3.1.4 to conform to NUREG-0800, Table 14.3.10-1. Similarly, the staff identified instances in EP-ITAAC Acceptance Criteria 8.1.1.2.E.4.b and EP-ITAAC 8.1.1.2.E.7.c with ambiguous criteria that did not conform to NUREG-0800, Table 14.3.10-1. The staff revised the acceptance criteria by replacing “and/or” with “or.” The staff noted that the revised EP-ITAAC are collectively adequate because they conform to the respective acceptance criteria in NUREG-0800, Section 14.3.10.[12] Therefore, the staff finds that the WLS SUP 14.3-1 is adequately addressed and that the EP-ITAAC in WLS COLA Part 10, Table 3.8-1, as modified in Table 13.3-1 of this report, and AP1000 DCD Table 3.3-1 are acceptable because they are consistent with NUREG-0800.

[12] The generic EP ITAAC in RG 1.206 Appendix B, Table C.II.1-B1 are identical to the generic EP ITAAC in NUREG-0800, Section 14.3, Table 14.3.10-1.

13.3.5 Post Combined License Activities

The license condition language in this section has been clarified from previously considered language. In a letter dated March 22, 2016 (ADAMS Accession No. ML16084A099), the applicant did not identify any concerns with the clarified license condition language. The changes do not affect the staff's above analysis of the conditions, and therefore, for the reasons discussed in the technical evaluation section above, the staff finds the following ITAAC and license conditions acceptable:

- The licensee shall perform and satisfy the ITAAC defined in SER Table 13.3-1, "Emergency Plan ITAAC."

- License Condition (13-3) – No later than one hundred eighty (180) days before the date scheduled for initial fuel load set forth in the notification submitted in accordance with 10 CFR § 52.103(a), the licensee shall submit to the Director of NRO, or the Director's designee, in writing, a fully developed set of plant-specific emergency action levels (EALs) for WLS Unit [1 and 2], in accordance with NEI 07-01, "Methodology for Development of Emergency Action Levels – Advanced Passive Light Water Reactors," Revision 0, with no deviations. The EALs shall have been discussed and agreed upon with State and local officials. (See Section 13.3.4.4 of this report.)

- License Condition (13-4) – Prior to the full participation exercise to be conducted in accordance with the requirements of 10 CFR Part 50, Appendix E, Duke Energy shall identify the specific locations of the reception centers and relocation sites and shall obtain Letters of Agreement for locations not under the Duke Energy's control.

- License Condition (13-5) – No later than 18 months before the latest date set forth in the schedule submitted in accordance with 10 CFR § 52.99(a) for completing the inspections, tests, and analyses in the ITAAC, the licensee shall have performed a detailed staffing analysis, in accordance with NEI 10-05, "Assessment of On-Shift Emergency Response Organization Staffing and Capabilities," Revision 0.

  No later than one hundred eighty (180) days before the date scheduled for initial fuel load set forth in the notification submitted in accordance with 10 CFR § 52.103(a), the licensee shall have revised the Emergency Plan to incorporate any changes identified in the staffing analysis that are needed to bring staffing to the required levels.

- License Condition (13-6) – No later than 180 days before the date schedule for initial fuel load set forth in the notification submitted in accordance with 10 CFR 52.103(a), Duke Energy shall submit to the Director of the Office of New Reactors (NRO), or the Director's designee, in writing, updated WLS Units 1 and 2, Letters of Agreement with the following entities, or their successors. These updated Letters of Agreement shall identify the specific nature of arrangements in support of emergency preparedness for WLS, and reflect expected assistance associated with hostile action at the WLS, as defined in 10 CFR Part 50, Appendix E, Section IV.A.7. The WLS Emergency Plan shall have been revised to include these updated Letters of Agreement after they have been executed.

  1. South Carolina Emergency Management Division

2. Piedmont Medical Center
3. Upstate Medical Center, Emergency Medical Services
4. Draytonville-McKown Mountain-Wilkinsville Volunteer Fire Department
5. Cherokee County Emergency Management
6. Cleveland County Emergency Management and Fire Marshall's Office
7. North Carolina Emergency Management
8. South Carolina Department of Health and Environmental Control
9. York County Emergency Management

- License Condition (13-7) – Prior to fuel load, DEC will demonstrate the integrated capability and functionality of the EOF for activation and operation of the facility to respond to emergency events at WLS and one additional nuclear site that is supported by the EOF. Integrated communication and data capability and functionality will include the Technical Support Centers for WLS and one additional nuclear site, and other Federal, State, and local coordination centers as appropriate.

- License Condition (13-8) – The licensee shall develop a schedule that supports planning for and conduct of NRC inspections of the operational programs listed in WLS COL FSAR Table 13.4-201, "Operational Programs Required by NRC Regulations." This schedule must be available to the NRC staff no later than 12 months after issuance of the COL. The schedule shall be updated every 6 months until 12 months before scheduled fuel load, and every month thereafter until the operational programs listed in WLS COL FSAR Table 13.4-201 have been fully implemented. This schedule shall include a schedule for submitting the EP implementing procedures to the NRC consistent with 10 CFR Part 50, Appendix E, Section V.

- License Condition (13-9) – The licensee shall develop a schedule that supports planning for and conduct of NRC inspections of the operational programs listed in WLS COL FSAR Table 13.4-201, "Operational Programs Required by NRC Regulations." This schedule must be available to the NRC staff no later than 12 months after issuance of the COL. The schedule shall be updated every 6 months until 12 months before scheduled fuel load, and every month thereafter until the operational programs listed in WLS COL FSAR Table 13.4-201 have been fully implemented. This schedule shall address an emergency response data system (ERDS) implementation program plan to the NRC consistent with 10 CFR Part 50, Appendix E, Section VI.

- License Condition (13-10) – No later than eighteen (18) months before the latest date set forth in the schedule submitted in accordance with 10 CFR § 52.99(a) for completing the inspections, tests, and analyses in the ITAAC, the licensee shall have performed an assessment of the on-site and augmented staffing capability for response to a multi-unit event. The staffing assessment shall be performed in accordance with NEI 12-01, “Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities,” Revision 0.

No later than one hundred eighty (180) days before the date scheduled for initial fuel load, as set forth in the notification submitted in accordance with 10 CFR § 52.103(a), the licensee shall revise the Emergency Plan to include the following:

(a) Incorporation of corrective actions identified in the staffing assessment required by this license condition; and

(b) Identification of how the augmented staff will be notified, given degraded communications capabilities.

- License Condition (13-11) – No later than eighteen (18) months before the latest date set forth in the schedule submitted in accordance with 10 CFR § 52.99(a) for completing the inspections, tests, and analyses in the ITAAC, the licensee shall have performed an assessment of on-site and off-site communications systems and equipment relied upon during an emergency event to ensure communications capabilities can be maintained during an extended loss of alternating current power. The communications capability assessment shall be performed in accordance with NEI 12-01, “Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities,” Revision 0.

  No later than one hundred eighty (180) days before the date scheduled for initial fuel load set forth in the notification submitted in accordance with 10 CFR § 52.103(a), the licensee shall have completed implementation of corrective actions identified in the communications capability assessment, including revisions to the Emergency Plan.

13.3.6 Conclusion

The staff reviewed the WLS COL application, including applicable portions of the referenced AP1000 DCD. The staff confirmed that the applicant addressed the required information relating to emergency planning and there is no additional information needed to support the WLS COLA. The results of the staff’s technical evaluation of the information incorporated by reference in the application are documented in NUREG-1793 and its supplements for the AP1000 DCD.

The EP-ITAAC are provided in Table 13.3-1, “WLS Units 1 & 2 ITAAC,” of this report. The staff concludes that, pursuant to 10 CFR 52.80(a), the applicant included in the WLS COL application the proposed inspections, tests, and analyses that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the license, the provisions of the Atomic Energy Act of 1954, as amended, and NRC rules and regulations.

As part of its review of the WLS COL application, FEMA provided its findings and determinations concerning the adequacy of offsite emergency planning, which are based on its review of State and local emergency plans. FEMA concluded that the offsite State and local emergency plans are adequate to cope with an incident at WLS, and there is reasonable assurance that these plans can be implemented. On the basis of its review of the FEMA findings and determinations, the staff concludes that the State and local emergency plans are adequate, and there is reasonable assurance that they can be implemented.

Based on its evaluation, the staff concludes that the onsite emergency plan establishes an adequate basis for an acceptable state of onsite emergency preparedness, and there is reasonable assurance that the plan can be implemented.

Based on FEMA's conclusions and the staff's evaluation the staff concludes that the emergency plans provide an adequate expression of the overall concept of operation and describe the essential elements of advanced planning and the provisions made to cope with emergency situations. Therefore, the staff concludes that the overall state of onsite and offsite emergency preparedness, when fully implemented, will meet the requirements of 10 CFR 50.33(g), 10 CFR 50.47, 10 CFR Part 50, Appendix E; 10 CFR 50.72(a)(3); 10 CFR 52.79(a)(21); 10 CFR 52.79(a)(22)(i); 10 CFR 52.80; and 10 CFR 100.21.

Further, in accordance with 10 CFR 50.47(a), the staff concludes that, subject to the required conditions and limitations of the full-power license, including the license condition listed in Section 13.3.5 of this report, there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency at the WLS, and that emergency preparedness at WLS Units 1 and 2 is adequate to support full-power operations.

Table 13.3-1 WLS Units 1 and 2 ITAAC

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
1.0 Emergency Classification System			
10 CFR 50.47(b)(4) – 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 facility licensees for determinations of minimum initial offsite response measures.	1.1 A standard emergency classification and emergency action level (EAL) scheme exists, and identifies facility system and effluent parameters constituting the bases for the classification scheme. [D.1**]	1.1 An inspection of the control room, technical support center (TSC), and emergency operations facility (EOF) will be performed to verify that they have displays for retrieving facility system and effluent parameters that constitute the bases for the classification scheme in the emergency plan implementing procedure addressing "Emergency Classification."	1.1.1 The specific parameters identified in the Emergency Action Level Thresholds in the emergency plan implementing procedure addressing "Emergency Classification" have been retrieved and displayed in the control room, TSC, and EOF.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
	[**D.1 corresponds to NUREG-0654 /FEMA-REP-1 evaluation criteria.] [**References in brackets throughout this table correspond to with NUREG-0654/FEMA-REP-1 Evaluation Criteria]		1.1.2 The ranges available in the control room, TSC, and EOF encompassed the values for the specific parameters identified in the Emergency Action Level Thresholds in the emergency plan implementing procedure addressing "Emergency Classification."
2.0 Notification Methods and Procedures			
10 CFR 50.47(b)(5) – Procedures have been established for notification, by the licensee, of State and local response organizations and for notification of emergency personnel by all organizations; the content of initial and follow-up messages to response organizations and the public has been established; and means to provide early notification and clear instruction to the populace within the plume exposure pathway Emergency	2.1 The means exist to notify responsible State and local organizations within 15 minutes after the licensee declares an emergency. [E.1]	2.1 A test will be performed of the capabilities.	2.1.1 A report exists that confirms communication has been established via the Selective Signaling Telephone System between the control room and the following: - Cherokee County Warning Point - York County Warning Point - Cleveland County Warning Point - South Carolina Warning Point - North Carolina Emergency

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
Planning Zone have been established.			Operations Center Radiological Warning Point
	2.2 The means exist to notify emergency response personnel. [E.2]	2.2 A test will be performed of the capabilities.	2.2 A report exists that confirms notification to the Lee Nuclear Station emergency response organization has been performed.
	2.3 The means exist to notify and provide instructions to the populace within the plume exposure pathway EPZ. [E.6]	NOTE: The means to notify and provide instructions to the populace within the plume exposure pathway EPZ is addressed by Acceptance Criteria 8.1.1.2.	
3.0 Emergency Communications			
10 CFR 50.47(b)(6) – Provisions exist for prompt communications among principal response organizations to emergency personnel and to the public.	3.1 The means exist for communications among the control room, TSC, EOF, principal State and local emergency operations centers (EOCs), and radiological field assessment teams. [F.1.d]	3.1 A test will be performed of the capabilities. NOTE: Additional ITAAC for the as-built TSC and OSC are addressed in Table 3.1-1 of Tier 1 of the AP1000 Design Control Document, Rev. 19.	3.1.1 A report exists that confirms communication has been established among the control room, OSC, and TSC. 3.1.2 A report exists that confirms communication have been established among the control room, TSC, and EOF. 3.1.3 A report exists that confirms communication via the Selective Signaling

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			Telephone System between the TSC and the following: - Cherokee County Warning Point - York County Warning Point - Cleveland County Warning Point - South Carolina Warning Point - North Carolina Emergency Operations Center Radiological Warning Point 3.1.4 A report exists that confirms communication has been established between the TSC, EOF, and radiological monitoring teams.
	3.2 The means exist for communications from the control room, TSC, and EOF to the NRC headquarters and regional office EOCs (including establishment of the Emergency Response Data System (ERDS) between the onsite computer system and the NRC Operations Center.) [F.1.f]	3.2 A test will be performed of the capabilities from the control room, TSC and EOF to the NRC, including ERDS.	3.2.1 A report exists that confirms communication has been established from the control room, TSC, and EOF to NRC Headquarters and Region II EOC. 3.2.2 A report exists that confirms ERDS data was provided from the plant computer system to

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			NRC Headquarters and Region II EOC.
4.0 Public Education and Information			
10 CFR 50.47(b)(7) – Information is made available to the public on a periodic basis on 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.	4.1 The licensee has provided space which may be used for a limited number of the news media at the EOF. [G.3.b]	4.1 An inspection of the Joint Information Center will be performed to verify that space is provided for a limited number of the news media.	4.1 The Joint Information Center has been located in the Duke Energy Center at 526 South Church Street, Charlotte, NC.
5.0 Emergency Facilities and Equipment			
10 CFR 50.47(b)(8) – Adequate emergency facilities and equipment to support the emergency response are provided and maintained.	5.1 The licensee has established a technical support center (TSC) and onsite operations support center (OSC). [H.1]	5.1 An inspection of the as-built TSC and OSC will be performed. NOTE: Additional ITAAC for the as-built TSC and OSC are addressed in Table	5.1.1 The TSC has been located in the Maintenance Building. 5.1.2 The TSC includes radiation monitors and a ventilation system with a high efficiency

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
		3.1-1 of Tier 1 of the AP1000 Design Control Document, Rev. 19.	particulate air (HEPA) and charcoal filter. 5.1.3 Back-up electrical power supply was available for the TSC. 5.1.4 The OSC was in a location separate from the control room.
	5.2 The licensee has established an Emergency Operations Facility (EOF). [H.2]	5.2 An inspection of the EOF will be performed.	5.2.1 The EOF had at least 243 square meters (2,625 square feet). 5.2.2 Voice transmission and reception have been accomplished between the EOF and TSC. 5.2.3 A report exists that confirms voice transmission and reception have been accomplished via the Selective Signaling Telephone System between the EOF and the following: - Cherokee County Warning Point

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			- York County Warning Point - Cleveland County Warning Point - South Carolina Warning Point - North Carolina Emergency Operations Center Radiological Warning Point
6.0 Accident Assessment			
10 CFR 50.47(b)(9) – Adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use.	6.1 The means exist to provide initial and continuing radiological assessment throughout the course of an accident. [I.2]	6.1 A test of the emergency plan will be conducted by performing an exercise or drill to verify the capability to perform accident assessment.	6.1 A report exists that confirms an exercise or drill has been accomplished including use of selected monitoring parameters identified in the EAL Thresholds in the emergency plan implementing procedure addressing “Emergency Classification,” to assess simulated degraded plant conditions and initiate protective actions in accordance with the following criteria: A. Accident Assessment and Classification 1. Initiating conditions identified, EALs parameters determined, and the emergency correctly

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			classified throughout the drill. B. Radiological Assessment and Control 1. Onsite radiological surveys performed and samples collected. 2. Radiation exposure to emergency workers monitored and controlled. 3. Field monitoring teams assembled and deployed. 4. Field team data collected and disseminated. 5. Dose projections developed. 6. The decision whether to issue radioprotective drugs to Duke emergency workers made. 7. Protective action recommendations developed and communicated to appropriate authorities.
	6.2 The means exist to determine the source term of releases of radioactive material within plant systems, and the	6.2 An analysis of emergency plan implementing procedures will be performed.	6.2 A methodology has been established to determine source term of releases of

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
	magnitude of the release of radioactive materials based on plant system parameters and effluent monitors. [I.3]		radioactive materials within plant systems.
	6.3 The means exist to continuously assess the impact of the release of radioactive materials to the environment, accounting for the relationship between effluent monitor readings, and onsite and offsite exposures and contamination for various meteorological conditions. [I.4]	6.3 An analysis of emergency plan implementing procedures will be performed.	6.3 A methodology has been provided to establish the relationship between effluent monitor readings and onsite and offsite exposures and contamination for various meteorological conditions.
	6.4 The means exist to acquire and evaluate meteorological information. [I.5]	6.4 An inspection of the control room, TSC, and EOF will be performed to verify the availability of the following meteorological data is available: - Wind speed (at 10 m and 60 m) - Wind direction (at 10 m and 60 m) - Air temperature (at 10 m and 60 m)	6.4 The specified meteorological data was available at the control room, TSC, and EOF.
	6.5 The means exist to make rapid assessments of actual or potential magnitude and locations of any radiological hazards	6.5 An analysis of emergency plan implementing procedures will be performed.	6.5 A methodology has been established to provide rapid assessment of the actual or potential magnitude and

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
	through liquid or gaseous release pathways, including activation, notification means, field team composition, transportation, communication, monitoring equipment, and estimated deployment times. [I.8]		locations of any radiological hazards through liquid or gaseous release pathways.
	6.6 The capability exists to detect and measure radioiodine concentrations in air in the plume exposure pathway EPZ, as low as 10E-7 μCi/cc (microcuries per cubic centimeter) under field conditions. [I.9]	6.6 A test of Duke field survey instrumentation will be performed to verify the capability to detect airborne concentrations as low as 10E-07 microcuries per cubic centimeter.	6.6 A report exists that confirms instrumentation used for monitoring I-131 to detect airborne concentrations as low as 10E-07 microcuries per cubic centimeter has been provided.
	6.7 The means exist to estimate integrated dose from the projected and actual dose rates, and for comparing these estimates with the EPA protective action guides (PAGs). [I.10]	6.7 An analysis of emergency plan implementing procedures will be performed to verify that a methodology is provided to establish means for relating contamination levels and airborne radioactivity levels to dose rates and gross radioactivity measurements for the following isotopes – Kr-88, Ru-106, I-131, I-132, I-133, I-134, I-135, Te-132, Xe-133,	6.7 The means for relating contamination levels and airborne radioactivity levels to dose rates and gross radioactivity measurements for the specified isotopes has been established.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
		Xe-135, Cs-134, Cs-137, Ce-144.	
7.0 Protective Response			
10 CFR 50.47(b)(10) – A range of protective actions has been developed for the plume exposure pathway EPZ for emergency workers and the public. In developing this range of actions, consideration has been given to evacuation, sheltering, and, as a supplement to these, the prophylactic use of potassium iodide (KI), as appropriate. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are developed and in place, and protective actions for the ingestion exposure pathway EPZ appropriate to the locale have been developed.	7.1 The means exist to warn and advise onsite individuals of an emergency, including those in areas controlled by the operator, including: [J.1] a. Employees not having emergency assignments b. Visitors c. Contractor and construction personnel d. Other people who may be in the public access areas, on or passing through the site, or within the owner controlled area.	7.1 A test of the onsite warning and communication capability will be performed during a drill or exercise.	7.1.1 A report exists that confirms that, during a drill or exercise, notification and instructions were provided to onsite workers and visitors, within the Protected Area, over the plant public announcement system. 7.1.2 A report exists that confirms that, during a drill or exercise, audible warnings were provided to individuals outside the Protected Area, but within the Owner Controlled Area.
8.0 Exercises and Drills			
10 CFR 50.47(b)(14) – Periodic exercises are (will be) conducted to evaluate major portions of emergency response capabilities, periodic	8.1 Licensee conducts a full-participation exercise to evaluate major portions of emergency response capabilities, which includes participation	8.1 A full participation exercise (test) will be conducted within the specified time periods of Appendix E to 10 CFR Part 50.	8.1.1.1 A report exists that confirms an exercise was conducted within the specified time periods of Appendix E to 10 CFR Part 50, onsite

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
drills are (will be) conducted to develop and maintain key skills, and deficiencies identified as a result of exercises or drills are (will be) corrected.	by the State and local agency within the plume exposure pathway EPZ, and the State within the ingestion control EPZ. [N.1]		exercise objectives listed below were met, and there are no uncorrected onsite exercise deficiencies.
			8.1.1.2 A report exists that confirms the following exercise objectives were satisfied by meeting the specified performance criteria: A. Accident Assessment and Classification 1. Demonstrate the ability to identify initiating conditions, determine emergency action level (EAL) parameters, and correctly classify the emergency throughout the exercise. Performance Criterion: a. Determine the correct emergency classification level based on events which were in progress, considering past events and their impact on the current conditions, within 15 minutes from the time the initiating condition(s) or EAL is identified.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			B. Notifications 1. Demonstrate the ability to alert, notify and mobilize site emergency response personnel. Performance Criteria: a. Complete the designated actions in accordance with emergency plan implementing procedures and perform the announcement within 15 minutes of the initial event classification for an Alert or higher. b. Mobilize site emergency responders in accordance with emergency plan implementing procedures within 15 minutes of the initial event classification for an Alert or higher. 2. Demonstrate the ability to notify responsible State, local government agencies within 15 minutes and the NRC within 60 minutes after declaring an emergency.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			Performance Criteria: a. Transmit information in accordance with approved emergency plan implementing procedures within 15 minutes of event classification. b. Transmit information in accordance with approved emergency plan implementing procedures, within 60 minutes of last transmittal for a follow-up notification to State and local authorities. 3. Demonstrate the ability to warn or advise onsite individuals of emergency conditions. Performance Criterion: a. Initiate notification of onsite individuals within 15 minutes of declaration. 4. Demonstrate the capability of the Public Alert and Notification System to operate properly when required. Performance Criteria:

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			a. 90% of the sirens operate properly, as indicated by the siren feedback system. b. The EAS is activated.
			C. Emergency Response 1. Demonstrate the capability to direct and control emergency operations. Performance Criterion: a. Command and control is demonstrated in the control room in the early phase of the emergency, and the technical support center (TSC) within 75 minutes of declaration minutes of an Alert or higher emergency classification. 2. Demonstrate the ability to transfer emergency direction from the control room to the TSC upon activation. Performance Criteria: a. Turnover briefings are conducted in accordance with emergency plan

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			implementing procedures. b. Documentation of transfer of duties is completed in accordance with emergency plan implementing procedures. 3. Demonstrate the ability to prepare for around-the-clock staffing requirements. Performance Criterion: a. Complete 24-hour staff assignments. 4. Demonstrate the ability to perform assembly and accountability within 30 minutes of an emergency requiring protected area assembly and accountability. Performance Criterion: a. Protected area (PA) personnel assembly and accountability completed within 30 minutes of an emergency requiring PA assembly and accountability.
			D. Emergency Response Facilities

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			1. Demonstrate activation of the operational support center (OSC), and full functional operation of the TSC and EOF within 75 minutes declaration of Alert or higher emergency classification. Performance Criterion: a. The TSC, OSC, and EOF are activated within 75 minutes of the initial notification. 2. Demonstrate the adequacy of equipment, security provisions, and habitability precautions for the TSC, OSC, and EOF as appropriate. Performance Criteria: a. Demonstrate the adequacy of the emergency equipment in the emergency response facilities as specified in emergency plan implementing procedures. b. The Security Force implements and follows applicable emergency plan implementing procedures.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			c. The Radiological Assessment Manager implements habitability controls in accordance with emergency plan implementing procedures if an onsite/offsite release has occurred. 3. Demonstrate the adequacy of communication for all emergency support resources. Performance Criteria: a. Emergency response facility personnel are able to operate communication systems in accordance with emergency plan implementing procedures. b. Emergency response communication systems listed in emergency plan implementing procedures are available and operational for the duration of the exercise.
			E. Radiological Assessment and Control

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			1. Demonstrate the ability to obtain onsite radiological surveys and samples. Performance Criteria: a. Radiation Protection Technicians demonstrate the ability to obtain appropriate instruments (range and type) and perform surveys. b. Airborne samples are taken in accordance with emergency plan implementing procedures. 2. Demonstrate the ability to continuously monitor and control radiation exposure to emergency workers. Performance Criteria: a. Emergency workers are issued self-reading dosimeters when radiation levels require, and exposures are controlled to 10 CFR Part 20 limits (unless the Emergency Coordinator authorizes emergency limits).

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			b. Exposure records are available. c. Emergency workers include Security and personnel within all emergency facilities. 3. Demonstrate the ability to assemble and deploy field monitoring teams within 75 minutes from the decision to do so. Performance Criterion: a. One Field Monitoring team is ready to be deployed within 15 – 30 minutes of their arrival onsite. In addition, an offsite monitoring team must be able to be dispatched within 75 minutes of an Alert or higher emergency classification. 4. Demonstrate the ability to collect and disseminate field team data. Performance Criteria: a. Field team collects data for dose rate and airborne radioactivity levels in accordance with emergency plan implementing procedures.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			b. Field team communicates data to the TSC or EOF in accordance with emergency plan implementing procedures. 5. Demonstrate the ability to develop dose projections. Performance Criterion: a. Timely and accurate dose projections are performed in accordance with emergency plan implementing procedures. 6. Demonstrate the ability to make the decision whether to issue radioprotective drugs (KI) to onsite emergency workers. Performance Criterion: a. KI is issued (simulated) if the estimated dose to the thyroid will exceed 25 rem committed dose equivalent (CDE). 7. Demonstrate the ability to develop appropriate protective action recommendations (PARs) and notify

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			appropriate authorities within 15 minutes after development. Performance Criteria: a. Total effective dose equivalent (TEDE) and CDE dose protections from the dose assessment computer code are compared, in accordance with emergency plan implementing procedures. b. PARs are developed within 15 minutes of data availability. c. PARs are transmitted to responsible State and local government agencies via voice or fax within 15 minutes of event classification or PAR development.
			F. Public Information 1. Demonstrate the capability to develop and disseminate clear, accurate, and timely information to the news media in accordance with emergency plan implementing procedures.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			Performance Criterion: a. The Joint Information Center (JIC) is activated within 75 minutes following the declaration of a Site Area Emergency or higher classification or following the Emergency Coordinator’s or JIC Director’s instruction to do so. 2. Demonstrate the capability to establish and effectively operate rumor control in a coordinated fashion. Performance Criteria: a. Calls are answered in a timely manner with the correct information, in accordance with emergency plan implementing procedures. b. Calls are returned or forwarded, as appropriate, to demonstrate responsiveness. c. Rumors are identified and addressed in accordance with emergency plan

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			implementing procedures.
			G. Evaluation 1. Demonstrate the ability to conduct a post-exercise critique, to determine areas requiring improvement and corrective action. Performance Criteria: a. An exercise time line is developed, followed by an evaluation of the objectives. b. Significant problems in achieving the objectives are discussed to ensure understanding of why objectives were not fully achieved. c. Recommendations for improvement in non-objective areas are discussed.
			8.1.2.1 A report exists that confirms onsite emergency response personnel were mobilized to fill emergency response positions and there were no uncorrected onsite exercise deficiencies. 8.1.2.2 A report exists that confirms onsite emergency response

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			personnel performed their assigned responsibilities as provided in Section II.B of the WLS Combined License Application Emergency Plan and there were no uncorrected onsite exercise deficiencies.
			8.1.3.1 The exercise is completed within the specified time periods of Appendix E to 10 CFR Part 50, offsite exercise objectives have been met, and there are either no uncorrected offsite deficiencies, or a license condition requires offsite deficiencies to be corrected prior to operation above 5% rated power.
9.0 Assignment of Responsibility – Organizational Control			
10 CFR 50.47(b)(1) – 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	9.1 The staff exists to provide 24-hour per day emergency response and manning of communications links, including continuous operations for a protracted period. [A.1.e.A.4**]	9.1 An inspection of the emergency plan implementing procedures will be performed.	9.1 Emergency plan implementing procedures provide for 24-hour per day emergency response staffing and manning of communication links, including continuous operations for a protracted period.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
established, and each principle response organization has staff to respond and to augment its initial response on a continuous basis.			
10.0 Onsite Emergency Organization			
10 CFR 50.47(b)(2) – 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.	10.1 The staff exists to provide minimum and augmented on-shift staffing levels, consistent with Table B-1 of NUREG-0654/FEMA-REP-1, Rev. 1. [B.5, B.7]	10.1 An inspection of the emergency plan implementing procedures will be performed.	10.1 Emergency plan implementing procedures provide minimum and augmented on-shift staffing levels, consistent with Table II-2 of the WLS Combined License (COL) Application Emergency Plan.

13.4 Operational Programs (Related to RG 1.206, Section C.III.1, Chapter 13, C.I.13.4, "Operational Program Implementation")

13.4.1 Introduction

In SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," October 28, 2005, the staff detailed its plan for reviewing operational programs in a COL application. The Commission approved the staff's plan in the related February 22, 2006, SRM, Although numerous programs support the operation of a nuclear power plant, SECY-05-0197 focused on those programs that meet the following three criteria:

1. required by regulation
2. reviewed in a COL application
3. inspected to verify program implementation as described in the FSAR

The programs that meet the above criteria are collectively referred to as “operational programs” and most are identified in SECY-05-0197.

13.4.2 Summary of Application

WLS COL FSAR, Revision 11, Section 13.4 incorporates by reference AP1000 DCD, Revision 19, Section 13.4.

In addition, in WLS COL FSAR Section 13.4 and in WLS COLA Part 10, “Proposed License Conditions and ITAAC,” the applicant provided the following:

AP1000 COL Information Item

- STD COL 13.4-1

The applicant provided additional information in STD COL 13.4-1 to address COL Information Item 13.4-1 and COL Action Item 13.4-1, identified in NUREG-1793, Appendix F and its supplements. This item states that COL applicants referencing the AP1000 certified design will address each operational program.

License Conditions

- Part 10, License Condition 3, “Operational Program Implementation”
- Part 10, License Condition 6, “Operational Program Readiness”

Both license conditions are related to STD COL 13.4-1. License Condition 3 addresses implementation milestones for those operational programs whose implementation is not addressed in the regulations. License Condition 6 includes the timing of information related to operational programs to support NRC inspection activities.

13.4.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the regulatory basis for acceptance of the supplementary information presented in this application is identified in the individual chapters of this report that address the evaluations of the specific operational programs, which are itemized in the next section, as clarified by the regulatory guidance in SECY-05-0197 and RG 1.206.

13.4.4 Technical Evaluation

The staff reviewed WLS COL FSAR Section 13.4 and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to operational programs. The results of the staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the DC and use this review to evaluate subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the Vogtle (VEGP) COL FSAR, Revision 5 to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from responses to RAIs.
- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.
- The staff verified that the site-specific differences were not relevant.

The staff completed its review and concluded that the evaluation performed for the standard content is directly applicable to the WLS COL application. This standard content material is identified in this report by use of italicized, double-indented formatting. Section 1.2.3 of this report provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the Bellefonte (BLN) Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.4.4:

> *Although the staff concluded that the evaluation performed for the standard content is directly applicable to the VEGP COL application, there were differences in the response provided by the VEGP applicant from that provided by the BLN applicant regarding the standard content material. These differences affect the two license conditions and the table listing the operational programs. These differences are evaluated by the staff below, following the standard content material.*
>
> *<u>AP1000 COL Information Item</u>*
>
> - *STD COL 13.4-1*
>
> *The applicant provided supplemental information by adding the following statement to Section 13.4 of the VEGP COL FSAR:*

Operational programs are specific programs that are required by regulations. Table 13.4-201 lists each operational program, the regulatory source for the program, the section of the FSAR in which the operational program is described, and the associated implementation milestone(s).

Each operational program is evaluated by the staff in the applicable SER chapters.

License Conditions

- *License Condition 3, "Operational Program Implementation"*

- *License Condition 6, "Operational Program Readiness"*

These two proposed license conditions are evaluated by the NRC staff as part of its evaluation of each of the operational programs in the applicable SER chapters.

The following portion of this technical evaluation section provides the staff's general evaluation of the operational programs and associated license conditions and is reproduced from Section 13.4.4 of the BLN SER:

The NRC staff's review of the acceptability of the supplemental information added by STD COL 13.4-1 and the proposed license conditions is based on four considerations. The first consideration is the acceptability of the individual operational programs, including the implementation of the different phases of these operational programs. The second consideration is whether the applicant correctly identified those operational programs whose implementation requirements are not addressed in the regulations, and, therefore, need to be included in License Condition 3. The third consideration is whether the applicant correctly specified in License Condition 6 the timing of information related to operational programs to support NRC inspection activities. The fourth consideration is whether the list of operational programs in BLN COL FSAR Table 13.4-201 is complete.

In regard to the first consideration, the SER sections referenced in the above table address the NRC staff's regulatory evaluation of the individual operational programs. For each of these operational programs, the staff has either concluded that the applicant has satisfied the applicable regulatory guidance (including the implementation requirements when specified in the regulations), or the staff's review is still ongoing. For those operational program reviews that are ongoing, the staff's final conclusions will be provided in the SER sections referenced in the above table at a later date.

In regard to the second consideration, the NRC staff verified that those operational programs, whose implementation requirements are not specified in the regulations, are captured in License Condition 3.

In regard to the third consideration, the NRC staff compared License Condition 6 to the recommended license condition in SECY-05-0197 related to the timing of *information to support NRC inspection activities of operational programs. The staff finds that the applicant used language similar to the recommended license condition specified in SECY-05-0197 to develop License Condition 6. It should be noted that License Condition 6 addresses additional scheduler requirements (Sections b. through d.) that are not related to the operational programs evaluated in this section of the SER, and, therefore, are not evaluated in this SER section.*

In regard to the fourth consideration, the NRC staff compared the operational programs provided by the applicant in BLN COL FSAR Table 13.4-201 (included in the above table) to the operational programs specified in SECY-05-0197. The staff finds that the applicant has included all the operational programs specified in SECY-05-0197, including the two operational programs (Motor-Operated Valve Testing Program and the Safeguards Contingency Program) added by the NRC to the list of operational programs provided by the NEI in its letter dated August 31, 2005.

There are differences between BLN COL FSAR Table 13.4-201 and the table of operational programs in SECY-05-0197 with respect to implementation milestone information. The first difference is the SECY paper states that there are no required implementation milestones in the regulations for the Maintenance Rule Program and the Quality Assurance Program (Operation), while BLN COL FSAR Table 13.4-201 references regulations that require implementation milestones for these two programs. The staff has reviewed the regulation references provided by the applicant and concludes that they do provide appropriate requirements for implementation milestones. Further support for this conclusion is the regulatory guidance in Section C.I.13.4 of RG 1.206. The example table located in this section of the RG references the same implementation regulatory guidance for the Maintenance Rule Program and the Quality Assurance Program (Operation) as does BLN COL FSAR Table 13.4-201.

The second difference is that the SECY paper states that 10 CFR Part 50, Appendix J, specifies implementation requirements for the Containment Leakage Rate Testing Program, while BLN COL FSAR Table 13.4-201 states that the implementation milestones for this program will be controlled by a license condition. The staff has reviewed the implementation milestone proposed in License Condition 3 for the Containment Leakage Rate Testing Program, and finds that it is more stringent than the regulatory guidance in Appendix J. Therefore, the staff finds this difference to be acceptable.

The applicant added an operational program to BLN COL FSAR Table 13.4-201, the Initial Test Program, which is not in the list of operational programs specified in SECY-05-0197. The option of adding operational programs to this list is specifically allowed by SECY-05-0197. Further support for the acceptability of adding the Initial Test Program is that the example table located in Section C.I.13.4 of RG 1.206 also lists this operational program.

Therefore, the NRC staff concludes that the additional information (STD COL 13.4-1) provided by the applicant in BLN COL FSAR Section 13.4, in *conjunction with the conditions specified in BLN COL FSAR, Part 10, License Conditions 3 and 6, complies with the applicable regulatory guidance provided in SECY-05-0197.*

Evaluation of Site-specific Response to Standard Content

The staff notes that the VEGP applicant separated the fitness-for-duty (FFD) program from the overall security program and added a new operational program, Cyber Security, to the list of operational programs in FSAR Table 13.4-201. The implementation requirements for these additional operational programs comply with the considerations identified above in the standard content material, and are, therefore, acceptable. In addition, the VEGP applicant also made minor changes to operational program implementation details in License Condition 3 and also modified Sections a. through d. associated with License Condition 6. The changes to these two license conditions are evaluated by the staff in the applicable SER chapters and do not affect the evaluation of operational programs covered in this section of the SER. Therefore, the conclusions reached by the NRC staff related to STD COL 13.4-1 are directly applicable to the VEGP COL application.

The BLN SER text refers to an SER table listing operational programs. This table was not reproduced for the VEGP SER since it duplicates the information in VEGP COL FSAR Table 13.4-201.

The staff also notes that the applicant added the operational program, SNM Material Control and Accounting Program, to the list of operational programs in WLS COL FSAR Table 13.4-201. The implementation requirements for this additional operational program comply with the considerations identified above in the standard content material, and is therefore acceptable.

13.4.5 Post Combined License Activities

The license conditions for each of the operational programs are discussed in the applicable SER chapters. Therefore, there are no post-COL activities related to this section.

13.4.6 Conclusion

The staff reviewed the WLS COL application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to operational programs, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff concludes that the relevant information presented in the WLS COL FSAR is acceptable based on the regulatory guidance in SECY-05-0197, in conjunction with the applicable regulations specified in the individual sections of this report that evaluated each of the operational programs discussed above. The staff based its conclusion on the following:

- STD COL 13.4-1, as related to operational programs, is acceptable because each of the operational programs in WLS COL FSAR Table 13.4-201 has been found acceptable by the staff in other sections of this report, as noted in Section 13.4.4 above. In addition, the guidance in SECY-05-0197 and RG 1.206 was used to verify that the applicant's list of operational programs is complete.

13.5 Plant Procedures

13.5.1 Introduction

Descriptions of the administrative and operating procedures that the applicant uses to ensure routine operating, off-normal, and emergency activities are conducted in a safe manner are provided. In its plant procedures, the applicant provided a brief description of the nature and content of the procedures and a schedule for the preparation of appropriate written administrative and operating procedures. The applicant delineated in the description of the procedures the functional position for procedural revision and approval prior to implementation. Inspection of procedures will occur as part of the construction inspection program.

13.5.2 Summary of Application

WLS COL FSAR, Revision 11, Section 13.5, incorporates by reference AP1000 DCD, Revision 19, Section 13.5.

In addition, in WLS COL FSAR Section 13.5, the applicant provided the following:

AP1000 COL Information Item

- STD COL 13.5-1

The applicant provided additional information in STD COL 13.5-1 to resolve COL Information Item 13.5-1 (COL Action Item 13.5-1), which addresses plant procedures.

- WLS COL 13.5-1

The applicant provided additional information in WLS COL FSAR Section 13.5.2.2.5 related to procedures to control radionuclide inventories and personnel doses in the Radwaste Building. This information, as well as related additional WLS COL FSAR information in WLS COL 11.2-1 and proposed License Condition 13 in Part 10 of the WLS COL application. This information is reviewed in Section 11.2 of this report.

The applicant provided additional information in WLS COL FSAR Section 13.5.1 related to the process for implementing the safety/security interface requirements of 10 CFR 73.58. This information is reviewed in Section 13.4.1.17 of this report.

The applicant provided additional information in WLS COL FSAR Section 13.5.2.2.8 related to security procedures provided in the Security Plan related to Special Nuclear Material (SNM) Physical Protection Program. This information is reviewed in Section 1.5.5 of this report.

13.5.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of NRC regulations for plant procedures are given in NUREG-0800. Sections 13.5.1.1 and 13.5.2.1.

The applicable regulations and regulatory guidance are as follows:

- 10 CFR 50.34(a), “Preliminary safety analysis report”
- 10 CFR 50.34(b)
- RG 1.33, “Quality Assurance Program Requirements (Operation)”

13.5.4 Technical Evaluation

The staff reviewed WLS COL FSAR Section 13.5 and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The staff’s review confirmed that the information in the application and incorporated by reference addresses the required information relating to plant procedures. The results of the staff’s evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the DC and use this review to evaluate subsequent COL applications. To ensure that the staff’s findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from responses to RAIs.
- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.
- The staff verified that the site-specific differences were not relevant.

The staff completed its review and concluded that the evaluation performed for the standard content is directly applicable to the VEGP COL application. This standard content material is identified in this report by use of italicized, double-indented formatting. Section 1.2.3 of this report provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the BLN Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.5.4:

<u>AP1000 COL Information Item</u>

- *STD COL 13.5-1, addressing plant procedures*

The applicant provided the following additional information to resolve COL Information Item 13.5-1, which addresses the plant procedures of the COL applicant. COL Information Item 13.5-1 states:

> *Combined License applicants referencing the AP1000 certified design will address plant procedures including the following:*
>
> - *Normal operation*
> - *Abnormal operation*
> - *Emergency operation*
> - *Refueling and outage planning*
> - *Alarm response*
> - *Maintenance, inspection, test and surveillance*
> - *Administrative*
> - *Operation of post-72 hour equipment*

The commitment was also captured as COL Action Item 13.5-1 in Appendix F of the staff's FSER for the AP1000 DCD (NUREG-1793).

The applicant provided additional text in BLN COL FSAR Section 13.5 to describe the administrative, operating and maintenance procedures that the operating organizational staff uses to conduct routine operating, abnormal, and emergency activities in a safe manner.

In BLN COL FSAR Section 13.5, the applicant described the different classifications of procedures that the operators will use, including normal, abnormal, emergency, refueling and outage, and alarm response procedures. The staff finds this information acceptable because it meets the criteria in NUREG-0800, Chapter 13.5.2.1.

In BLN COL FSAR Section 13.5, the applicant stated that the format and content of procedures are controlled by the applicable AP1000 writer's guideline. The DCD, Section 13.5.1, describes a referenced document, APP-GW-GLR-040, "Plant Operations Maintenance and Surveillance Procedures," dated August 23, 2007, which includes the AP1000 writer's guidelines. The staff finds this acceptable because the applicant-provided procedure format and content are consistent with the guidance in NUREG-0800, Section 13.5.2.1.

In BLN COL FSAR Section 13.5.1, the applicant describes the nature and content of administrative procedures for both Category (A) - Controls, and Category (B) - Specific Procedures. The staff finds this acceptable because the listed procedures are consistent with the guidance in NUREG-0800, Section 13.5.1.1.

In BLN COL FSAR Section 13.5.2, the applicant stated that EP procedures are discussed in the Emergency Plan and that security procedures are discussed in the Security Plan. The evaluation of EP procedures may be found in Section 13.3 of this SER. The evaluation of security procedures is found in Section 13.6 of this SER.

In BLN COL FSAR Section 13.5.2, the applicant stated the Quality Assurance Program description (QAPD) provides a description of procedural requirements for maintenance, instrument calibration and testing, inspection, and material control. The evaluation of QAPD procedures is found in Section 17.5 of this SER.

In BLN COL FSAR, Section 13.5.2.1, the applicant stated that information related to EOPs is addressed in the DCD. The DCD, Section 13.5.1, describes the program for developing and implementing EOPs and the required content of EOPs procedures in the referenced document, APP-GW-GLR-040. In addition, this information clarifies the procedure development program (PDP) as described in the procedures generation package (PGP) for EOPs, provides a description of the EOP [emergency operating plan] *verification and validation* (V&V) *program, and describes the program for training operators on EOPs, including an explanation of how the recommendations of TMI Action Plan, Item I.C.1, will be met. The staff finds the program for developing and implementing EOPs acceptable because it meets the criteria in NUREG-0800, Section 13.5.2.1.*

<u>*Evaluation of Plant Procedure Issues Not Address*[ed] *in the Standard Content Evaluation*</u>

In VEGP COL FSAR Table 1.9-202, "Conformance with SRP Acceptance Criteria," the applicant identified two exceptions to the criteria of NUREG-0800, Section 13.5, which recommend[s] *providing a schedule for procedure development in the FSAR, and including a description of procedures to be used by operators in the FSAR. The staff notes that the BLN COL FSAR Table 1.9-202 includes these same two exceptions to the criteria of Section 13.5 of NUREG-0800. The guidance of NUREG-0800, Section 13.5.2.1, states that while the submittal should describe the different classifications of procedures that operators will use, it is not necessary that each applicant's procedures conform precisely. In addition, the procedures, regardless of title or classification, are to be available to accomplish the functions identified in RG 1.33. NUREG-0800 makes allowance for "general areas." The staff finds the two exceptions to the criteria of NUREG-0800, Section 13.5, to be acceptable because the applicant's procedure classification follows the guidance in NUREG-0800, Section 13.5.*

> *In RAI [request for additional information] 13.6-36, the staff requested the VEGP applicant address the requirements of 10 CFR 73.58, "Safety/security requirements for nuclear power plants." In its response dated May 14, 2010, the applicant stated that management controls and processes used to establish and maintain an effective interface between nuclear safety and physical security are addressed by administrative controls. The VEGP applicant committed to revise FSAR Section 13.5.1 to include the safety/security interface implementation process in the list of procedural instructions provided in plant administrative procedures. The NRC staff's review of this safety/security procedural issue, which includes tracking the incorporation of the relevant material into the VEGP COL application, is addressed in Section 13.6.4.1.17 of this SER.*

<u>Supplemental Information</u>

- WLS COL 13.5-1

The applicant provided additional information in WLS COL FSAR Section 13.5.2.2.5 related to procedures to control radionuclide inventories and personnel doses in the Radwaste Building. This information, as well as related additional WLS COL FSAR information in Levy Nuclear Plant (LNP) COL 11.2-1 and proposed License Condition 13 in Part 10 of the WLS COL application. This information is reviewed in Section 11.2 of this report.

The applicant provided additional information in WLS COL FSAR Section 13.5.1 related to the process for implementing the safety/security interface requirements of 10 CFR 73.58. This information is reviewed in Section 13.6.4.1.17 of this report.

The applicant provided additional information in WLS COL FSAR Section 13.5.2.2.8 related to security procedures provided in the Security Plan related to Special Nuclear Material (SNM) Physical Protection Program. This information is reviewed in Section 1.5.5 of this report.

13.5.5 Post Combined License Activities

There are no post COL activities related to this section.

13.5.6 Conclusion

The staff reviewed the WLS COL application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to plant procedures, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the recommendations of NUREG-0800, Sections 13.5.1.1 and 13.5.2.1. The staff based its conclusion on the following:

- STD COL 13.5-1, as related to plant procedures, is acceptable because it describes the procedures used by the applicant's operating organizational staff to conduct routine administrative, operating, abnormal, and emergency activities in a safe manner, in accordance with the regulatory guidance in NUREG-0800, Sections 13.5.1.1 and 13.5.2.1.

- In WLS COL FSAR Table 1.9-202, the applicant identified two exceptions to the criteria of NUREG-0800, Section 13.5, related to providing WLS COL FSAR descriptions of, and a development schedule for, procedures to be used by operators. The guidance of NUREG-0800, Section 13.5.2.1, makes allowances for "general areas," stating that while the FSAR submittal should describe the different classifications of procedures used by operators, it is not expected that each applicant's procedures conform precisely. The staff finds the two exceptions acceptable because the applicant's procedure classification follows the guidance in NUREG-0800, Section 13.5.

13.6 Physical Security

13.6.1 Introduction

The COL application for the WLS COL Units 1 and 2 describes the WLS COL applicant's physical protection program, which is intended to meet NRC regulations for the use of the design basis threat (DBT) to design safeguards systems to protect against acts of radiological sabotage as stated in 10 CFR 73.1, "Purpose and Scope." The overall purpose of the applicant's physical protection program is 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.

The physical protection program includes the design of a physical protection system that ensures the capabilities to detect, assess, interdict, and neutralize threats of radiological sabotage are maintained at all times. The applicant incorporates by reference the standard AP1000 design that includes design of physical protection systems within the design of the vital island and vital structures, as described in the Westinghouse Electric Company (Westinghouse) DC document for the AP1000 standard design Tier 1 and Tier 2 information, including Technical Report (TR)-49, "AP1000 Enhancement Report, TR-94, "AP1000 Safeguards Assessment Report," and TR-96, "Interim Compensatory Measures Report." Part 8 of the WLS COL application consists of the WLS Units 1 and 2 Physical Security Plan (PSP), Training and Qualification Plan (T&QP), and Safeguards Contingency Plan (SCP). WLS COL FSAR Section 13.6 describes the physical protection program and the physical protection system that are not addressed within the scope of the standard AP1000 design for meeting NRC performance and prescriptive requirements for physical protection stated in 10 CFR Part 73, "Physical Protection of Plants and Material." Due to security restraints, the staff's evaluation of the physical security protection program presented in this publicly-available report does not include the same level of detail as the safeguards information version. Those persons with the correct access authorization and need-to-know may view the safeguards information version of the WLS COL application Section 13.6 SER, which is located in the NRC Secure Local Area Network.

13.6.2 Summary of Application

WLS COL FSAR, Revision 11, Section 13.6, incorporates by reference AP1000 DCD, Revision 19, Section 13.6.

- WLS COLA Part 8 – Safeguards/Security Plans

In a December 12, 2007, letter, the applicant submitted a PSP to the NRC as part of the WLS COL application for proposed WLS Units 1 and 2. In a May 12, 2009, letter, the applicant submitted Revision 1 to the PSP. In a November 17, 2011, letter, the applicant submitted PSP, Revision 2. In an April 10, 2013, letter the applicant submitted PSP, Revision 3.

In addition, in WLS COL FSAR Section 13.6, the applicant provided the following:

AP1000 COL Information Items

- STD COL 13.6-1

The applicant provided additional information in STD COL 13.6-1 to address COL Information Item 13.6-1, which provides information related to the security plan. The security plan consists of three parts, the PSP, T&QP, and SCP.

- STD COL 13.6-5

The applicant provided additional information in STD COL 13.6-5 to address COL Information Item 13.6-5, which provides information related to the cyber security program. This COL item is evaluated in Section 13.8 of this report.

License Conditions

- Part 10, License Condition 3 D.3, and G.9

The applicant proposed a license condition in WLS COLA Part 10, which provides the milestones to implement applicable portions of the Security Program:

- Part 10, License Condition 5

The applicant proposed a license condition in WLS COLA Part 10, which proposed the maintenance of the PSP, T&QP, and the SCP when nuclear fuel is onsite (protected area), and continuing until all nuclear fuel is permanently removed from the site.

- Part 10, License Condition 6

The applicant proposed a license condition to provide a schedule to support NRC inspection of operational programs including the PSP, T&QP, and the SCP.

13.6.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793, and its supplements. In addition, the relevant requirements of NRC regulations for the physical security, and the associated acceptance criteria, are summarized in NUREG-0800, Section 13.6.1.

The applicable regulatory requirements for physical protection are as follows:

- The provisions of 10 CFR 52.79(a)(35)(i) and (ii) require that information submitted for a COL describe how the applicant will meet the requirements of 10 CFR Part 73, “Physical Protection of Plants and Material”; and provide a description of the implementation of the PSP. The provisions of 10 CFR 52.79(a)(36)(i) through (iv) require that the application include an SCP in accordance with the criteria set forth in 10 CFR Part 73, Appendix C, “Nuclear Power Plant Safeguards Contingency Plans,” and a T&QP in accordance with 10 CFR Part 73, Appendix B, “General Criteria for Security Personnel.” The provisions also require that the applicant provide a description of the implementation of the SCP and the T&QP and that the applicant protect the PSP, T&QP, and SCP in accordance with the requirements of 10 CFR 73.21, “Protection of Safeguards Information: Performance Requirements” and 10 CFR 73.22, “Protection of Safeguards Information: Specific requirements.”

- 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. A COL applicant must describe how the regulatory requirements of 10 CFR Part 73 will be met that are applicable to nuclear power plants.

A COL applicant is required to identify and describe design features, analytical techniques, and technical bases for its design and how the provisions of physical protection system requirements in the NRC regulations will be met, using applicable regulatory guides and NUREG-0800. However, NRC regulatory guides and NUREG-0800 are not regulatory requirements and are not a substitute for compliance with established regulations. Where alternative methods are chosen or differences exist, the COL applicant is required to describe how the proposed alternatives to guidance or acceptance criteria provide acceptable methods of compliance with NRC regulations.

NUREG-0800, Section 13.6.1, Revision 1, June 15, 2010, was used by the staff to complete the physical security COL review.

Regulatory guidance documents, TRs, and accepted industry codes and standards that an applicant may apply to meet regulatory requirements include, but are not limited to the following:

Documents publicly available:

- RG 5.7, “Entry/Exit Control for Protected Areas, Vital Areas, and Material Access Areas,” Revision 1

- RG 5.12, “General Use of Locks in the Protection and Control of Facilities and Special Nuclear Materials”

- RG 5.44, “Perimeter Intrusion Alarm Systems,” Revision 3

- RG 5.62, “Reporting of Safeguards Events,” Revision 1
- RG 5.65, “Vital Area Access Controls, Protection of Physical Protection System Equipment and Key and Lock Controls”
- RG 5.66, “Access Authorization Program for Nuclear Power Plants”
- RG 5.68, “Protection Against Malevolent Use of Vehicles at Nuclear Power Plants”
- RG 5.74, “Managing the Safety/Security Interface”
- RG 5.75, “Training and Qualification of Security Personnel at Nuclear Power Reactor Facilities”
- NRC April 9, 2009, letter, NRC Staff Review of NEI 03-12, “Template for Security Plan, Training and Qualification, Safeguards Contingency Plan, [and Independent Spent Fuel Storage Installation Security Program]” (Revision 6).
- SECY-05-0197, “Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria,” October 28, 2005.

The following documents include security-related or safeguards information 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.”
- RG 5.76, “Physical Protection Programs at Nuclear Power Reactors.”
- RG 5.77, “Insider Mitigation Program.”
- NEI 03-12, Revision 6, “Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Installation Security Program.”
- NUREG/CR-6190, “Update of NUREG/CR-6190 Material to Reflect Postulated Threat Requirements.”

13.6.4 Technical Evaluation

The staff reviewed WLS COL FSAR Section 13.6 and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The staff’s review confirmed that the information in the application and incorporated by reference addresses the required information relating to physical security. The results of the staff’s evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the DC and use this review to evaluate subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from responses to RAIs.
- The staff compared the VEGP PSP, T&QP, and SCP to the corresponding WLS programs. The staff concluded that these plans are sufficiently similar to warrant standard content treatment.
- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.
- The staff verified that the site-specific differences were not relevant.

The staff completed its review and concluded that the evaluation performed for the standard content is directly applicable to the WLS COL application, with the exception discussed in the following paragraph. This standard content material is identified in this report by use of italicized, double-indented formatting. One clarification to the standard content material presented below is that the staff's detailed evaluation of the physical protection program, which is site-specific, is provided in the safeguards information version of the WLS COL application Section 13.6 SER, which is located in the NRC Secure Local Area Network.

There were site-specific RAIs issued to the WLS applicant that resulted in site-specific evaluations for several of the Security Plan review areas. There were also site-specific RAIs issued to the VEGP applicant that were not applicable to the WLS application. In addition, there are several Security Plan review areas with site-specific characteristics requiring a specific review by the staff. For these cases, the staff provides the WLS evaluation in the same location as provided in the VEGP SER, but without the use of italicized, double-indented formatting.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.6.4:

> <u>*AP1000 COL Information Item*</u>
>
> - *STD COL 13.6-1*
>
> *The NRC staff reviewed STD COL 13.6-1 related to COL Information Item 13.6-1, which identified the need for a COL applicant to address the security plan. STD COL 13.6-1 supplemented Section 13.6 of the VEGP COL FSAR by stating the following text is to be added after Section 13.6 of the VEGP ESP SSAR:*
>
>> *The Security Plan consists of the Physical Security Plan, the Training and Qualification Plan, and the Safeguards Contingency*

> *Plan. The Security Plan is submitted to the Nuclear Regulatory Commission as a separate licensing document in order to fulfill the requirements of 10 CFR 52.79(a)(35) and 52.79(a)(36). The Security Plan meets the requirements contained in 10 CFR Part 73 and will be maintained in accordance with the requirements of 10 CFR 52.98. The Plan is categorized as Security Safeguards Information and is withheld from public disclosure pursuant to 10 CFR 73.21.*

Section 13.6 of the VEGP COL FSAR also refers to FSAR Table 13.4-201, "Operational Programs Required by NRC Regulations," as providing the milestones for implementing the security program and cyber security program.

The NRC staff's evaluation of the PSP is documented in Section 13.6.4.1 of this SER. The NRC staff's evaluation of the T&QP is documented in Section 13.6.4.2 of this SER. The NRC staff's evaluation of the SCP is documented in Section 13.6.4.3 of this SER. The NRC staff's evaluation of the safety/security interface is documented in Section 13.6.4.1.17 of this SER. Section 13.6.5 of this SER includes the post-combined license activities. Section 13.6.6 of this SER includes the NRC staff's overall conclusions regarding each of the plan submissions.

The NRC staff's evaluation of the physical protection program is provided in detail in the safeguards information version of the VEGP COL application Section 13.6 SER, which is located in the NRC's Secure Local Area Network, document number ES1000015157. Due to security restraints, the NRC staff's evaluation of the physical protection program presented in this publicly-available SER does not include the same level of detail as the safeguards information version. Those persons with the correct access authorization and need-to-know may view the safeguards information version of the VEGP COL application Section 13.6 SER.

<u>License Conditions</u>

- *Part 10, License Condition 3, Items C.5, D.3, and G.9*

The applicant proposed license condition in Part 10 of the VEGP COL application, which provides the milestones for implementing applicable portions of the Security Program. Specifically, the applicant proposed the following:

> *C. Receipt of Materials – The licensee shall implement each operational program identified below prior to initial receipt of byproduct, source, or special nuclear materials onsite (excluding Exempt Quantities as described in 10 CFR 30.18)*
>
> > *C.5 – Security Program (applicable portions)*
>
> *D. Fuel Receipt – The licensee shall implement each operational program identified below prior to initial receipt of fuel onsite.*

D.3 – Security Program (applicable portions)

G. Fuel Loading – The licensee shall implement each operational program identified below prior to initial fuel load.

G.9 – Physical Security

- *Part 10, License Condition 5*

The applicant proposed license condition in Part 10 of the VEGP COL application, which proposed the maintenance of the PSP, T&QP, and the SCP when nuclear fuel is onsite, and continuing until all nuclear fuel is permanently removed from the site. Specifically, the applicant proposed the following:

> *The licensee shall maintain in effect the provisions of the physical security plan, security personnel training and qualification plan, and safeguards contingency plan, and all amendments made pursuant to the authority of 10 CFR 50.90, 50.54(p), 52.97, and Section VIII of Appendix D to Part 52 when nuclear fuel is onsite, and continuing until all nuclear fuel is permanently removed from the site.*

In a letter dated October 22, 2010, the applicant proposed to revise the [security plan] *milestone included in VEGP COL FSAR Table 13.4-201 to implement the* [security plan] *prior to receipt of fuel onsite (protected area.) The NRC staff finds the implementation milestone for the security ~~program~~* [plan] *(~~security~~ prior to receipt of fuel onsite (protected area)) appropriate and in accordance with the requirement in 10 CFR 73.55. Therefore the staff finds that the proposed License Condition 3, Items C.5, D.3, and G.9 and License Condition 5 are not necessary. The incorporation of proposed changes to the VEGP COL FSAR is tracked as* ***Confirmatory Item 13.6-1.***

Resolution of Standard Content Confirmatory Item 13.6-1

Confirmatory Item 13.6-1 is an applicant commitment to revise its FSAR Table 13.4-201 regarding the implementation milestones for the security program. The staff verified that the VEGP COL FSAR was appropriately revised. As a result, Confirmatory Item 13.6-1 is now closed.

- *Part 10, License Condition 6*

The applicant proposed a license condition to provide a schedule to support the NRC's inspection of operational programs including the PSP, T&QP, and the SCP. Specifically, the applicant proposed the following:

> *The licensee shall submit to the appropriate Director of the NRC, a schedule, no later than 12 months after issuance of the COL, that supports planning for and conduct of NRC inspections of operational programs listed in the operational program FSAR Table 13.4-201. The schedule shall be updated every 6 months*

until 12 months before scheduled fuel loading, and every month thereafter until either the operational programs in the FSAR table have been fully implemented or the plant has been placed in commercial service, whichever comes first.

The staff reviewed the above proposed license condition against the recommendations in SECY-05-0197 as endorsed by the related SRM dated February 22, 2006. The staff concludes these proposed license conditions conform to the guidance in SECY-05-0197 and is, therefore, acceptable.

13.6.4.1 Physical Security Plan

The applicant submitted Part 8 of the COL application for the VEGP PSP, T&QP and SCP, to meet the requirements of 10 CFR 52.79(a)(35) and (36). Part 2, FSAR, Chapter 13, Section 13.6 references the VEGP 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. The VEGP submitted PSP makes references to 10 CFR 50.34(c)(2) and (d)(2). The correct references should be 10 CFR 52.79(a)(35) and (36). It is noted that this is a template error, and both references require that the same criteria be met.

Security plans must describe how the applicant will implement Commission requirements and those site-specific conditions that affect implementation as required by 10 CFR 73.55(c)(1)(i).

The requirements are provided in 10 CFR 73.55(c), and (d) to establish, maintain, and implement a PSP to meet the requirements of 10 CFR 73.55, and 10 CFR Part 73, Appendices B and C. The applicant must show 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. The applicant 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 has reviewed the applicant's description in PSP Section 1 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because the applicant'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 52.79(a)(35) require:

(i) A PSP, describing how the applicant will meet the requirements of 10 CFR Part 73 (and 10 CFR Part 11, if applicable, including the identification and description of jobs as required by 10 CFR 11.11(a) of this chapter, at the proposed facility). The plan must list tests, inspections, audits, and other means to be used to demonstrate compliance with the requirements of 10 CFR Parts 11 and 73, if applicable.

(ii) A description of the implementation of the PSP.

The provisions of 10 CFR 52.79(a)(36) require:

(i) An SCP in accordance with the criteria set forth in 10 CFR Part 73, Appendix C. The safeguards contingency plan shall include plans for dealing with threats, thefts, and radiological sabotage, as defined in 10 CFR Part 73 of this chapter, relating to the special nuclear material and nuclear facilities licensed under this chapter and in the applicant's possession and control. Each application for this type of license shall include the information in the applicant's SCP. (Implementing procedures required for this plan need not be submitted for approval).

(ii) A T&QP in accordance with the criteria set forth in 10 CFR Part 73, Appendix B.

(iii) A cyber security plan (CSP) in accordance with the criteria set forth in 10 CFR 73.54 of this chapter.

(iv) A description of the implementation of the SCP, T&QP, and CSP.

(v) Each applicant who prepares a PSP, an SCP, a T&QP, or a CSP, shall protect the plans and other related Safeguards Information against unauthorized disclosure in accordance with the requirements of 10 CFR 73.21 of this chapter.

The provisions of 10 CFR 52.79(a)(44) require a description of the FFD program required by 10 CFR Part 26 “Fitness for Duty Program” and its implementation.

Requirements are established in 10 CFR 73.55(c)(2) to ensure protection of safeguards information (SGI) against unauthorized disclosure in accordance with 10 CFR 73.21. The applicant’s WLS COLA Part 8 submittal acknowledges that the PSP, the TQ&P and the SCP discuss specific features of the physical security system or response procedures and are SGI.

PSP Section 1 describes the applicant’s commitment to satisfy 10 CFR 50.34(c), 10 CFR 50.34(d), and 10 CFR Part 73 by submitting a PSP, and to controlling the PSP and appendices as Safeguards Information according to 10 CFR 73.21.

The provisions of 10 CFR Part 73, Appendix C, Section II.B.3.b, requires a description of the physical layout of the site.

PSP Section 1.1 provides descriptions of location, site layout, and facility configuration. The PSP describes the physical structures and their locations on the site, description of the protected area, and a description of the site in relation to nearby town, 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 marshalling and coordinating response activities.

In addition, WLS COL FSAR Chapter 2, "Site Characteristics," provides general plant descriptions that include details of the 16 to 80-km (10 to 50-mi) radius of the geographical area of the WLS Units 1 and 2 site, a site area map, and general plant and site descriptions. WLS COL FSAR Chapter 1, references the AP1000 DCD for the principal design and operating characteristics for the design and construction of the WLS Units 1 and 2. WLS COLA Part 1, "General Information," describes the name of the applicant and principal business locations.

The staff reviewed the facility physical layout provided in PSP Section 1.1 and as supplemented by WLS COL FSAR. The staff concluded that the applicant included site-specific conditions that affect the applicant's capability to satisfy the requirements of a comprehensive PSP. The applicant has adequately described the physical structures and their locations onsite and the site in relation to nearby towns, roads, and other environmental features important to the effective coordination of response operations. The applicant described the main and alternate entry routes for law-enforcement assistance forces and the location of control points for marshaling and coordinating response activities in the site-specific law enforcement response plan. The staff concludes that the applicant's security plans have met the requirements for content of a PSP as stated above. Therefore, the staff finds the "Facility Layout" described in the PSP and the WLS COL FSAR acceptable.

13.6.4.1.2 **Performance Objectives**

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.6.4.1:

> ***13.6.4.1.2 Performance Objectives***
>
> *The provisions of 10 CFR 73.55(b)(1) requires, in part, that the applicant 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. The provisions of 10 CFR 73.55(b)(2) establish, in part, 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 the applicant to establish a physical protection program designed to ensure 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, provide defense-in-depth, supporting processes, and implementing procedures, which 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 herein 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 paragraph (a) through (r). VEGP Units 3 and 4 uses the corrective action program to track, trend, correct and prevent recurrence of failures and deficiencies in the physical protection program,

The NRC staff has reviewed the applicant's description in PSP Section 2 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because the applicant'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(b), and is, therefore, acceptable.

13.6.4.1.3 Performance Evaluation Program

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.6.4.1:

13.6.4.1.3 Performance Evaluation Program

Requirements are established in 10 CFR 73.55(b)(4) through (b)(11) for the applicant to analyze and identify site-specific conditions, establish programs, plans, and procedures that address performance evaluations, access authorization, cyber security, insider mitigation, fitness for duty (FFD), corrective actions, and operating procedures. 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 describes 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 the applicant's response strategy. Other assessment methods include formal and informal exercises or drills, self-assessments, internal and external audits and evaluations.

Section 3.0 of the PSP describes 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 the applicant's response strategy. Other assessment methods include formal and informal exercises or drills, self-assessments, internal and external audits and evaluations.

The performance evaluation processes and criteria that assess the effectiveness of the security program, including adequate protection against radiological sabotage, will be established in facility procedures and the deficiencies identified are managed through the corrective action program.

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 ~~dills~~ [drills] *and force-on-force exercises to evaluate security systems effectiveness and response performances of security personnel. In addition, Section 17 of the PSP describes additional detail regarding the applicant's processes for reviews, evaluations and audits that will complement the performance evaluation program.*

The NRC staff has reviewed the applicant'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 the applicant'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(b)(6), 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, periodically re-qualified, and equipped to implement the physical protection program as required by 10 CFR 73.55(b) and 10 CFR Part 73, Appendices B and C.

As explained below, PSP Section 4.0 describes how the applicant meets the requirements of 10 CFR 73.55(d)(1).

<u>Security Organization Management</u>

PSP Section 4.1 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, contractor, or other qualified personnel.

The PSP describes 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. PSP Section 4.1 also describes the roles and responsibilities of the Security Organization. The PSP provides that at least one full-time, Response Team Leader that has the authority for command and control of all security operations is onsite at all times.

The security force implementing the security functions as described in this section of the plan will be either a proprietary force, contractor, or other qualified personnel. The training qualification provision is described in the T&QP.

The staff has reviewed the applicant's description in PSP Sections 4 and 4.1 for the implementation of the site-specific physical protection program in accordance with NRC regulations and NUREG-0800 acceptance criteria. Since the applicant'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 PSP the meets the requirements of 10 CFR 73.55(d), and is therefore acceptable.

13.6.4.1.5 Qualification for Employment in Security

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.6.4.1:

> ***13.6.4.1.5 Qualification for Employment in Security***
>
> *The requirements of 10 CFR 73.55(d)(3) state, in part, that the applicant 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 applicant's T&QP.*
>
> *Section 5 of the PSP describes that employment qualifications for members of the security force are delineated in the T&QP.*
>
> *The NRC staff has reviewed the applicant'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 the applicant'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(d)(3), and is, therefore, acceptable.*

13.6.4.1.6 Training of Facility Personnel

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.6.4.1:

> ***13.6.4.1.6 Training of Facility Personnel***
>
> *Consistent with requirements in 10 CFR 73.55(d)(3),10 CFR 73.56 and 10 CFR Part 73, Appendix B, Section VI.C.1, all personnel who are authorized unescorted access to the applicant's PA 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 are 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 the applicant's PA.*

The NRC staff has reviewed the applicant'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 the applicant'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 73.56 and 10 CFR Part 73, Appendix B, and is, therefore, acceptable.

13.6.4.1.7 Security Personnel Training

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.6.4.1:

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 prior to performing their duties.

Section 7 of the PSP describes that all security personnel are trained, qualified and perform tasks at levels specific for their assignments in accordance with the applicant's T&QP.

The NRC staff has reviewed the applicant'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 the applicant'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(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 SER.

13.6.4.1.8 Local Law Enforcement Liaison

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.6.4.1:

13.6.4.1.8 Local Law Enforcement Liaison

The following requirement is stated in 10 CFR 73.55(k)(9) "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 its 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 the applicant'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 the applicant'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(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 applicant 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 10 CFR Part 73, Appendix B and the T&QP. The provisions of 10 CFR Part 73, Appendix B, Section VI.G.2(a) state, 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. The provisions of 10 CFR Part 73, Appendix B, Sections VI.G.2(b) and (c) delineate the minimum equipment requirements for security personnel and armed response personnel.

PSP Section 9 describes the equipment, including armament, ammunition, and communications equipment that is provided to security personnel in order to ensure that security personnel are capable of performing the function stated in the NRC-approved security plans, applicant's protective strategy, and implementing procedures.

The staff reviewed the applicant's description in PSP Section 9 for the implementation of the site-specific physical protection program in accordance with NRC regulations and NUREG-0800 acceptance criteria. Since the applicant'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(d)(3) and Appendix B, Section VI.G.2 and is therefore acceptable.

13.6.4.1.10 Work Hour Controls

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.6.4.1:

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. 10 CFR 26.205 establishes requirements for work hours. 10 CFR 26.205(a) requires 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.

Section 10 of the PSP describes 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 13.7 of this SER/

13.6.4.1.11 Physical Barriers

The following requirements are established in 10 CFR 73.55(e): "Each applicant 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 10 CFR 73.55(b). (1) The applicant 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 regulation 10 CFR 73.55(b)(3)(ii) states, that the physical protection program must: "Provide defense-in-depth through the integration of systems, technologies, programs, equipment, supporting processes, and implementing procedures as needed to ensure the effectiveness of the physical protection program."

PSP Section 11 provides a general description of how the applicant will implement its program for physical barriers, and that this implementation will meet the performance objectives and requirements of 10 CFR 73.55(b).

Owner Controlled Area (OCA) Barriers

PSP Section 11.1 describes WLS use of OCA barriers at the site.

Vehicle Barriers

PSP Sections 11.2.1 and 11.2.2 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 applicant's PSP also provides that the inspection, monitoring, and maintenance of the vehicle barrier system (VBS) will be included in the facility procedures.

Waterborne Threat Measures

The provisions of 10 CFR 73.55(e)(10)(ii) require the applicant to "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. In accordance with the site-specific analysis, provide periodic surveillance and observation of waterway approaches and adjacent areas."

PSP Section 11.2.3 states that a site-specific analysis for a water-borne DBT has been conducted and documented. The analysis determined that there is no waterborne access to WLS Units 1 and 2.

Protected Area Barriers

The provisions of 10 CFR 73.55(e)(8)(i) require that the protected area perimeter must be protected by physical barriers that are designed and constructed to do the following: (1) limit access to only those personnel, vehicles, and materials required to perform official duties; (2) channel personnel, vehicles, and materials to designated access control portals; and (3) be separated from any other barrier designated as a vital area physical barrier, unless otherwise identified in the PSP.

The descriptions of the protected area barrier are provided in the PSP Section 11.3. The staff notes that these descriptions meet the definitions of physical barriers and protected areas in 10 CFR 73.2 and requirements of 10 CFR 73.55(e)(8).

PSP Section 11.3 describes the extent to which the protected area barrier at the perimeter is separated from a vital area/island barrier. The security plan identifies where the protected area barrier is not separated from a vital area barrier, as required by 10 CFR 73.55(e)(8)(i)(c).

PSP Section 11.3 describes isolation zones. As required in 10 CFR 73.55(e)(7), the isolation zone is maintained in outdoor areas adjacent to the protected area perimeter barrier and is designed to ensure the ability to observe and assess activities on either side of the protected area perimeter.

Vital Area Barriers

The provisions of 10 CFR 73.55(e)(9) require that "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 NRC and identified in the security plans." In addition, 10 CFR 73.55(e)(5) requires that the physical barriers to access of certain vital areas shall be bullet resisting.

PSP Section 11.4 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. PSP Section 11.4 also describes that the reactor control room, central alarm station (CAS) and the location within which the last access control function for access to the protected area is performed, must be bullet resisting.

In RAI 97, Question 13.06-49, the staff requested that the applicant provide clarification regarding functionality in certain vital areas. In a July 6, 2011, response, the applicant confirmed that the response provided in VEGP Reference (R)-COLA RAI 13.06-13 (VEGP eRAI 3394) is also applicable to WLS. On the basis of its review, the staff finds the revised description in the November 17, 2011, PSP Revision 2, acceptable, as it provides the additional information on how the applicant meets the requirements of 10 CFR 73.55(e)(9).

Target Set Equipment

The provisions of 10 CFR 73.55(f) require the following, "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. The licensee shall consider cyber attacks in the development and identification of target sets. 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 10 CFR 73.55(f)(1) and be accounted for in the licensee's protective strategy. 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."

PSP Section 11.5 describes 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.

The staff identified several RAIs relating to target sets for the purpose of reviewing the Westinghouse physical protection program. Westinghouse provided design details as background information to assist an applicant with the development of site-specific target set analyses. The staff evaluated the applicant's responses, and found them acceptable for the DC review of the AP1000 physical protection program. In TR-94, Westinghouse stated in, APP-GW-GLR-066, "AP1000 Safeguards Assessment Report" that target sets were created to aid in the development of the AP1000 physical security system, and that final target sets will be developed by the COL applicant prior to fuel onsite (inside protected areas).

The staff reviewed the applicant's description in PSP Sections 11.5 and 14.5, SCP Section 7 and information in Westinghouse TR-94, APP-GW-GLR-066, "AP1000 Safeguards Assessment Report," for the implementation of the site-specific physical protection program in accordance with NRC regulations and NUREG-0800 acceptance criteria. Since the applicant's description in PSP Sections 11.5 and 14.5, SCP Section 7, and the information in Westinghouse TR-94 are consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description provided in the PSP Sections 11.5 and 14.5 and SCP Section 7 meets the requirements of 10 CFR 73.55(f)(1), (3), and (4) and is, therefore, acceptable. The target sets, target set analysis and site protective strategy are in the facility implementing procedures. They will be subject to future NRC inspections in accordance with 10 CFR 73.55(c)(7)(iv) and 10 CFR Part 73, Appendix C, Section II.B.5(iii).

<u>Delay Barriers</u>

The provisions of 10 CFR 73.55(e)(3)(ii) require that physical barriers must "provide deterrence, delay, or support access control" to perform the required function of the applicant physical protection program. The PSP describes the use of delay barriers at WLS Units 1 and 2.

PSP Section 11.6 includes a description of the use of Delay Barriers to meet requirement of 10 CFR 73.55(e).

The staff reviewed the applicant'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 NRC regulations and NUREG-0800 acceptance criteria. Since the applicant'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 meet the requirements of 10 CFR 73.55(e), and are, therefore, acceptable.

13.6.4.1.12 Security Posts and Structures

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.6.4.1:

> ***13.6.4.1.12 Security Posts and Structures***
>
> *The provisions of 10 CFR 73.55(e)(5) require that the reactor control room, the CAS, and the location within which the last access control function for access to the PA is performed, must be bullet-resisting.*
>
> *Section 12 of the PSP describes 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 the applicant'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 the applicant'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(e)(5), and is, therefore, acceptable.*

13.6.4.1.13 Access Control Devices

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.6.4.1:

> ***13.6.4.1.13 Access Control Devices***
>
> *It is stated in 10 CFR 73.55(g)(1) that, consistent with the function of each barrier or barrier system, the applicant shall control personnel, vehicle, and material access, as applicable, at each access control point in accordance with the physical protection program design requirements of 10 CFR 73.55(b).*
>
> *The provisions of 10 CFR 73.55(g)(6) require control of access control devices as stated: "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."*
>
> <u>*Types of Security-Related Access Control Devices*</u>
>
> *Section 13.1 of the PSP describes that the applicant uses security-related access control devices to control access to protected and vital areas and security systems.*
>
> <u>*Control and Accountability*</u>
>
> *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 and the accountability and inventory control process, and the circumstances that require changes in security-related locks. The applicant uses facility procedures to produce, control, and recover keys, locks, and combinations for all areas and equipment, which serve to reduce the probability of compromise. The 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 the applicant'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 the applicant'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 meet the requirements of 10 CFR 73.55(g)(1) and (6), and are, therefore, acceptable.

13.6.4.1.14 Access Requirements

Access Authorization and Fitness for Duty

The provisions of 10 CFR 73.55(b)(7) require the applicant shall establish, maintain, and implement an access authorization program in accordance with 10 CFR 73.56 and shall describe the program in the PSP. The provisions of 10 CFR Part 26 require the applicant to establish and maintain a FFD program.

PSP Section 14.1 describes that the access authorization program implements regulatory requirements utilizing the provisions in RG 5.66, "Nuclear Power Plant Access Authorization Program," Revision 1, July 2009. The staff finds that RG 5.66, is an acceptable method to meet the requirements of 10 CFR 73.55(b)(7).

The staff reviewed the applicant'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. Since the applicant'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(b)(7), 10 CFR 73.56 and 10 CFR Part 26 and is, therefore, acceptable.

Insider Mitigation Program

The provisions of 10 CFR 73.55(b)(9) require that the applicant shall establish, maintain, and implement an insider mitigation program and shall describe the program in the PSP. 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 applicant's capability to prevent significant core damage and spent fuel sabotage. The insider mitigation program must include elements from: the access authorization program, the FFD program, the cyber security program and the physical protection program.

PSP Section 14.2 describes how the applicant will establish, maintain, and implement an insider mitigation program utilizing the guidance in RG 5.77, "Insider Mitigation Program." The insider mitigation program requires elements from the access authorization program described in 10 CFR 73.56; FFD program described in 10 CFR Part 26; the cyber security program described in 10 CFR 73.54; and the physical security program described in 10 CFR 73.55. In addition, PSP Section 14.2 describes the integration of the programs discussed above to form a cohesive and effective insider mitigation program. The applicant addressed the observations for the detection of tampering. The staff notes that RG 5.77 is an acceptable method to meet the requirements of 10 CFR 73.55(b)(9).

The staff reviewed the applicant's description in PSP Section 14.2 for the implementation of the site-specific physical protection program in accordance with NRC regulations and NUREG-0800 acceptance criteria. Since the applicant'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(b)(9) and is, therefore, acceptable.

Picture Badge Systems

Requirements for badges are stated in 10 CFR 73.55(g)(6)(ii). "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, identification badges may be removed from the protected area under limited conditions and only by authorized personnel. Records of all badges shall be retained and shall include name and areas to which persons are granted unescorted access.

The provisions of 10 CFR 73.55(g)(7)(ii) require that individuals not employed by the applicant but who require frequent or extended unescorted access to the protected area and/or vital areas to perform duties and responsibilities required by the applicant at irregular or intermittent intervals, shall satisfy the access authorization requirements of 10 CFR 73.56 and 10 CFR 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.

PSP Section 14.3 describes the site picture badge system. Identification badges will be displayed while individuals are inside the protected area or vital areas. When not in use, badges may be removed from the protected area 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 protected area. 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 staff reviewed the applicant's description in PSP Section 14.3 for the implementation of the site-specific physical protection program in accordance with NRC regulations and NUREG-0800 acceptance criteria. Since the applicant'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(g)(6) and (7) and is, therefore, acceptable.

Searches

The provisions of 10 CFR 73.55(h) require, in part, that applicants meet the objective to detect, deter, and prevent the introduction of firearms, explosives, incendiary devices, or other items that could be used to commit radiological sabotage. To accomplish this, applicant'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.

PSP Section 14.4 provides an overview description of the search process for vehicle, personnel, and materials. The search process is conducted using security personnel, specifically trained non-security personnel and technology. Detailed discussions of actions to be taken in the event unauthorized materials are discovered are found in implementing procedures.

Vehicle Barrier Access Control Point

The provisions of 10 CFR 73.55(h)(2)(ii) through (v) provide the requirements for the applicant to search vehicles at the OCA and 10 CFR 73.55(h)(3) provides requirements for searches of personnel, vehicles, and materials prior to entering the protected area.

PSP Section 14.4.1 describes the process for the search of personnel, vehicles, and materials at predetermined locations prior to granting access to designated facility areas identified by the applicant as needed to satisfy the physical protection program. The applicant stated that it has developed specific implementing procedures to address vehicle and materials searches at these locations.

Protected Area Packages and Materials Search

PSP Section 14.4.2 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 these items can enter the WLS Units 1 and 2 protected area. Detailed provisions for conducting these searches are found in applicant implementing procedures and include the search and control of bulk materials and products. Applicant 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

PSP Section 14.4.3 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 protected area. Detailed provisions for conducting these searches are found in the applicant's implementing procedures. The applicant's implementing procedures also address the search methodologies for vehicles that must enter the protected area under emergency conditions.

Protected Area Personnel Searches

PSP Section 14.4.4 describes the process for searches of all personnel requesting access into protected areas. The PSP describes the search 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 prior to granting access into the protected area. All persons except official Federal, State, and local law enforcement agency personnel on official duty are subject to these searches upon entry to the protected area. Detailed discussions of observation and control measures are found in implementing procedures.

Protected Area Access Controls

PSP Section 14.4.5 of the PSP describes the process for controlling access at all points where personnel or vehicles could gain access into the applicant's protected area. The plan notes that principal personnel access to the protected area is through a lockable portal. Personnel are only permitted into the protected area after positive identification (ID) verification, access authorization verification, and a search is performed per PSP Section 14.4. Vehicles are controlled through positive control methods described in the facility procedures.

Escort and Visitor Requirements

The provisions of 10 CFR 73.55(g)(7) state in part, that the applicant may permit escorted access to protected and vital areas to individuals who have not been granted unescorted access in accordance with the requirements of 10 CFR 73.56 and 10 CFR Part 26. Provision in 10 CFR 73.55(g)(8) establishes escort requirements. Applicants are required to implement procedures to process, escort and control visitors. Procedures shall address confirmation of identity of visitors, maintenance of a visitor control register, visitor badging and escort controls including, training, communications, and escort ratios.

PSP Section 14.4.6 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 provisions 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 staff reviewed the applicant'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 NRC regulations and NUREG-0800 acceptance criteria. Since the applicant'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(h)(2), 10 CFR 73.55(h)(3), 10 CFR 73.55(g)(7) and 10 CFR 73.55(g)(8) and are, therefore, acceptable.

Vital Area Access Controls

The provisions of 10 CFR 73.55(g)(4) require that applicants control access into vital areas consistent with established access authorization lists. In response to a site-specific credible threat or other credible information, applicants shall 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 provisions of 10 CFR 73.56(j) require the applicant to 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 to perform their duties and responsibilities. The list must be approved by a cognizant applicant manager or supervisor responsible for directing the work activities of the individual who is granted unescorted access to each vital area, and be updated and re-approved no less frequently than every 31 days.

PSP Section 14.5 describes vital areas and states that the applicant 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 that is reassessed and reapproved at least once every 31 days. Additional access control measures are described in the facility procedures.

In RAI 97, Question 13.06-50, the staff requested that the applicant clarify how the minimum vital areas and equipment are protected, including any proposed revision to this section of the security plan. The applicant responded that PSP Section 14.5 will be revised, as necessary, to clearly identify any regulatory minimum vital areas that are bounded by the larger vital areas included in the list. In a July 6, 2011, response, the applicant stated that the VEGP R-COLA RAI 13.6-19October 16, 2009, response is applicable to WLS Units 1 and 2. In Enclosure 22 of their July 6, 2011, response, the applicant provided a description that clearly identifies the minimum vital areas. On the basis of its review, the staff finds the revised description in the PSP Revision 2, dated November 17, 2011, acceptable, as it provides the additional information on how the applicant meets 10 CFR 73.55(e)(9) and 10 CFR 73.55(g)(4).

The staff reviewed the applicant's description in PSP Section 14.5 for the implementation of the site-specific physical protection program in accordance with NRC regulations and NUREG-0800 acceptance criteria. Since the applicant'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(g)(4) and is, therefore, acceptable.

13.6.4.1.15 Surveillance Observation and Monitoring

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.6.4.1.15:

> ***13.6.4.1.15 Surveillance Observation and Monitoring***
>
> *The provisions of 10 CFR 73.55(i)(1) require that the applicant establish and maintain intrusion detection systems that satisfy the design requirements of 10 CFR 73.55(b) and provide, at all times, the capability to detect and assess unauthorized persons and facilitate the effective implementation of the protective strategy.*

Illumination

The provisions of 10 CFR 73.55(i)(6) require, in part, that all areas of the facility are provided with illumination necessary to satisfy the design requirements of 10 CFR 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 applicant may augment the facility illumination system by means of low-light technology to meet the requirements of this section or otherwise implement the protective strategy. The applicant shall describe in the security plans how the lighting requirements of this section are met and, if used, the type(s) and application of low-light technology.

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. A discussion of the implementation of technology using fixed and non-fixed low light level cameras or alternative technological means is provided. The applicant has addressed the potential for loss of lighting and the compensatory actions that would be taken if that event were to occur.

Surveillance Systems

The provisions of 10 CFR 73.55(i)(1) require, in part, that the applicant implement, establish, and maintain intrusion detection and assessment, surveillance, observation and monitoring systems to satisfy the design requirements of 10 CFR 73.55(b), and of the applicant's OCA.

Section 15.2 of the PSP describes 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 assessing alarms or unauthorized activities. Certain systems provide real-time and recorded play back of recorded video images. The specifics of surveillance systems are described in facility implementing procedures.

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 play-back/recorded video images of the detected activities before and after each alarm annunciation. The PSP describes how the applicant will meet regulatory requirements for redundancy, tamper indication and uninterruptable power supply.

Central Alarm Station (CAS) and Secondary Alarm Station (SAS) Operation

The provisions of 10 CFR 73.55(i)(4) provide requirements for alarm stations. It is required, in 10 CFR 73.55(i)(4)(i), that both alarm stations must be designed and equipped to ensure that a single act, in accordance with the DBT of radiological sabotage defined in 10 CFR 73.1, cannot disable both alarm stations. The applicant shall ensure the survivability of at least one alarm station to maintain the ability to perform the following functions: 1) detect and assess alarms; 2) initiate and coordinate an adequate response to an alarm; 3) summon offsite assistance; and 4) provide command and control. 10 CFR 73.55(i)(4)(iii) requires that alarm stations must be equal and redundant.

Section 15.4 of the PSP describes the functional operations of the CAS and the SAS. The PSP provides that the alarm stations are equipped, such that no single act will disable both alarm stations. The applicant's PSP provides 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.

Security Patrols

Owner Controlled Area (OCA) Surveillance and Response

The provisions of 10 CFR 73.55(e)(6) require that the applicant establish and maintain physical barriers in the OCA as needed to satisfy the physical protection program design requirements of 10 CFR 73.55(b). It is required, in 10 CFR 73.55(i)(5)(ii), in part, that the applicant provide continuous surveillance, observation and monitoring of the OCA and that 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 provides that details regarding the implementation of OCA surveillance techniques are found in facility procedures. The PSP provides a discussion regarding the implementation of manned and video options for patrolling and surveillance of the OCA.

Protected and Vital Area Patrols

The provisions of 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 pathways, check external areas of the PA and vital area portals, periodically inspect vital areas, conduct random patrols of accessible target set equipment, be trained to recognize obvious tampering and if detected, initiate an appropriate response in accordance with established plans and procedures.

Section 15.5.2 of the PSP describes the process employed by the applicant 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. The applicant has addressed the observations for the detection of tampering in Section 14.2 of the PSP and in the facility procedures.*

The NRC staff has reviewed the applicant'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 the applicant'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(b) and (i), and are, therefore, acceptable.

13.6.4.1.16 **Communications**

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.6.4.1.16:

13.6.4.1.16 Communications

The provisions of 10 CFR 73.55(j)(1) through (6) describe the requirements for establishment and maintenance of continuous communication capabilities 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 LLEA and the control room. Non-portable communications must remain operable from independence power sources. The applicant must identify areas where communications could be interrupted or not maintained.

<u>*Notifications (Security Contingency Event Notifications)*</u>

Section 16.1 of the PSP describes that the applicant have a process to ensure that continuous communications are established and maintained between the onsite security force staff and the offsite support agencies.

<u>*System Descriptions*</u>

Section 16.2 of the PSP describes the establishment and maintenance of the communications system. Detailed descriptions of security systems are included in the facility procedures. VEGP 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 communication protocols are available between the CAS, SAS and the control room.

The NRC staff has reviewed the applicant'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 the applicant'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(j)(1) through (6), and are, therefore, acceptable.

13.6.4.1.17 Review, Evaluation and Audit of the Physical Security Program

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.6.4.1.17.

13.6.4.1.17 Review, Evaluation and Audit of the Physical Security Program

The provisions of 10 CFR 73.55(m) require, 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 original plan implementation, or a change in personnel, procedures, equipment or facilities, which 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 describes that the physical security program is reviewed 12 months following initial implementation and at least every 24 months by individuals independent of both security program management and 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 upon 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 potentially could adversely affect safety/security.

The results and recommendations of the physical security program review, management's finding on whether the physical security program is currently effective 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 corrective action program.

In RAI 13.6-36, the NRC staff requested that the applicant address the requirements of 10 CFR 73.58, “Safety/security requirements for nuclear power reactors.” In its response dated May 14, 2010, the applicant stated that management controls and processes used to establish and maintain an effective interface between nuclear safety and physical security are addressed by administrative procedures. The applicant committed to revise VEGP COL FSAR Section 13.5.1 to include the safety/security interface implementation process in the list of procedural instructions provided in plant administrative procedures.

On the basis of its review, the NRC staff finds that since the applicant will revise VEGP COL FSAR Section 13.5.1 to incorporate the requirements for safety/security interfaces, the response to RAI 13.6-36 meets the requirements of 10 CFR 73.58 and is, therefore, acceptable. The incorporation of changes to the VEGP COL FSAR Section 13.5.1 is being tracked as ***Confirmatory Item 13.6-2****.*

<u>Resolution of Standard Content Confirmatory Item 13.6-2</u>

Confirmatory Item 13.6-2 is an applicant commitment to revise its FSAR Section 13.5 regarding the requirements of safety/security interfaces. The staff verified that the VEGP COL FSAR was appropriately revised. As a result, Confirmatory Item 13.6-2 is now closed.

The NRC staff has reviewed the applicant’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 the applicant’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(m), and is, therefore, acceptable.

During the review of the Levy Nuclear Plant, Units 1 and 2, COLA, the staff raised concerns in LNP RAI 13.06.01-1 regarding how the LNP applicant, once licensed, would analyze and identify changes in the site-specific conditions related to the AP1000's structures, systems, and components (SSCs) (described in certain technical reports), resulting from changes made to the LNP Units 1 and 2 COL between issuance of the COL and the security program implementation milestones provided in LNP FSAR Table 13.4-201 to ensure that the security plan continues to meet 10 CFR 73.55(b)(4). This LNP RAI also requested that the applicant clarify how the applicant, once licensed, will ensure that the as-built plant continues to meet all physical protection program design and performance criteria in 10 CFR 73.55 at the time the physical protection program is implemented.

In the staff’s safety review of LNP Units 1 and 2 COL FSAR Section 13.4.1.18, the staff noted the applicant’s responses and proposed changes to their application acceptable as follows:

In the Duke Energy Florida (DEF) response letter, “Revised Response to NRC RAI Letter 119 – Related to Standard Review Plan Section 13.6, Physical Security, for the Levy Nuclear Plant, Units 1 and 2, Combined License Application,” dated August 7, 2014, (ADAMS Accession Number ML14220A433), the applicant stated that a future revision of the LNP COL application will reflect the changes discussed in this response.

Associated LNP COL Application Revisions:

COLA Part 2, FSAR Chapter 13 will be revised to add text to Section 13.5.1, "Administrative Procedures" under the statement: "The plant administrative procedures provide procedural instructions for the following: ," 19th bullet as shown below. The left-hand margin annotation for this added text will be "LNP COL 13.5-1."

A process for implementing the safety/security interface requirements of 10 CFR 73.58.

A process is in effect at the time of issuance of the combined license and was developed using NRC endorsed industry guidance. This process is used to manage safety/security interface while the security procedures and emergency plan implementing procedures are being developed and implemented.

The staff finds that the response to RAI 13.06.01 meets the requirements of 10 CFR 73.55(b)(4), and is acceptable, because it provides a commitment to implement administrative procedures to manage the safety/security interface during the construction phase and throughout the operational phase.

The staff reviewed the applicant’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. As set forth above, the applicant’s description in the PSP meets the requirements of 10 CFR 73.55(b)(4), and 10 CFR 73.55(m), and therefore is acceptable. The staff confirmed that the applicant incorporated the proposed changes to the LNP COL FSAR Section 13.5.1 in Revision 7 of the FSAR.

In November 13, 2014, letter, the WLS applicant endorsed the above response and changes to the LNP Units 1 and 2 COL application. The staff verified these same changes in the WLS Units 1 and 2 COL application and finds them acceptable.

13.6.4.1.18 Response Requirements

The provisions of 10 CFR 73.55(k) require, in part, that the applicant 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 applicant must ensure that necessary equipment is in supply, working, and readily available. The applicant must ensure training has been provided to all armed members of the security organization who will be available onsite to implement the applicant’s protective strategy as described in the facility procedures and 10 CFR Part 73, Appendix C. The applicant must have facility procedures to reconstitute armed response personnel and have established working agreement(s) with LLEA. The applicant must have implemented a threat warning system to accommodate heightened security threats and coordination with NRC representatives.

PSP Section 18 describes an armed response team, as well as its responsibilities, training, and equipment, and the number of armed response force personnel required to be immediately available at all times to implement each site's protective strategy. The PSP provides for training in accordance with the requirements of 10 CFR Part 73, Appendix B that 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 LLEA. The PSP also describes procedures to manage the threat warning system.

In RAI 97, Question 13.06-44, the staff requested that the applicant clarify PSP Section 18, which details the minimum number of armed responders continuously in the protected area. The staff also requested that the applicant explain how this number correlates with the expected number detailed in Westinghouse TR 94, AP1000 Safeguards Assessment Report.

In a July 6, 2011, letter, the applicant provided an explanation of how they determined the minimum numbers of armed responders needed for the WLS site. The applicant also provided a metric showing the staffing relationship between Westinghouse TR 94, AP1000 Safeguards Assessment Report, and staffing positions and responsibility for WLS Site Units 1 and 2.

On the basis of its review, the staff finds the response to RAI 97, Question 13.06-44 acceptable. The applicant's metric provided the needed clarification on the minimum number of armed responders continuously in the protected area and the expected number detailed in Westinghouse TR 94, AP1000 Safeguards Assessment Report.

The staff reviewed the applicant's description in PSP Section 18 for the implementation of the site-specific physical protection program in accordance with NRC regulations and NUREG-0800 acceptance criteria. Since the applicant'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(k) and is, therefore, acceptable.

13.6.4.1.19 **Special Situations Affecting Security**

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.6.4.1.19:

> ***13.6.4.1.19 Special Situations Affecting Security***
>
> *The provisions of 10 CFR 73.58 require that each operating nuclear power reactor applicant with a license issued under 10 CFR Part 50 shall comply with the following requirements: the applicant 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; 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); where potential conflicts are identified, the applicant shall communicate them to appropriate 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 assessments to manage increased risk of special situations affecting security.

Refueling/Major Maintenance

Section 19.1 of the PSP describes that, for refueling or major maintenance activities, the PSP describes that security procedures identify measures for implementation of actions prior to refueling or major maintenance activities. These measures include controls to ensure that a search is conducted prior to revitalizing an area, that protective barriers and alarms are fully operational, and post-maintenance performance testing to ensure operational readiness of equipment in accordance with 10 CFR 73.55(n)(8).

Construction and Maintenance

Section 19.2 of the PSP describes that during periods of construction and maintenance when temporary modifications are necessary, that the applicant will implement measures that provide for equivalency in the physical protective measures and features impacted by the activities, such that physical protection measures are not degraded. The process for making such changes or modifications is included in the facility procedures.

The NRC staff has reviewed the applicant'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 the applicant'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(n)(8) and 10 CFR 73.58, and are, therefore, acceptable.

13.6.4.1.20 **Maintenance, Testing and Calibration**

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.6.4.1.20:

13.6.4.1.20 Maintenance, Testing and Calibration

In accordance with 10 CFR 73.55(n), the applicant is required to 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 have the capability of performing their intended functions. The regulation requires that the applicant describe their maintenance testing and calibrations program in the PSP, and that the implementing procedures describe the details and intervals for conducting these activities. Applicant procedures must identify criteria for documenting deficiencies in the corrective action program and ensuring data *protection in accordance with 10 CFR 73.21. The applicant must conduct periodic operability testing of the intrusion alarm system and must conduct performance testing in accordance with the PSP and implementing procedures. Communication equipment must be tested not less than daily, and search equipment must also be tested periodically. Procedures must be established for testing equipment located in hazardous areas, and procedures must be established for returning equipment to service after each repair.*

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 the applicant shall conduct intrusion detection testing in accordance with recommended testing procedures described in RG 5.44," Perimeter Intrusion Alarm System." 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 NRC staff has reviewed the applicant's description in PSP 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 the applicant'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(n), and are, therefore, acceptable.

13.6.4.1.21 Compensatory Measures

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.6.4.1.21:

13.6.4.1.21 Compensatory Measures

The provisions of 10 CFR 73.55(o) require, in part, that the applicant shall identify criteria and measures to compensate for degraded or inoperable equipment, systems, and components to meet the requirements of this section. 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. Compensatory measures must be implemented within specific time frames necessary to meet the appropriate portions of 10 CFR 73.55(b) and described in the security plans.

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 assure 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, fixed and non-fixed closed circuit television, play-back 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 the applicant'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 the applicant'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(o), and are, therefore, acceptable.

13.6.4.1.22 Records

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.6.4.1.22:

13.6.4.1.22 Records

The provisions of 10 CFR Part 26, 10 CFR 73.55(q), 10 CFR 73.56(k) and (o), 10 CFR Part 73, Appendix B. Section VI.H., Appendix C, Section II.C and 10 CFR 73.70, require that the applicant must retain and maintain all records required to be kept by the 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 three years after the record is superseded, unless otherwise specified by the Commission. The applicant 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 applicant 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 applicant or a contractor. Review and audit reports must be maintained and available for inspection for a period of three 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 mention 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; vital area access transaction records; vitalization and de-vitalization records; vital area access list reviews; security plans and procedures; security patrols, inspections and tests; maintenance; CAS and SAS alarm annunciation and security response records; local law enforcement agency 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 the applicant'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 the applicant'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(q), 10 CFR 73.55(o) and 10 CFR 73.70, and are, therefore, acceptable.*

13.6.4.1.23 Digital Systems Security

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.6.4.1.23:

13.6.4.1.23 Digital Systems Security

Section 23 of the PSP addresses digital systems security. The applicant 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.

The NRC staff's review of the cyber security plan is found Section 13.8 of this SER.

13.6.4.1.24 Temporary Suspension if Security Measures

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.6.4.1.24:

13.6.4.1.24 Temporary Suspension if Security Measures

The provisions of 10 CFR 73.55(p) allow the applicant to "suspend implementation of affected requirements of this section under the following conditions: In accordance with 10 CFR 50.54(x) and 50.54(y) of this chapter, the licensee may suspend any security measures under this section in an emergency when this action is immediately needed to protect the public health and safety and no action consistent with license conditions and technical specifications that can provide adequate or equivalent protection is immediately apparent. This suspension of security measures must be approved as a minimum by a licensed senior operator before taking this action. During severe weather when the suspension of affected security measures is immediately needed to protect the personal health and safety of security force personnel and no other immediately apparent action consistent with the license conditions and technical specifications can provide adequate or equivalent protection. This suspension of security measures must be approved, as a minimum, by a licensed senior operator, with input from the security supervisor or manager, before taking this action."

<u>Suspension of Security Measures in Accordance with 10 CFR 50.54(x) and (y)</u>

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 provides a description of 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 are reported and documented in accordance with the provisions of 10 CFR 73.71. This suspension of security measures must be approved, as a minimum, by a licensed senior operator, with input from the security supervisor or manager, before taking this action. 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 the applicant'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 the applicant'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(p), and are, therefore, acceptable.

13.6.4.1.25 Appendix A Glossary of Terms and Acronyms

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.6.4.1.25:

13.6.4.1.25 Appendix A Glossary of Terms and Acronyms

Appendix A, "Glossary of Terms and Acronyms," was reviewed and found to be consistent with the NRC endorsed NEI 03-12, Revision 6 template.

13.6.4.1.26 Conclusions on the Physical Security Plan

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.6.4.1.26:

13.6.4.1.25 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 SER, the PSP meets the requirements of 10 CFR 73.55(a) through (r). The target sets, 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 COL application and are, therefore, 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). The NRC staff 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.*

13.6.4.2 Appendix B Training and Qualification Plan

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.6.4.2:

13.6.4.2 Appendix B Training and Qualification Plan

13.6.4.2.1 Introduction

13.6.4.2.1 Introduction

The provisions of 10 CFR 73.55(c)(4) state that the applicant establish, maintain, implement, and follow a T&QP that describes how the criteria set forth in 10 CFR Part 73, Appendix B will be implemented.

The provisions of 10 CFR 73.55(d)(3) state that the applicant 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 10 CFR Part 73, Appendix B 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 applicant training programs to ensure each individual is trained, qualified, and periodically requalified 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 armed and unarmed individuals shall be requalified at least annually in accordance with the requirements of the Commission-approved T&QP.

The T&QP describes that it is written to address the requirements found in 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 applicable to the T&QP. Additional section-by-section evaluations and discussions are found in the following paragraphs.

13.6.4.2.2 Employment Suitability and Qualification

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.6.4.2.2:

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 provisions of 10 CFR Part 73, Appendix B, Section VI.B.1(a) require, in part, that 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) attained the age of 21 for an armed capacity or the age of 18 for an unarmed capacity; (3) not have any felony convictions that reflect on the individual's reliability; and (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. Applicants 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 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 follows the regulation in 10 CFR Part 73, Appendix B, Section VI.B.1(a).

Physical Qualifications

The provisions of 10 CFR Part 73, Appendix B, Section VI.B.2 require, in part, that individuals whose duties and responsibilities are directly associated with the effective implementation of the Commission-approved security plans, applicant 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 details those individuals that 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 that are found on the critical task matrix shall demonstrate the necessary physical qualifications prior to duty.

Physical Examination

It is stated in 10 CFR Part 73, Appendix B, Section VI.B.2(a)(2), 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, applicant protective strategy, and implementing procedures.

The provisions of 10 CFR Part 73, Appendix B, Section VI.B.2(a)(3) state, in part, that the physical examination must be administered by a licensed health professional with the final determination being made by a licensed physician to verify the individual's physical capability to perform assigned duties and responsibilities.

The provisions of 10 CFR Part 73, Appendix B, Section VI.B.2(b) through (e) provide the minimum requirements that individuals must meet, and include 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 medical examinations before returning 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 that 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 the applicant'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 the applicant'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, Sections VI.B.1 and VI.B.2, and are, therefore, acceptable.

Medical Examinations and Physical Fitness Qualifications

The provisions of 10 CFR Part 73, Appendix B, Section VI.B.4(a) require, 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, and that the applicant 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 provisions of 10 CFR Part 73, Appendix B, Section VI.B.4(b) require, 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 the applicant'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 the applicant'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.4(a) and 10 CFR Part 73, Appendix B, Section VI.B.4(b), and is, therefore, acceptable.

Psychological Qualifications

General Psychological Qualifications

The provisions of 10 CFR Part 73, Appendix B, Section VI.B.3(a) require, in part, that armed and unarmed individuals shall demonstrate the ability to apply good judgment, mental alertness, 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 details that individuals whose 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

The provisions of 10 CFR Part 73, Appendix B, Section VI.B.3(b) require, in part, 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 this section, have no emotional instability that would interfere with the effective performance of assigned duties and responsibilities.

The provisions of 10 CFR Part 73, Appendix B, Section VI.B.3(c) require 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 administration of psychological and emotional determination that will be conducted by appropriately licensed and trained individuals.

The NRC staff has reviewed the applicant'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 the applicant'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, Sections VI.B.3(a), (b) and (c), and are, therefore, acceptable.

Documentation

The provisions of 10 CFR Part 73, Appendix B, Section VI.H.1 require, 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 describes that qualified training instructors create the documentation of 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 the applicant'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 the applicant'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.H.1 and is, therefore, acceptable.

Physical Requalification

The provisions of 10 CFR Part 73, Appendix B, Section VI.B.5 require that: (a) at least annually, armed and unarmed individuals shall be required to demonstrate *the capability to meet the physical requirements of this appendix and the applicant's 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 describes that physical requalification is conducted at least annually, and documented as described in the PSP.

The NRC staff has reviewed the applicant'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 the applicant'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 provisions of 10 CFR Part 73, Appendix B, Section VI.C.1 provide for 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. Each individual who is performing assigned duties and responsibilities identified in the commission-approved security plans shall be trained before assignment in accordance with the requirements of 10 CFR Part 73, Appendix B, and the T&QP and the PSP. Such personnel must be trained and qualified in the use of all equipment or devices required to effectively perform all assigned duties and responsibilities.

T&QP Section 3.1 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 prior to performing their duties.

The staff reviewed the applicant's description in T&QP Sections 3.0 and 3.1 for the implementation of the site-specific physical protection program in accordance with NRC regulations and NUREG-0800 acceptance criteria. Since the applicant'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.C.1 and is, therefore, acceptable.

On-the-job Training

The provisions of 10 CFR Part 73, Appendix B, Sections VI.C.2(a) through (c) provides requirements for on-the-job training. On-the-job training performance standards and criteria must ensure that each individual demonstrates the requisite knowledge, skills and abilities. 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) re-deployment; (8) communications (primary and alternate); (9) use of assigned equipment; (10) target sets; (11) table top drills; (12) command and control duties; (13) applicant's protective strategy.

The T&QP provides a comprehensive discussion of the applicant's approach to meeting the requirements for on-the-job training.

The staff reviewed the applicant's description in T&QP Section 3.2 for the implementation of the site-specific physical protection program in accordance with NRC regulations and NUREG-0800 acceptance criteria. Since the applicant'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, Sections VI.C.2(a) through (c) and is, therefore, acceptable.

Critical Task Matrix

The provisions of 10 CFR Part 73, Appendix B, Section VI.C.2(b) require, in part, that each individual who is assigned duties and responsibilities identified in the NRC-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 includes a critical task matrix as T&QP Table 1. This matrix addresses the means through which each individual will demonstrate the required proficiencies. Tasks that individuals must perform are listed in RG 5.75.

The staff reviewed the applicant's description in T&QP Section 3.3 for the implementation of the site-specific physical protection program in accordance with NRC regulations and NUREG-0800 acceptance criteria. Since the applicant'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.C.2(b) and is, therefore, acceptable.

Initial Training and Qualification Requirements

The provisions of 10 CFR Part 73, Appendix B, Sections VI.C.1(a) through (b) provide the requirements for duty training.

The provisions of 10 CFR Part 73, Appendix B, Section VI.D.1(a) provide the requirements for demonstration of qualification.

T&QP Section 3.4 details that individuals are trained and qualified prior to performing security-related duties within a security organization and must meet the minimum qualifying standards in T&QP Sections 3.4.1 and 3.4.2.

Written Examination

The provisions of 10 CFR Part 73, Appendix B, Section VI.D.1(b)(1) included in the PSP provide that written exams must include those elements listed in the Commission-approved T&QP to demonstrate an acceptable understanding of assigned duties and responsibilities, and will include the recognition of potential tampering involving both safety and security equipment and systems.

Hands on Performance Demonstration

The provisions of 10 CFR Part 73, Appendix B, Section VI.D.1(b)(2) require 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 each individual demonstrates the knowledge, skills, and abilities required to effectively perform the task.

T&QP Sections 3.4.1 and 3.4.2 describe the measures that are implemented by the applicant that meet the requirements and as has otherwise been described in 10 CFR Part 73, Appendix B, Section VI.D.1(b)(2).

The staff reviewed the applicant'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 NRC regulations and NUREG-0800 acceptance criteria. Since the applicant'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, Sections VI.C.1 and D.1 and is, therefore, acceptable.

Continuing Training and Qualification

The provisions of 10 CFR Part 73, Appendix B, Section VI.D.2 state, in part, that armed and unarmed individuals shall be re-qualified at least annually in accordance with the requirements of this appendix and the NRC-approved T&QP. The results of requalification must be documented by a qualified training instructor and attested by a security supervisor.

T&QP Section 3.5 provides discussion regarding the management of the requalification program to ensure that each individual is trained and qualified. In part, the applicant'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 the date the training was actually completed.

The staff reviewed the applicant's description in T&QP Section 3.5 for the implementation of the site-specific physical protection program in accordance with NRC regulations and NUREG-0800 acceptance criteria. Since the applicant'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.D.2 and is, therefore, acceptable.

Annual Written Examination

The provisions of 10 CFR Part 73, Appendix B, Section VI.D.(b)(3) provide 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 NRC-approved T&QP to demonstrate an acceptable understanding of assigned duties and responsibilities.

T&QP Section 3.5.1 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; tactics and tactical deployment and engagement.

The staff reviewed the applicant's description in T&QP Section 3.5.1 for the implementation of the site-specific physical protection program in accordance with NRC regulations and NUREG-0800 acceptance criteria. Since the applicant'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.D.1.(3) and is, therefore, acceptable.

Demonstration of Knowledge Skills and Abilities

The provisions of 10 CFR Part 73, Appendix B, Sections VI, A., B., C., D., (A.4, C.3(d), D.1(a), D.1(b)(2)) state, in part, that an individual must demonstrate required knowledge, skills and abilities, to carry out assigned duties and responsibilities.

T&QP Section 3.5.2 provides that all knowledge, skills and abilities will be demonstrated in accordance with a systematic approach to training (SAT) program as described in RG 5.75.

The staff reviewed the applicant's description in T&QP Section 3.5.2 for the implementation of the site-specific physical protection program in accordance with NRC regulations and NUREG-0800 acceptance criteria. Since the applicant'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, Sections VI.A, B, C, and D and is, therefore, acceptable.

Weapons Training and Qualification

General Firearms Training

The provisions of 10 CFR Part 73, Appendix B, Section VI.E provide that armed members of the security organization shall be trained and qualified in accordance with the requirements of this appendix and the NRC-approved T&QP. Training must be conducted by certified firearms instructors who shall be recertified at least every 3 years. Applicants shall conduct annual firearms familiarization, and armed members of the security organization must participate in weapons range activities on a nominal 4 month periodicity.

T&QP Section 3.6.1 of the T&QP addresses the requirements in 10 CFR Part 73, Appendix B, Sections 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. Each armed member of the security organization is trained and qualified by a certified firearms instructor for the use and maintenance of each assigned weapon to include but not limited to, marksmanship, assembly, disassembly, cleaning, storage, handling, clearing, loading, unloading, and reloading, for each assigned weapon.

The staff reviewed the applicant's description in T&QP Section 3.6.1 for the implementation of the site-specific physical protection program in accordance with NRC regulations and NUREG-0800 acceptance criteria. Since the applicant'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.E.1 and is, therefore, acceptable.

General Weapons Qualification

The provisions of 10 CFR Part 73, Appendix B, Section VI.F.1 Weapons Qualification and Requalification Program require that qualification firing must be accomplished in accordance with NRC requirements and the NRC-approved T&QP for assigned weapons. The results of weapons qualification and requalification must be documented and retained as a record.

T&QP Section 3.6.2 of the T&QP provides that all armed personnel are qualified and re-qualified with assigned weapons. All weapons qualification and re-qualification will be documented and retained as a record.

The staff reviewed the applicant's description in T&QP Section 3.6.2 for the implementation of the site-specific physical protection program in accordance with NRC regulations and NUREG-0800 acceptance criteria. Since the applicant'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.F.1, and is, therefore, acceptable.

Tactical Weapons Qualification

The provisions of 10 CFR Part 73, Appendix B, Section VI.F.2 require that the applicant conduct tactical weapons qualification. The applicant T&QP must describe the firearms used, the firearms qualification program, and other tactical training required to implement the NRC-approved security plans, applicant protective strategy, and implementing procedures. Applicant-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.

T&QP Section 3.6.3 provides that a tactical qualification course of fire is used to assess armed security force personnel in tactical situations to ensure they are able to demonstrate required tactical knowledge, skills and abilities remain proficient.

The staff reviewed the applicant's description in T&QP Section 3.6.3 for the implementation of the site-specific physical protection program in accordance with NRC regulations and

NUREG-0800 acceptance criteria. Since the applicant'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.F.2 and is, therefore, acceptable.

Firearms Qualification Courses

The provisions of 10 CFR Part 73, Appendix B, Section VI.F.3 state, in part, that the applicant 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 provisions of 10 CFR Part 73, Appendix B, Section VI.F.4 describe required courses of fire.

T&QP Section 3.6.4 provides a description of the firearms qualification scores for each courses of fire used to ensure armed members of the security organization are properly trained and qualified. Courses of fire are used individually for handguns, and semiautomatic rifles, and enhanced weapons.

The staff reviewed the applicant's description in T&QP Section 3.6.4 for the implementation of the site-specific physical protection program in accordance with NRC regulations and NUREG-0800 acceptance criteria. Since the applicant'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.F.3, and 10 CFR Part 73, Appendix B, Section VI.F.4 and is, therefore, acceptable.

Firearms Requalification

The provisions of 10 CFR Part 73, Appendix B, Section VI.F.5 provide that armed members of the security organization shall be re-qualified for each assigned weapon at least annually in accordance with NRC requirements and the NRC-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.

T&QP Section 3.6.5 describes that armed members of the security organization will re-qualify at least annually with each weapon assigned, using the courses of fire provided in the T&QP.

The staff reviewed the applicant's description in T&QP Section 3.6.5 for the implementation of the site-specific physical protection program in accordance with NRC regulations and NUREG-0800 acceptance criteria. Since the applicant'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.F.5 and is, therefore, acceptable.

Weapons, Personal Equipment and Maintenance

The provisions of 10 CFR Part 73, Appendix B, Section VI.G provide the requirements for weapons, personal equipment, and maintenance. These requirements provide that the applicant shall provide armed personnel with weapons that are capable of performing the function stated in the NRC-approved security plans, applicant protective strategy, and implementing procedures. In addition, the applicant shall 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, applicant protective strategy, and implementing procedures.

T&QP Section 3.7 describes that personnel are provided with weapons and personal equipment necessary to meet the plans and the protective strategy. The equipment provided is described in T&QP Section 9.0, and maintenance is performed as described in T&QP Section 20.0. The staff's review of T&QP Section 9, "Security Personnel Training," and T&QP Section 20, "Maintenance, Testing, and Calibration," is in Sections 13.6.4.1.9 and 13.6.4.1.20 of this report.

The staff reviewed the applicant's description in T&QP Section 3.7 for the implementation of the site-specific physical protection program in accordance with NRC regulations and NUREG-0800 acceptance criteria. Since the applicant'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.G and is, therefore, acceptable.

Documentation

The provisions of 10 CFR Part 73, Appendix B, Section VI.H require that the applicant shall retain all reports, records, or other documentation required by this appendix in accordance with the requirements of 10 CFR 73.55(r). The applicant shall retain each individual's initial qualification record for 3 years after termination of the individual's employment and shall retain each re-qualification record for 3 years after it is superseded. The applicant 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.

T&QP Section 3.8 provides that records are retained in accordance with T&QP Section 22 "Records." The T&QP also describes how the applicant will retain each individual's initial qualification record for 3 years after termination of the individual's employment and shall retain each re-qualification record for 3 years after it is superseded.

The staff reviewed the applicant's description in T&QP Section 3.8 for the implementation of the site-specific physical protection program in accordance with NRC regulations and NUREG-0800 acceptance criteria. Since the applicant'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.H and is, therefore, acceptable.

13.6.4.2.4 Performance Evaluation Program

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.6.4.2.4:

13.6.4.2.4 Performance Evaluation Program

10 CFR Part 73, Appendix B, Section VI.C.3, Performance Evaluation Program

(a) Applicants shall develop, implement and maintain a performance evaluation program that is documented in procedures, which describes how the applicant will demonstrate and assess the effectiveness of their onsite physical protection program and protective strategy, including the capability of the armed response team to carry out their assigned duties and responsibilities during safeguards contingency events. The performance evaluation program and procedures shall be referenced in the applicant's T&QP.

(b) *The performance evaluation program shall include procedures for the conduct of tactical response drills and force-on-force exercises designed to demonstrate and assess the effectiveness of the applicant's physical protection program, protective strategy and contingency event response by all individuals with responsibilities for implementing the SCP. The performance evaluation program must be designed to ensure, in part, that each member of each shift who is assigned duties and responsibilities required to implement the SCP and applicant protective strategy participates in at least one tactical response drill on a quarterly basis and one force-on-force exercise on an annual basis.*

Section 4 of the T&QP details the performance evaluation program consistent with the requirements of 10 CFR Part 73, Appendix B, Sections VI.C.3(a) through (m). Additional details of the performance evaluation program are described in the facility procedures.

The NRC staff has reviewed the applicant'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 the applicant'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.C.3 and is, therefore, acceptable.

13.6.4.2.5 Definitions

The provisions of 10 CFR Part 73, Appendix B, Section VI.J state, in part, that terms defined in 10 CFR Part 50, 10 CFR Part 70 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 considered SGI and has not been included in this SER.

The NRC staff has reviewed the applicant's description in 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 the applicant'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, and are, 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 target sets, 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 COL application and are, therefore, 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). 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.

13.6.4.3 Appendix C Safeguards Contingency Plan

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.6.4.3.1:

13.6.4.3.1 Background Information

This category of information identifies the perceived dangers and incidents that the plan addresses and a general description of how the response is organized.

<u>Purpose of the Safeguards Contingency Plan</u>

The provisions of 10 CFR Part 73, Appendix C, Section II.B.1.b state that the applicant should discuss general goals, objectives and operational concepts underlying the implementation of the SCP.

Section 1.1 of the SCP describes the purpose and goals of the SCP, including guidance to security and management for contingency events.

<u>Scope of the Safeguards Contingency Plan</u>

The provisions of 10 CFR Part 73, Appendix C, Section II.B.1.c delineate 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 into other site emergency response procedures.*

Section 1.2 of the SCP details the scope of the SCP 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 provisions of 10 CFR Part 73, Appendix C, Section II.B.1 require that, consistent with the DBT specified in 10 CFR 73.1(a)(1), the applicant 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.

The applicant adequately addresses perceived danger, provides a purpose of the plan, and describes the scope of the plan.

Definitions

Section 1.4 of the SCP describes 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 is found in Appendix A of the PSP.

The NRC staff has reviewed the applicant'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 the applicant's description in the SCP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description provided in the SCP meets the requirements of 10 CFR Part 73, Appendix C, Section II.B and are, therefore, acceptable.

13.6.4.3.2 Generic Planning Base

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.6.4.3.2:

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, to include 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 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 Contingency Plan

The provisions of 10 CFR Part 73, Appendix C, Section II.B.2.a require, 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 they are perceived initially by the applicant's personnel. Applicants 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, bomb threats, or other threat warnings—either verbal, such as telephoned threats, or implied, such as escalating civil disturbances.

The provisions of 10 CFR Part 73, Appendix C, Section II.B.2.b require, in part, that the plan define the specific objective to be accomplished relative to 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 provisions of 10 CFR Part 73, Appendix C, Section II.B.2.c require, in part, that the applicant 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 the applicant'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 the applicant's description in the SCP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description provided in the SCP meets the requirements of 10 CFR Part 73, Appendix C Section II.B.2 and are, therefore, acceptable.

13.6.4.3.3 Responsibility Matrix

The provisions of 10 CFR Part 73, Appendix C, Section II.B.4 state that this category of information consists of the detailed identification of responsibilities and specific actions to be taken by the applicant's organizations and/or personnel in response to safeguards contingency events. To achieve this result the applicant must address the following.

- The provisions of 10 CFR Part 73, Appendix C, Section II.B.4.a require, in part, that the applicant develop site procedures that consist of matrixes detailing the organization and/or personnel responsible for decisions and actions associated with specific responses to safeguards contingency events. The responsibility matrix and procedures must be referenced in the applicant's SCP.

- The provisions of 10 CFR Part 73, Appendix C, Section II.B.4.b require, in part, that the responsibility matrix procedures shall be based on the events outlined in the applicant's generic planning base and include specific objectives to be accomplished, description of responsibilities for decisions and actions for each event, and overall description of response actions each responding entity.

- The provisions of 10 CFR Part 73, Appendix C, Section II.B.4.c require, in part, that responsibilities are to be assigned in a manner that precludes conflict of duties and responsibilities that would prevent the execution of the SCP and emergency response plans.
- The provisions of 10 CFR Part 73, Appendix C, Section II.B.4.d require, in part, that the applicant ensure that predetermined actions can be completed under the postulated conditions.

SCP Section 3 includes the responsibility matrix, as required by 10 CFR Part 73, Appendix C, Section II.B.4.a. The responsibility matrix integrates the response capabilities of the security organization (described in SCP Section 4) with the background information relating to decision/actions and organizational structure (described in SCP Section 1) as required by 10 CFR Part 73, Appendix C, Section II.B.4.a. The responsibility matrix provides an overall description of the response actions and their interrelationships, as required by 10 CFR Part 73, Appendix C, Section II.B.4.a. 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 as required by 10 CFR Part 73, Appendix C, Section II.B.4.a. In part, the applicant shall ensure that predetermined actions can be completed under the postulated conditions as required by 10 CFR Part 73, Appendix C, Section II.B.4.d.

The staff reviewed the applicant's description in SCP Section 3 for the implementation of the site-specific physical protection program in accordance with NRC regulations and NUREG-0800 acceptance criteria. Since the applicant's description in the SCP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description provided 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 provisions of 10 CFR Part 73, Appendix C, Section II.B.3 require, in part, that the applicant planning base include factors affecting the SCP specific for each facility.

Licensee Organization

The provisions of 10 CFR Part 73, Appendix C, Section II.B.3.a require 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/Communication Protocols

SCP Section 4.1.1 details the duties and communications protocols of each member of the security organization responsible for implementing any portion of the applicant's protective strategy, which will allow for coordination and maintenance of command and control functions as required by 10 CFR Part 73, Appendix C, Section II.B.3.a.

Security Chain of Command/Delegation of Authority

SCP Section 4.1.2 P describes in detail the chain of command and delegation of authority during normal operations is discussed in the PSP. The chain of command and delegation of authority during contingency events is also described in the responsibility matrix portions of the SCP. The chain of command and delegation of authority during normal operations is discussed in the PSP. Accordingly, the staff concludes that the applicant has described the chain of command and delegation of authority during contingency events as required by 10 CFR Part 73, Appendix C, Section II.B.3.a.

Physical Layout

The provisions of 10 CFR Part 73, Appendix C, Section II.B.3(b) require, 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 have an effect upon 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.

SCP Section 4.2 references PSP Section 1.1 for layouts of the OCA, PA, vital areas, site maps, and descriptions of site features. The staff confirmed that these layouts, maps, and descriptions include the detailed information required by 10 CFR Part 73, Appendix C, Section II.B.3.b and described above.

Safeguards Systems

The provisions of 10 CFR Part 73, Appendix C, Section II.B.3.c require, in part, that the SCP include a description of the physical security systems that support and influence how the applicant will respond 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.

SCP Section 4.3 describes that safeguards systems are described in PSP Sections 9, 11, 12, 13, 15 and 16, and in facility implementing procedures/documents. SCP Section 8 describes how physical security systems will be used to respond to a threat at the site, as required by 10 CFR Part 73, Appendix C, Section II.B.3.c. As further required by 10 CFR Part 73, Appendix C, Section II.B.3.c, the SCP description begins with physical protection measures proposed at the outermost facility perimeter, and moves inward through those measures proposed protect target set equipment.

Law Enforcement Assistance

The provisions of 10 CFR Part 73, Appendix C, Section II.B.3.d require, in part, that the applicant provide a listing of available law enforcement agencies and 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.

SCP Section 4.4 states in detail the role of LLEA in the site protective strategy. In accordance with 10 CFR Part 73, Appendix C, Section II.B.3.d, these details include LLEA response capabilities, LLEA criteria for response, and the working agreements or arrangements for communicating with these LLEAs. Additional details regarding LLEA are included in PSP Section 8 and SCP Section 5.6.

Policy Constraints and Assumptions

The provisions of 10 CFR Part 73, Appendix C, Section II.B.3.e require, in part, that the SCP include a discussion of State laws, local ordinances, and company policies and practices that govern applicant response to incidents and must include, but is not limited to, the following: (1) use of deadly force; (2) recall of off-duty employees; (3) site jurisdictional boundaries; and (4) use of enhanced weapons, if applicable.

SCP Section 4.5 details the site security policies, including the use of deadly force provisions for the recall of off-duty employees, site jurisdictional boundaries, and authority to request offsite assistance, as required by 10 CFR Part 73, Appendix C, Section II.B.3.e.

Administrative and Logistical Considerations

The provisions of 10 CFR Part 73, Appendix C, Section II.B.3.f require, in part, that the applicant provide descriptions of applicant practices, which influence how the security organization responds to a safeguards contingency event to include, but is not limited to, a description of 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.

SCP Section 4.6 outlines administrative duties of the Security Manager, Security Shift Team Leader, facility procedures and administrative forms.

The staff reviewed the applicant’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 NRC regulations and NUREG-0800 acceptance criteria. Since the applicant’s description in the SCP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description provided 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 outlines the response by the applicant to threats to the facility. As set forth below, the applicant described in detail how they protect against the DBT with onsite and offsite organizations, consistent with the regulations in 10 CFR 50.54(p)(1) and (hh)(1), 10 CFR 73.55(k), 10 CFR Part 73, Appendix B, Section VI and 10 CFR Part 73, Appendix C, Section II.B.3. In addition, 10 CFR Part 73, Appendix C, “Introduction,” states, in part, it is important to note that an applicant’s SCP is intended to be complementary to any emergency plans developed pursuant to 10 CFR Part 50, Appendix E, “Emergency Planning and Preparedness for Production and Utilization Facilities”; and 10 CFR 52.79, “Contents of Applications; Technical Information and Final Safety Analysis Report.”

Response to Threats

SCP Section 5.1 describes how 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

SCP Section 5.2 notes individuals from the Responsibility Matrix and their role in the site protective strategy. This section also notes the minimum number of individuals and their contingency equipment for implementation of the protective strategy. The applicant described the armed response team consistent with 10 CFR 73.55(k)(4), (5), (6), and (7); 10 CFR Part 73, Appendix B, Section VI; and 10 CFR Part 73, Appendix C, Section II.B.3.

Supplemental Security Officer

SCP Section 5.3 describes in detail the role of supplemental security officers in the site protective strategy. The applicant described the use of supplemental security officers, consistent with the requirements in 10 CFR 73.55(k)(4).

Facility Operations Response

SCP Section 5.4 details the role of operations personnel in the applicant's protective strategy, including responsibilities, strategies, and conditions for operator actions as discussed in 10 CFR 50.54(hh).

Emergency Plan Response

SCP Section 5.5 notes the integration of the 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 (LLEA)

SCP Section 5.6 documents the current agreements with applicable LLEA, and therefore 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. Details on the response of the LLEA are located in PSP Section 8.

State Response Agencies

SCP Section 5.7 documents the current agreements with applicable LLEA and, therefore, 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. Furthermore, SCP Section 5.7 provides a general description of the LLEA response capability and meets the corresponding portions of 10 CFR 73.55(k)(9).

Federal Response Agencies

SCP Section 5.8 documents the current agreements with applicable LLEA and, therefore, 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. Furthermore. SCP Section 5.7 provides a general description of the LLEA response capability and meets the corresponding portions of 10 CFR 73.55(k)(9).

Response to ISFSI Events

SCP Section 5.9 meets the requirements of 10 CFR 73.55(k)(9) and 10 CFR Part 73, Appendix C, Section II.B.3.d describes the Response Requirements for ISFSI as a part of the protective strategy.

The staff reviewed the applicant's description in SCP Sections 5.0 through 5.9 for the implementation of the site-specific physical protection program in accordance with NRC regulations and NUREG-0800 acceptance criteria. Since the applicant's description in the SCP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description provided in the SCP meets the requirements of 10 CFR 50.54(p)(1) and (hh); 10 CFR 73.55(k); 10 CFR Part 73, Appendix B, Section VI; and 10 CFR Part 73, Appendix C, Section II.B.3 and is, therefore, acceptable. In addition, Appendix C, "Introduction," states, in part, that it is important to note that an applicant's SCP is intended to be complementary to any emergency plans developed pursuant to 10 CFR Part 50, Appendix E and 10 CFR 52.17.

13.6.4.3.6 Defense in Depth

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.6.4.3.6:

> ***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 the applicant'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 the applicant's description in the SCP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description provided 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

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.6.4.3.7:

> *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 the applicant'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 the applicant's description in the SCP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description provided in the SCP meets the requirements of 10 CFR 10 CFR 73.55(b) and is, therefore, acceptable.

13.6.4.3.8 Protective Strategy

The provisions of 10 CFR Part 73, Appendix C, Section II.B.3.c(v) require that applicants 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) include measures that limit the exposure of security personnel to possible attack; (4) include a description of 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.

SCP Section 8 describes the site protective strategy.

The staff reviewed the applicant's description in SCP Section 8 for the implementation of the site-specific physical protection program in accordance with NRC regulations and NUREG-0800 acceptance criteria. Since the applicant's description in the SCP is consistent with the acceptance criteria in NUREG-0800, Section 13.6.1, the staff finds that the description provided 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

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.6.4.3.9:

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, in accordance with the DBT of radiological sabotage as stated in 10 CFR 73.1. The target sets, 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 COL application and are, therefore, 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). 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.

13.6.5 Post Combined License Activities

For the reasons discussed in the technical evaluation section above, the staff finds the following license condition acceptable:

- License Condition (13-12) – No later than 12 months after issuance of the COL, the licensee shall submit to the Director of NRO, a schedule that supports planning for and conduct of NRC inspection of the physical security programs. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until the physical security program has been fully implemented.

13.6.6 Conclusion

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to physical security, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff concludes that the relevant information presented in the WLS COL FSAR is acceptable based on the applicable regulations specified in Section 13.6.4 of this report. The staff based its conclusion on the following:

- STD COL 13.6-1, as related to the physical protection program, is acceptable based on the following discussion. The staff's review of the WLS Units 1 and 2 PSP, T&QP, and SCP focused on ensuring the necessary programmatic elements are included in these plans 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.

  The staff determined that these plans include the necessary programmatic elements that, when effectively implemented, will provide the required high assurance. The burden to effectively implement these plans remains with the applicant. Effective implementation is dependent on the procedures and practices the applicant develops to satisfy the programmatic elements of its PSP, T&QP, and SCP. The target set analysis and site protective strategy are in facility implementing procedures which were not subject to staff review as part of this COL application and are, therefore, 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 provided by the applicant's PSP Section 3, a performance evaluation program will be implemented that periodically tests and evaluates the effectiveness of the overall protective strategy. This program provides that deficiencies be corrected. In addition, NRC inspectors will conduct periodic force-on-force exercises that will test the effectiveness of the applicant's protective strategy. Based on the results of the applicant's own testing and evaluation, NRC baseline inspections and force-on-force exercises, enhancements to the applicant's PSP, T&QP, and SCP may be necessary to ensure the overall protective strategy can be effectively implemented. As such, the staff approval of the applicant'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 with the programmatic elements of these plans as a result of the periodic applicant or NRC conducted drills or exercises that test the effectiveness of the overall protective strategy, the applicant shall correct the plans to address these deficiencies in a timely manner and the applicant should 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.”

The COL applicant’s security plan information is withheld from public disclosure in accordance with the provisions of 10 CFR 73.21.

13.6.A Site-Specific ITAAC for Physical Security

This section does not exist in either the AP-1000 DCD or the WLS COL FSAR. The staff has added this section to the SER in order to address issues regarding the applicant’s site-specific ITAAC for physical security.

13.6.A.1 Introduction

WLS COLA Part 10, “Proposed License Conditions and ITAAC,” Appendix B, “Inspections, Tests, Analysis, and Acceptance Criteria” describes the license conditions for the plant’s physical protection systems or features to provide physical protection of the site-specific protective strategy and elements of a site security program. The COL application incorporates by reference AP1000 DCD Tier 1 Section 2.6.9, including plant layout and configurations of barriers, and lists ITAAC related to the site-specific design for achieving detection, assessment, communications, delay, and response for physical protection against potential acts of radiological sabotage and theft of special nuclear material.

The design bases or supporting security analyses and assumptions related to the design descriptions of security-related features incorporated by reference from the AP1000 DCD are in TR-94, APP-GW-GLR-066. Descriptions of site-specific security structures, programs and contingency measures are in the WLS PSP, which includes the site PSP, T&QP, and the SCP.

13.6.A.2 Summary of Application

WLS COL FSAR, Revision 11, Section 14.3 incorporates by reference AP1000 DCD, Revision 19, Section 14.3. WLS COLA, Part 10, incorporates by reference AP1000 DCD Tier 1, Section 2.6.9, which includes the physical security-inspections, tests, analyses, and acceptance criteria (PS-ITAAC) that are within the scope of the AP1000 standard design. Site-specific PS-ITAAC that are outside the scope of AP1000 DCD Tier 1, Section 2.6.9 are provided in WLS COLA Part 10, Appendix B, Table 2.6.9-2.

In addition, in WLS COL FSAR Section 14.3, the applicant provided the following:

Supplemental Information

- STD SUP 14.3-1

The applicant provided supplemental information related to physical security in STD SUP 14.3-1 in WLS COL FSAR Section 14.3.2.3.2.

License Condition

- Part 10, License Condition 1

The applicant provided a license condition in WLS COLA, Revision 9, Part 10, which will incorporate the ITAAC identified in the tables in WLS COLA, Part 10, Appendix B. The staff evaluated this license condition in Chapter 1 of this report.

13.6.A.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of NRC regulations are given in 10 CFR Part 73. The regulation includes specific security and performance requirements that, when adequately implemented, are designed to protect nuclear power reactors against acts of radiological sabotage, prevent the theft or diversion of special nuclear material, and protect safeguards information against unauthorized release.

The provisions of 10 CFR 52.80, Subpart A require that information submitted for a COL include the proposed ITAAC that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the ITAAC are met, the facility has been constructed and will operate in conformity with the COL, the provisions of the Atomic Energy Act of 1954, as amended, and NRC regulations.

The WLS Units 1 and 2 design descriptions, commitments, and acceptance criteria for the security features, including the plant's layout and determination of vital equipment and areas, for a certified design are based on physical protection systems or hardware provided for meeting requirements of the following NRC regulations:

- 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities"10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants"10 CFR 73.1(a)(1), "Radiological Sabotage"
- 10 CFR 73.55, "Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage," Appendix B, "General Criteria for Security Personnel"; Appendix C, "Nuclear Power Plant Safeguards Contingency Plans"; Appendix G, "Reportable Safeguards Events"; and Appendix H, "Weapons Qualification Criteria"
- 10 CFR Part 74, "Material control and accounting of special nuclear material"
- 10 CFR 100.21(f), "Non-seismic siting criteria"

Regulatory requirements and acceptance criteria related to physical protection systems or hardware are identified in NUREG-0800, Section 14.3.12.

Regulatory guidance documents that are applicable to this evaluation are:

- RG 1.91, “Evaluations of Explosions Postulated to Occur at Transportation Routes Near Nuclear Power Plants,” Revision 1
- RG 1.206, “Combined License Applications for Nuclear Power Plants”
- RG 4.7, “General Site Suitability Criteria for Nuclear Power Stations,” Revision 2
- RG 5.7, Entry/Exit Control for Protected Areas, Vital Areas, and Material Access Areas,” Revision 1
- RG 5.12, “General Use of Locks in the Protection and Control of Facilities and Special Nuclear Materials”
- RG 5.44, “Perimeter Intrusion Alarm Systems,” Revision 3
- RG 5.62, “Reporting of Safeguards Events,” Revision 1
- RG 5.65, “Vital Area Access Controls, Protection of Physical Protection System Equipment and Key and Lock Controls”
- RG 5.66, “Access Authorization Program for Nuclear Power Plants”
- Information Notice 86-83, “Underground Pathways into Protected Areas, Vital Areas, and Controlled Access Areas,” September 19, 1986
- Regulatory Information Summary (RIS) 2005-04, “Guidance on the Protection of Unattended Openings that Intersect a Security Boundary or Area,” April 14, 2005. (Exempt from public disclosure in accordance with 10 CFR 2.390, “Public inspections, exemptions, requests for withholding.”)

The COL applicant is required to describe commitments for establishing and maintaining a physical protection system (engineered and administrative controls), organization, programs, and procedures for implementing a site-specific strategy that, if adequately implemented, provide high assurance for protection of the plant against the DBT. The site-specific physical protection system described must be reliable and available and implement the concept of defense-in-depth protection in order to provide a high assurance of protection. The security operational programs and the physical protection system are required to meet the specific performance requirements of 10 CFR Part 26, “Fitness for Duty Programs”; 10 CFR 73.54, “Protection of Digital Computer and Communication Systems and Networks”; 10 CFR 73.55; 10 CFR 73.56, “Personnel access authorization requirements for nuclear power plants”; 10 CFR 73.57, “Requirements for criminal history records checks of individuals granted unescorted access to a nuclear power facility or access to Safeguards Information”; and 10 CFR 73.58. Physical protection hardware within the scope of the AP1000 design is addressed in the AP1000 DCD.

13.6.A.4 Technical Evaluation

The staff reviewed WLS COL FSAR Section 14.3 and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to ITAAC for physical security. The results of the staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this report provides a discussion of the strategy used by the staff to perform one technical review for each standard issue outside the scope of the DC and use this review to evaluate subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from responses to RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff completed its review and concluded that the evaluation performed for the standard content is directly applicable to the WLS COL application. This standard content material is identified in this report by use of italicized, double-indented formatting. The staff confirmed that the July 6, 2011, WLS Letter No. 097 contained the same technical information provided in the June 11, 2010, VEGP letter discussed in the standard content material below.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.6.A.4:

> *Supplemental Information*
>
> - *STD SUP 14.3-1*
>
> *STD SUP 14.3-1 adds the following after DCD Section 14.3.2.2 as new Section 14.3.2.3.2:*
>
> > *Generic PS-ITAAC have been developed in a coordinated effort between the NRC and the Nuclear Energy Institute (NEI) as outlined in Appendix C.II.I-C of Regulatory Guide 1.206. These generic ITAAC have been tailored to the AP1000 design and site-specific security requirements.*
>
> *In Part 10, Appendix B of the VEGP Units 3 and 4 COL application, SNC describes the ITAAC for the plant's physical protection systems or features to provide physical protection of the site-specific protective strategy and elements of a site security program. The COL application incorporates by reference Tier 1 Section 2.6.9 of the AP1000 DCD, including plant layout and configurations of barriers, and listed ITAAC related to the site-specific design for achieving detection, assessment, communications, delay, and response for physical*

protection against potential acts of radiological sabotage and theft of special nuclear material. DCD Tier 1 Section 2.6.9 includes the physical security ITAAC that are in the scope of the AP1000 standard design. Site-specific physical security ITAAC that are outside the scope of AP1000 DCD Tier 1 Section 2.6.9 are provided in Table 2.6.9-2 of Appendix B to Part 10 of the VEGP COL application.

The NRC staff's evaluation of the PS-ITAAC (STD SUP 14.2-1) is documented in the Sections 13.6.A.4.1 through 13.6.A.4.3 of this SER.

13.6.A.4.1 Detection and Assessment Hardware

The applicant submitted the following ITAAC for detection and assessment hardware in their letter dated June 11, 2010, "Response to Request for Additional Information Letter No. 047, Supplement 2, Physical Security Inspections, Tests, Analyses, and Acceptance Criteria," This letter was used to complete the evaluation below.

1. *The external walls, doors, ceiling, and floors in the location within which the last access control function for access to the protected area is performed are bullet resistant to at least Underwriters Laboratory Ballistic Standard 752, Level 4. (Item 6 in Appendix A to Section 14.3.12 of NUREG-0800.)*

2. *Physical barriers for the protected area perimeter are not part of vital area barriers. (Item 2.a in Appendix A to Section 14.3.12 of NUREG-0800.)*

3.

    a) *Isolation zones exist in outdoor areas adjacent to the physical barrier at the perimeter of the protected area that allows 20 feet of observation on either side of the barrier. (Item 3.a in Appendix A to Section 14.3.12 of NUREG-0800.)*

    b) *Where permanent buildings do not allow a 20-foot observation distance on the inside of the protected area, the building walls are immediately adjacent to, or an integral part of, the protected area barrier. (Item 3.c in Appendix A to Section 14.3.12 of NUREG-0800.) The isolation zones are monitored with intrusion detection equipment that provides the capability to detect and assess unauthorized persons. (Item 3.b in Appendix A to Section 14.3.12 of NUREG-0800.)*

4. *The intrusion detection and assessment equipment at the protected area perimeter:*

    a) *Detects penetration or attempted penetration of the protected area barrier and concurrently alarms in both the Central Alarm Station and Secondary Alarm Station. (Item 4.a in Appendix A to Section 14.3.12 of NUREG-0800.)*

b) *The intrusion detection and assessment equipment at the protected area perimeter remains operable from an uninterruptible power supply in the event of the loss of normal power. (Item 4.c in Appendix A to Section 14.3.12 of NUREG-0800.)*

6. *An access control system with numbered picture badges is installed for use by individuals who are authorized access to protected areas without escort. (Item 9 in Appendix A to Section 14.3.12 of NUREG-0800*.)

8.

a) *Penetrations through the protected area barrier are secured and monitored. (Item 2.b in Appendix A to Section 14.3.12 of NUREG-0800.)*

b) *Unattended openings (such as underground pathways) that intersect the protected area boundary or vital area boundary will be protected by a physical barrier and monitored by intrusion detection equipment or provided surveillance at a frequency sufficient to detect exploitation. (Item 2.c in Appendix A to Section 14.3.12 of NUREG-0800.)*

On the basis of its review the NRC staff determined that the applicant has adequately revised Table 2.6.9-2 for Part 10 to the VEGP COL application PS-ITAAC items 2(a), 2(b), 2 (c), 3(a), 3(b), 3(c), 4(a), 4(c), 6(partially), and 9 identified in Appendix A to Section 14.3.12 of NUREG-0800.

The VEGP COL application references the AP1000 DCD, which addressed NUREG-0800, Section 14.3.12 PS-ITAAC 4(b), 5, 6 (partially), 10, 11(a), 11(b), 11(c) and 14. The staff has determined that PS-ITAAC 6, described in NUREG-0800, Section 14.3.12 has been fully addressed between the VEGP submission and the AP1000 DCD.

In a supplemental response to RAI 14.3.12-1, the applicant stated:

The information contained in SRP ITAAC number 11(d) is redundant to existing ITAAC in the AP1000 Design Certification Document (DCD). AP1000 DCD security ITAAC numbers 1, 4, 5(a), 5(b), 5(c), 13(a), 13(b), 13(c), and 15(b) demonstrate that the central and secondary alarm stations are equal and redundant, by being constructed, located, protected, and equipped to the standards for the central alarm station.

In RAI SRP 14.3.12-NSIR-7, Revision 1, Westinghouse stated:

No corresponding ITAAC has been provided for SRP 14.3.12 ITAAC number 11(d). The information contained in SRP ITAAC number 11(d) is redundant to existing ITAACs. AP1000 security ITAAC numbers 1, 4, 5(a), 5(b), 5(c), 13, and 15(b) demonstrate that the central and secondary alarm stations are constructed, located, protected, and equipped to the standards for the central alarm station.

On the basis of its review, the NRC staff determined that the applicant has adequately shown that NUREG-0800, Section 14.3.12 detection and assessment hardware ITAAC 11(d) is addressed.

13.6.A.4.2 Delay or Barrier Design

The applicant submitted the following ITAAC for Delay or Barrier Design in their "Response to Request for Additional Information Letter No. 047, Supplement 2, Physical Security Inspections, Tests, Analyses, and Acceptance Criteria," Dated June 11, 2010. This letter was used to complete the evaluation below.

5. *Access control points are established to:*

    a) *Control personnel and vehicle access into the protected area. (Item 8.a in Appendix A to Section 14.3.12 of NUREG-0800.)*

    b) *Detect firearms, explosives, and incendiary devices at the protected area personnel access points. (Item 8.b in Appendix A to Section 14.3.12 of NUREG-0800.)*

7. *Access to vital equipment physical barriers requires passage through the protected area perimeter barrier. (Item 1.b in Appendix A to Section 14.3.12 of NUREG-0800.)*

On the basis of its review, the NRC staff determined that the applicant has adequately addressed NUREG-0800, Section 14.3.12 delay or barrier design PS-ITAAC 1(b)(partially),8(a) and 8(b).

The VEGP COL application references the AP1000 DCD, which addressed NUREG-0800, Section 14.3.12 PS-ITAAC 1(a), 1(b)(partially), 7, 13(a) and 13(b). The staff has determined that PS-ITAAC 1(b) described in NUREG-0800, Section 14.3.12 has been fully addressed between the VEGP submission and the AP1000 DCD.

13.6.A.4.3 Systems, Hardware, or Features Facilitating Security Response and Neutralization

The applicant submitted the following ITAAC for Systems, Hardware, or Features Facilitating Security Response and Neutralization in their "Response to Request for Additional Information Letter No. 047, Supplement 2, Physical Security Inspections, Tests, Analyses, and Acceptance Criteria," Dated June 11, 2010. This letter was used to complete the evaluation below.

9. *Emergency exits through the protected area perimeter are alarmed and secured with locking devices to allow for emergency egress. (Item 15 in Appendix A to Section 14.3.12 of NUREG-0800.)*

On the basis of its review, the NRC staff determined that the applicant has adequately addressed NUREG-0800, Section 14.3.12 delay or barrier design PS-ITAAC 15(partially).

The VEGP COL application references the AP1000 DCD, which addressed NUREG-0800, Section 14.3.12 PS-ITAAC 12, 15(partially) 16(a), 16(b) and 16(c). The staff has determined that PS-ITAAC 15 described in NUREG-0800, Section 14.3.12 has been fully addressed between the VEGP submission and the AP1000 DCD.

On the basis of its review, the staff finds that since the applicant revised WLS COL FSAR Part 10 to incorporate the requirements for PS-ITAAC, the response to RAI 14.03.12-1, 2 and 3 has adequately addressed NUREG-0800, Section 14.3.12, and is, therefore, acceptable.

13.6.A.5 Post Combined License Activities

For the reasons discussed in the technical evaluation section above, the staff proposes to include the following ITAAC for physical security:

- The licensee shall perform and satisfy the ITAAC defined in Table 13.6A-1, "Site Specific Physical Security."

13.6.A.6 Conclusion

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to PS-ITAAC, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff concludes that the relevant information presented in WLS COL FSAR and the additional information received in the July 6, 2011, letter is acceptable based on the applicable regulations specified in Section 13.6.A.4 of this report. The staff based its conclusion on the following:

- STD SUP 14.3-1, as related to PS-ITAAC, is acceptable based on the following discussion. The staff finds that the applicant adequately describes the physical security systems or provides and/or facilitates the implementation of the site-specific protective strategy and security programs. The applicant adequately describes the site-specific PS-ITAAC for meeting the requirements of 10 CFR 73.55 and provides the technical bases for establishing a PS-ITAAC for the protection against acts of radiological sabotage and theft of special nuclear material. The applicant includes systems and features as stated in WLS COL FSAR Chapter 13 and referenced TRs. The applicant provided adequate descriptions of objectives, prerequisites, test methods, data required, and acceptance criteria for security related ITAAC for the approval of the WLS COL.

Table 13.6.A 1 – Site Specific Physical Security Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, and Analyses	Acceptance Criteria
1. The external walls, doors, ceiling, and floors in the location within which the last access control function for access to the protected area is performed are bullet- resistant to at least Underwriters Laboratory Ballistic Standard 752, level 4.	Type test, analysis, or a combination of type test and analysis will be performed for the external walls, doors, ceilings, and floors in the location within which the last access control function for access to the protected area is performed.	The external walls, doors, ceilings, and floors in the location within which the last access control function for access to the protected area is performed are bullet- resistant to at least Underwriters Laboratory Ballistic Standard 752, level 4.
2. Physical barriers for the protected area perimeter are not part of vital area barriers.	An inspection of the protected area perimeter barrier will be performed.	Physical barriers at the perimeter of the protected area are separated from any other barrier designated as a vital area barrier.
3. a) Isolation zones exist in outdoor areas adjacent to the physical barrier at the perimeter of the protected area that allows 20 feet of observation on either side of the barrier. Where permanent buildings do not allow a 20-foot observation distance on the inside of the protected area, the building walls are immediately adjacent to, or an integral part of, the protected area barrier.	Inspections will be performed of the isolation zones in outdoor areas adjacent to the physical barrier at the perimeter of the protected area.	Isolation zones exist in outdoor areas adjacent to the physical barrier at the perimeter of the protected area and allow 20 feet of observation and assessment of the activities of people on either side of the barrier. Where permanent buildings do not allow a 20-foot observation and assessment distance on the inside of the protected area, the building walls are immediately adjacent to, or an integral part of, the protected area barrier and the 20-foot observation and assessment distance does not apply.

b) The isolation zones are monitored with intrusion detection equipment that provides the capability to detect and assess unauthorized persons.	Inspections will be performed of the intrusion detection equipment within the isolation zones.	The isolation zones are equipped with intrusion detection equipment that provides the capability to detect and assess unauthorized persons.
4. The intrusion detection and assessment equipment at the protected area perimeter: a) detects penetration or attempted penetration of the protected area barrier and concurrently alarms in both the central alarm station and secondary alarm station, and b) remains operable from an uninterruptible power supply in the event of the loss of normal power.	Tests, inspections or a combination of tests and inspections of the intrusion detection and assessment equipment at the protected area perimeter and its uninterruptible power supply will be performed.	The intrusion detection and assessment equipment at the protected area perimeter: a) detects penetration or attempted penetration of the protected area barrier and concurrently alarms in the central alarm station and secondary alarm station, and b) remains operable from an uninterruptible power supply in the event of the loss of normal power.

5. Access control points are established to: a) control personnel and vehicle access into the protected area. b) detect firearms, explosives, and incendiary devices at the protected area personnel access points.	Tests, inspections, or combination of tests and inspections of installed systems and equipment at the access control points to the protected area will be performed.	The access control points for the protected area: a) are configured to control personnel and vehicle access. b) include detection equipment that is capable of detecting firearms, incendiary devices, and explosives at the protected area personnel access points.
6. An access control system with numbered picture badges is installed for use by individuals who are authorized access to protected areas and vital areas without escort.	A test of the access control system with numbered picture badges will be performed.	The access authorization system with numbered picture badges can identify and authorize protected area and vital area access only to those personnel with unescorted access authorization.
7. Access to vital equipment physical barriers requires passage through the protected area perimeter barrier.	Inspection will be performed to confirm that access to vital equipment physical barriers requires passage through the protected area perimeter barrier.	Vital equipment is located within a protected area such that access to vital equipment physical barriers requires passage through the protected area perimeter barrier.

8. a) Penetrations through the protected area barrier are secured and monitored.	Inspections will be performed of penetrations through the protected area barrier.	Penetrations and openings through the protected area barrier are secured and monitored.
b) Unattended openings (such as underground pathways) that intersect the protected area boundary or vital area boundary will be protected by a physical barrier and monitored by intrusion detection equipment or provided surveillance at a frequency sufficient to detect exploitation.	Inspections will be performed of unattended openings that intersect the protected area boundary or vital area boundary.	Unattended openings (such as underground pathways) that intersect the protected area boundary or vital area boundary are protected by a physical barrier and monitored by intrusion detection equipment or provided surveillance at a frequency sufficient to detect exploitation.
9. Emergency exits through the protected area perimeter are alarmed and secured with locking devices to allow for emergency egress.	Tests, inspections, or a combination of tests and inspections of emergency exits through the protected area perimeter will be performed.	Emergency exits through the protected area perimeter are alarmed and secured by locking devices that allow prompt egress during an emergency.

13.7 Fitness for Duty

13.7.1 Introduction

Pursuant to 10 CFR 52.79(a)(44), COL applications must include a description of the fitness for duty program required by 10 CFR Part 26, “Fitness for Duty Programs,” and its implementation. The FFD program is designed to provide reasonable assurance that: (1) individuals are trustworthy and reliable as demonstrated by the avoidance of substance abuse; (2) individuals are not under the influence of any substance, legal or illegal, or mentally or physically impaired from any cause, which in any way adversely affects their ability to safely and competently perform their duties; (3) measures are established and implemented for the early detection of individuals who are not fit to perform their duties; (4) the construction site is free from the presence and effects of illegal drugs and alcohol; (5) the work places are free from the presence and effects of illegal drugs and alcohol; and (6) the effects of fatigue and degraded alertness on an individual's ability to safely and competently perform his or her duties are managed commensurate with maintaining public health and safety.

13.7.2 Summary of Application

WLS COL FSAR Section 13.7 is a new section added after AP1000 DCD Section 13.6. The references that are currently in AP1000 DCD Section 13.7 have been redistributed to other WLS COL FSAR sections. There is no information associated with the FFD program incorporated by reference from the AP1000 DCD.

In addition, in WLS COL FSAR Section 13.7, the applicant provided the following:

Supplemental Information

- STD SUP 13.7-1

The applicant provided standard supplemental information in WLS COL FSAR Section 13.7 describing the FFD program for both the construction phase and the operating phase of the units. The construction phase program will be consistent with NEI 06-06, "Fitness for Duty Program Guidance for New Nuclear Power Plant Construction Sites," and the construction phase program will be implemented prior to onsite construction of safety- and security-related SSCs. The operations phase program will be consistent with 10 CFR Part 26.

License Conditions

- Part 10, License Condition 6

The applicant proposed a license condition to provide a schedule to support NRC inspection of operational programs included in WLS COL FSAR Table 13.4-201 including the FFD program.

13.7.3 Regulatory Basis

The applicable regulatory requirements for STD SUP 13.7-1 are as follows:

- 10 CFR Part 26, "Fitness for Duty Programs"
- 10 CFR 52.79(a)(44)

13.7.4 Technical Evaluation

The staff reviewed WLS COL FSAR Section 13.7 to ensure that the COL application represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application addresses the required information relating to the FFD program.

Section 1.2.3 of this report provides a discussion of the process used by the staff to perform one technical review for each standard issue outside the scope of the DC and use this review to evaluate subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from responses to RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff completed its review and concluded that the evaluation performed for the standard content is directly applicable to the WLS COL application. This standard content material is identified in this report by use of italicized, double-indented formatting. Instead of confirming that all responses to RAIs identified in the corresponding standard content evaluation were endorsed by the WLS applicant (which is a typical step when comparing the two applications), the staff provides its evaluation of similar RAIs issued to WLS, following the standard content material. The one confirmatory item in the standard content material retains the number assigned in the VEGP SER, and is also addressed following the standard content material.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.7.4:

> *Supplemental Information*
>
> - *STD SUP 13.7-1*
>
> *The applicant provided a new Section 13.7 in the VEGP COL FSAR describing the FFD program. STD SUP 13.7-1 added the following text to Section 13.7:*
>
>> *The Fitness for Duty (FFD) Program (Program) is implemented and maintained in two phases; the construction phase program and the operating phase program. The construction and operations phase programs are implemented as identified in* [FSAR] *Table 13.4-201.*
>>
>> *The construction phase program is consistent with NEI 06-06 (*[FSAR] *Reference 201). The workforce population subject to random testing during construction is determined on a weekly basis by averaging the total number of active construction badges over each preceding seven-day period. The random selection from each week's workforce population is identified by a standard computer-generated random number generator using this number of active badges as the range of numbers considered in the weekly random testing selection.*
>>
>> *The operations phase program is consistent with 10 CFR Part 26.*

The staff notes that Reference 201 in the above text refers to Revision 4 of NEI 06-06.

The NRC staff's review of STD SUP 13.7-1 included the following: (1) the adequacy of the FFD program for the construction phase; (2) the adequacy of the FFD program for the operations phase; and (3) the implementation schedule proposed by the applicant for both the construction phase and operations phase FFD operational programs.

The NRC staff issued three RAIs to obtain further clarification on the applicant's FFD Program. The first two RAIs discussed below are associated with the resolution of STD SUP 13.7-1.

In RAI 13.6-33, the staff asked how the applicant intends to update its FFD program for the construction phase. NEI 06-06 provides examples of the FFD program that is required and, if this guidance is endorsed by the NRC, will provide an acceptable method of complying with the NRC's regulations. If the NRC endorses NEI 06-06, does the applicant intend to update its FFD program for the construction phase to comply with NEI 06-06? If future revisions to NEI 06-06 are endorsed by the NRC, does the applicant intend to update its FFD program for the construction phase to comply with certain clarifications, additions, and exceptions in these future, endorsed revisions, as necessary?

The applicant replied that it submitted an FFD Program for NRC approval as part of the Limited Work Authorization (LWA) request, and that the program is now being implemented as part of the construction activities. If NEI 06-06 is endorsed by the NRC, SNC plans to transition to a program that follows the guidance in NEI 06-06. The COL application currently commits to NEI 06-06, Revision 4, and will be changed in a future revision to commit to NEI 06-06, Revision 5. The applicant will evaluate substantial changes in subsequent revisions to NEI 06-06 and modify the construction phase FFD program to incorporate those substantial changes determined to be appropriate.

The applicant's response to RAI 13.6-33, as well as its supplemental response, revises Section 13.7 to address the issues discussed above. The relevant portion of the proposed revised text, to be included in a future revision of the VEGP COL FSAR, is included below:

> *The Fitness for Duty Program (FFD) is implemented and maintained in multiple and progressive phases dependent on the activities, duties, or access afforded to certain individuals at the construction site. In general, two different FFD programs will be implemented: a construction FFD program and an operations FFD program. The construction and operations phase programs are illustrated in* [FSAR] *Table 13.4-201.*
>
> *The construction FFD program is consistent with NEI 06-06 (*[FSAR] *Reference 201). NEI 06-06 applies to persons constructing or directing the construction of safety- and*

> *security-related structures, systems, or components performed onsite where the new reactor will be installed and operated. Management and oversight personnel, as further described in NEI 06-06, and security personnel prior to the receipt of special nuclear material in the form of fuel assemblies (with certain exceptions) will be subject to the operations FFD program that meets the requirements of 10 CFR Part 26, Subparts A through H, N, and O. At the establishment of a protected area, all persons who are granted unescorted access will meet the requirements of an operations FFD program. Prior to issuance of a Combined License, the construction FFD program at a new reactor construction site for those subject to Subpart K will be reviewed and revised as necessary should substantial revisions occur to either NEI 06-06 following NRC endorsement or the requirements of 10 CFR Part 26.*

The staff notes that Reference 201 in the above text refers to Revision 5 of NEI 06-06.

In RAI 13.6-34, the staff asked the applicant to: (1) describe how FSAR Table 13.4-201, Item 15, related to the security operational program, comports with 10 CFR 26.3, "Scope," and 10 CFR 26.4, and the guidance provided in the NRC's letter to NEI dated December 2, 2009, entitled "Status of U.S. Nuclear Regulatory Commission Review and Endorsement of NEI 06-06, 'Fitness for Duty Program Guidance for New Nuclear Power Plant Construction Sites,'" and (2) provide site-specific information to clearly and sufficiently describe the applicant's FFD program. This information would include, but is not limited to, any deviations or exceptions to the requirements of 10 CFR Part 26 as further described in NEI 06-06.

The applicant stated that the response to RAI 13.6-33 provided the changes to the COL application that will describe the FFD program required by 10 CFR Part 26. Site-specific information is also provided in that response to clarify which program will be used to cover the various classifications of workers that must be covered in accordance with 10 CFR Part 26. The applicant's response to RAI 13.6-35 (below) revises FSAR Table 13.4-201, Item 20 to address the guidance provided in the NRC's December 2, 2009 letter. The proposed revision to Item 20 of FSAR Table 13.4-201, to be included in a future revision of the VEGP COL FSAR, is included below.

Item	Program Title	Program Source (required by)	FSAR Section	Implementation Milestone	Implementation Requirements
20.	*Fitness for Duty (FFD) Program for Construction (workers and first-line supervisors)*	*10 CFR 26.4(f)*	*13.7*	*Prior to initiating 10 CFR Part 26 construction activities*	*10 CFR Part 26, Subpart K*
	FFD Program for Construction (management and oversight personnel)	*10 CFR 26.4(e)*	*13.7*	*Prior to initiating 10 CFR Part 26 construction activities*	*10 CFR Part 26, Subparts A - H, N, and O*
	FFD Program for Security Personnel	*10 CFR 26.4(e)(1)*	*13.7*	*Prior to initiating 10 CFR Part 26 construction activities*	*10 CFR Part 26, Subparts A - H, N, and O*
		10 CFR 26.4(a)(5) or 26.4(e)(1)		*Prior to the earlier of:* *A. Licensee's receipt of SNM in the form of fuel assemblies, or* *B. Establishment of a protected area, or* *C. The 10 CFR 52.103(g) finding*	*10 CFR Part 26, Subparts A - I, N, and O*
	FFD Program for FFD Program personnel	*10 CFR 26.4(g)*	*13.7*	*Prior to initiating 10 CFR Part 26 construction activities*	*10 CFR Part 26, Subparts A, B, D - H, N, O, and C per licensee's discretion*
	FFD Program for persons required to physically report to the Technical Support Center (TSC) or Emergency Operations Facility (EOF)	*10 CFR 26.4(c)*	*13.7*	*Prior to the conduct of the first full-participation emergency preparedness exercise under 10 CFR Part 50, App. E, Section F.2.a*	*10 CFR Part 26, Subparts A - I, N, and O, except for §§ 26.205 – 209*

Item	Program Title	Program Source (required by)	FSAR Section	*Implementation* Milestone	Requirements
	FFD Program for Operation	*10 CFR 26.4(a) and (b)*	*13.7*	*Prior to the earlier of:* *A. Establishment of a protected area, or* *B. The 10 CFR 52.103(g) finding*	*10 CFR Part 26, Subparts A - I, N, and O, except for individuals listed in § 26.4(b), who are not subject to §§ 26.205 – 209*

In its December 2, 2009, letter to NEI, the NRC stated that during the review and approval process for NEI 06-06, the applicant should provide the following statements in its application:.

- *NEI 06-06, Revision 5 was used in the development of the construction site FFD program.*
- *The applicant will review and revise its construction site FFD program as necessary to ensure that it comports with the NRC-endorsed version of NEI 06-06.*
- *If the NRC staff's review of NEI 06-06 results in substantive changes to the most recent, docketed FFD program description provided by the applicant, the applicant must amend its application to reflect the changes.*

The applicant's proposed revisions to FSAR Section 13.7 satisfactorily address the three items described above. The December 2, 2009, letter also provided implementation milestones for consideration by applicants. The staff confirmed that the proposed revisions to FSAR Table 13.4-201, Item 20, include all of the implementation milestones in the December 2, 2009, letter.

Therefore, based on the staff's acceptance of the proposed revisions to FSAR Section 13.7 and to FSAR Table 13.4-201, Item 20, as noted above, the NRC staff concludes that the applicant has satisfactorily addressed STD SUP 13.7-1 by providing sufficient information on the FFD program for both the construction phase and the operating phase of the units. The inclusion of this information in a future revision of the VEGP COL FSAR is ***Confirmatory Item 13.7-1****.*

<u>*Resolution of VEGP Site-Specific Confirmatory Item 13.7-1*</u>

Confirmatory Item 13.7-1 is an applicant commitment to revise its FSAR Section 13.7 and Table 13.4-201 regarding the FFD program for the construction phase and the operating phase of the units. The staff verified that the VEGP COL FSAR was appropriately revised. As a result, Confirmatory Item 13.7-1 is now closed.

License Conditions

In RAI 13.6-35, the staff asked the applicant if proposed License Condition 3, A.1, and G.7, described in Part 10 of the COL application comports with FSAR Table 13.4-201, Item 15, which itemizes the aspects of the security operational program.

The staff further evaluated the need for License Condition 3, A.1 and G.7, for the VEGP COL application and determined it was not needed because the implementation milestones for FFD are governed by 10 CFR Part 26. The staff communicated this information to SNC, which then submitted Supplement 1 to its response to this RAI, removing this license condition for FFD.

- *Part 10, License Condition 6*

The applicant proposed a license condition in Part 10 of the VEGP COL application to provide a schedule to support the NRC's inspection of operational programs, including the FFD program.

The proposed license condition is consistent with the policy established in SECY 05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," for operational programs and is acceptable.

Evaluation of WLS RAI Responses

The staff issued one RAI to the WLS applicant that mirrored the RAI 13.6-33 issued to the VEGP applicant, and one RAI that was not common to VEGP was issued.

The staff's evaluation of the responses provided by the WLS applicant to the two RAIs related to the FFD program is discussed below. The applicant responded to RAI 98, Question 13.07-1 on September 21, 2011, and to RAI 99, Question 13.07-2.

In RAI 98, Question 13.07-1, the staff requested that the applicant the following:

> Under 10 CFR 52.79(a)(44), the Applicant's FSAR must contain a description of the fitness for duty (FFD) program required by 10 CFR Part 26 and its implementation. How does the Applicant intend to update its FFD program for the construction phase? NEI 06-06 provides examples of the FFD program that is required and, if this guidance is endorsed by the NRC, will provide an acceptable method of complying with the NRC's regulations. If the NRC endorses NEI 06-06, does the Applicant intend to update its FFD program for the construction phase to comply with NEI 06-06? If future revisions to NEI 06-06 are endorsed by the NRC, does the Applicant intend to update its FFD program for the construction phase to comply with certain clarifications, additions, and exceptions in these future, endorsed revisions, as necessary?

In a September 22, 2011, response to RAI 98, Question 13.07-1, the applicant stated that its FFD Program will be developed based on the guidance given in NEI 06-06, Revision 5, and that subsequent revisions to NEI 06-06 would be subject to review by the applicant at that time for incorporation of any changes determined to be appropriate. The relevant portion of the revised text is included below:

> The Fitness for Duty Program (FFD) is implemented and maintained in multiple and progressive phases dependent on the activities, duties, or access afforded to certain individuals at the construction site.
>
> In general, two different FFD programs will be implemented: a construction FFD program and an operations FFD program. The construction and operations phase programs are outlined in Table 13.4-201.
>
> The construction FFD program is consistent with NEI 06-06 ([FSAR] Reference 201). NEI 06-06 applies to persons constructing or directing the construction of safety- and security-related structures, systems, or components performed onsite where the new reactor will be installed and operated. Management and oversight personnel, as further described in NEI 06-06 and security personnel prior to the receipt of special nuclear material in the form of fuel assemblies (with certain exceptions) will be subject to the operations FFD program that meets the requirements of 10 CFR Part 26, Subparts A through H, N, and 0. At the establishment of a protected area, all persons who are granted unescorted access will meet the requirements of an operations FFD program.
>
> The operations phase program is consistent with 10 CFR Part 26.

The staff considers RAI 98, Question 13.07-1 resolved by submittal of WLS COL FSAR, Revision 4.

In RAI 99, Question 13.07-2, the staff requested that applicant "provide site-specific information to clearly and sufficiently describe your FFD program in terms of the scope and level of detail to enable the staff to make a decision of acceptability."

In a November 22, 2011, response to RAI 99, Question 13.07-2, the applicant stated they "have not entered into a contract with a specific contractor for engineering, construction, and procurement at this time for the Lee site. Thus, the site-specific details for FFD programs have not been fully organized down to the details at the contractor level at this time. FSAR Section 13.7 has been updated to reflect the current plan for FFD programs in scope and level of detail for the Lee site known at this time." The following is the applicant's response to RAI 99, Question 13.07-2:

> This update is supplementing the previous changes made per Duke Energy's Response to Letter 098, RAI 13.7-1, dated September 21, 2011
>
> The Fitness for Duty Program (FFD) is implemented and maintained in multiple and progressive phases dependent on the activities, duties, or access afforded to certain individuals at the construction site.
>
> In general, two different FFD programs will be implemented: a construction FFD program and an operations FFD program. The construction and operations phase programs are outlined in Table 13.4-201.

The construction FFD program is consistent with NEI 06-06 ([FSAR] Reference 201). NEI 06-06 applies to persons constructing or directing the construction of safety- and security-related structures, systems, or components performed onsite where the new reactor will be installed and operated. Management and oversight personnel, as further described in NEI 06-06, and security personnel prior to the receipt of special nuclear material in the form of fuel assemblies (with certain exceptions) will be subject to the operations FFD program that meets the requirements of 10 CFR Part 26, Subparts A through H, N, and 0.

At the establishment of a protected area, all persons who are granted unescorted access will meet the requirements of an operations FFD program. Prior to issuance of a Combined License, the Duke-approved construction FFD program (elements Subpart K) will be reviewed and revised, as necessary, should substantial revisions occur to NEI 06-06 following NRC endorsement of the requirements of 10 CFR Part 26.

The following site-specific information is provided:

- The construction site is defined in the Physical Security Plan, Appendix E. and is under control of the Primary Site Contractor. The 10 CFR Part 26 requirements are implemented for the construction site area based on the description provided in Table 13.4-201.

- Construction Workers and First Line Supervisors (Primary Site Contractor employees and subcontractors) are covered by a Duke-approved Construction FFD Program (elements Subpart K).

- Duke employees and Duke subcontractor's construction management and oversight personnel are covered by a Duke Operations FFD Program and the Primary Site Contractor's employees and the Primary Site Contractor's subcontractors, construction management, and oversight personnel are covered by a Duke-approved FFD Program (elements Subpart A-H, N, and O).

- Duke security personnel are covered by a Duke Operations FFD Program and the Primary Site Contractor's security personnel are covered by a Duke-approved FFD Program (elements Subpart A-H, N, and O). This coverage is applicable from the start of construction activities to the earlier of (1) the receipt of SNM in the form of fuel assemblies, (2) The establishment of a protected area, or (3) the 10 CFR 52.103(g) finding.

- The Duke FFD Program personnel are covered by a Duke Operations FFD program and the Primary Site Contractor's FFD Program personnel are covered by a Duke-approved FFD Program (elements Subpart A, B, D-H, N, O, and C per licensee's discretion).

- Duke security personnel protecting fuel assemblies are covered by a Duke Operations FFD Program (Elements Subpart A-I, N and O).

- Personnel required to physically report to the Technical Support Center (TSC) or Emergency Operations Facility (EOF), when that requirement is in effect, are covered by a Duke Operations FFD program (elements Subpart A- I, N, and O, except for -§ 26.205 - 209)

The applicant stated that site-specific information would be reflected in a future revision to the WLS COL FSAR. The staff confirmed that the applicant has made the appropriate changes in Revision 9 of the WLS COL FSAR and considers this issue resolved.

13.7.5 Post Combined License Activities

For the reasons discussed in the technical evaluation section above, the staff finds the following license condition acceptable:

- License Condition (13-13) – The licensee shall submit to the Director of NRO, a schedule, no later than 12 months after issuance of the COL, that supports planning for and conduct of NRC inspection of the FFD operational program. The schedule shall be updated every 6 months until 12 months before scheduled fuel load, and every month thereafter until the FFD operational program has been fully implemented.

13.7.6 Conclusion

The staff reviewed WLS COL FSAR Section 13.7 along with the applicant’s proposed revision to this section. The staff’s review confirmed that the applicant’s proposed revision to WLS COL FSAR Section 13.7 has adequately addressed the required information related to the FFD portion of the WLS COL FSAR Section 13.7 and is consistent with the applicable requirements of 10 CFR Part 26 and 10 CFR 52.79(a)(44). In addition, the applicant provided a commitment to NEI “Fitness for Duty Program Guidance for New Nuclear Power Plant Construction Sites,” NEI 06-06, Revision 5, August 2009, therefore the staff, finds it acceptable.

13.8 Cyber Security

This section does not exist in either the AP1000 DCD or the WLS COL FSAR. The staff has added this section to this report to address issues regarding cyber security.

13.8.1 Introduction

In a July 29, 2011, letter to the NRC, Duke submitted Revision 2 of the Cyber Security Plan (CSP) for WLS Units 1 and 2. The CSP applies to all critical digital assets (CDAs) required for WLS operation. In the submittal, the applicant described how the requirements of 10 CFR 73.54, “Protection of Digital Computer and Communication Systems and Networks,” will be implemented to protect digital computer and communications systems and networks associated with the following functions from those cyber attacks, up to and including the DBT described in 10 CFR 73.1, “Purpose and Scope.” The scope of 10 CFR 73.54 includes CDAs associated with the following:

- safety-related and important-to-safety functions

- security functions
- emergency preparedness functions, including offsite communications
- support systems and equipment which, if compromised, would adversely impact safety, security, or emergency preparedness functions

13.8.2 Summary of Application

The applicant addresses cyber security in WLS COL FSAR Section 13. WLS COL FSAR, Revision 11, Section 13.6 incorporates by reference AP1000 DCD, Revision 19, Section 13.6. The applicant's CSP includes deviations from RG 5.71, "Cyber Security Programs for Nuclear Facilities." The staff evaluated these deviations.

In addition, in WLS COL FSAR Section 13.6, the applicant provided the following:

AP1000 COL Information Item

- STD COL 13.6-5

The applicant provided additional information in STD COL 13.6-5 to address COL Information Item 13.6-5, which provides information related to the cyber security program.

License Conditions

- Part 10, License Condition 3, Item G.10

The applicant proposed a license condition in WLS COLA Part 10, which requires the applicant to implement the cyber security program prior to initial fuel load.

- Part 10, License Condition 6

The applicant proposed a license condition in WLS COLA Part 10 to provide a schedule to support NRC inspection of operational programs included in WLS COL FSAR Table 13.4-201 including the cyber security program.

13.8.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

The applicable regulatory requirements for cyber security are as follows:

- 10 CFR 73.1, "Purpose and scope"
- 10 CFR 73.54, "Protection of digital computer and communication systems and networks"
- 10 CFR 73.55, "Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage," paragraphs (a)(1), (b)(8), and (m)

- 10 CFR 73.58, “Safety/security interface requirements for nuclear power reactors”
- 10 CFR Part 73, “Physical protection of plants and materials,” Appendix G, “Reportable Safeguards Events”

The applicable regulatory guidance for cyber security is RG 5.71.

13.8.4 Technical Evaluation

The staff reviewed WLS COL FSAR Section 13.6 and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The staff’s review confirmed that the information in the application and incorporated by reference addresses the required information relating to cyber security. The results of the staff’s evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff’s review of the WLS CSP focused on ensuring that the necessary programmatic elements are included in this plan 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. The staff reviewed the WLS CSP to ensure the necessary programmatic elements that, when effectively implemented, will provide the required high assurance of adequate protection. Effective implementation is dependent on the procedures and practices the applicant develops to satisfy the programmatic elements of its CSP. The facility implementing procedures are subject to future NRC inspection.

Section 1.2.3 of this report provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review to evaluate subsequent COL applications. To ensure that the staff’s findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from responses to RAIs.
- The staff confirmed that the July 29, 2011, WLS submittal transmitting its CSP was identical to the June 14, 2010, VEGP submittal transmitting its CSP, with the only exceptions being to the title of the units and the identification of the position charged with oversight of the program.
- The staff verified that the site-specific differences were not relevant.

The staff completed its review and concluded that the evaluation performed for the standard content is directly applicable to the WLS COL application. This finding included verifying that the difference in the position charged with oversight of the program (the General Manager, Organizational Effectiveness at WLS and Vice President of Nuclear Operations Support at VEGP) does not affect the staff's conclusions regarding the applicant's CSP. This standard content material is identified in this SER by use of italicized, double-indented formatting. The one confirmatory item in the standard content material retains the number assigned in the VEGP SER.

The following portion of this technical evaluation section is reproduced from VEGP SER Section 13.8.4:

> _AP1000 COL Information Item_
>
> - *STD COL 13.6-5*
>
> *The NRC staff reviewed STD COL 13.6-5 related to COL Information Item 13.6-5, which identifies the need for a COL applicant to address cyber security. STD COL 13.6-5 supplemented Section 13.6 of the VEGP COL FSAR by stating the following text is to be added after Section 13.6 of the VEGP ESP SSAR:*
>
> > *The Cyber Security Plan is submitted to the Nuclear Regulatory Commission as a separate licensing document to fulfill the requirements contained in 10 CFR 52.79(a)(36) and 10 CFR 73.54. The Cyber Security Plan will be maintained in accordance with the requirements of 10 CFR 52.98. The Plan is withheld from public disclosure pursuant to 10 CFR 2.390.*
>
> *Section 13.6 of the VEGP COL FSAR also refers to FSAR Table 13.4-201, "Operational Programs Required by NRC Regulations," as providing the milestone for implementing the cyber security program.*
>
> *The VEGP applicant submitted its Revision 0 of its CSP in a letter dated June 14, 2010, to demonstrate that the cyber security program will provide high assurance that digital computer and communication systems and networks are adequately protected against cyber attacks, up to and including the DBT as described in 10 CFR 73.1. The CSP has been withheld from public disclosure pursuant to 10 CFR 2.390(d)(1). In its review of this plan, the NRC staff used the guidance in RG 5.71 to determine if the regulatory requirements described in Section 13.8.3 of this SER are satisfied.*
>
> *The applicant described the cyber security program based on 10 CFR 73.54, including the audit of the effectiveness of the cyber security program as required by 10 CFR 73.55(m), submittal of CSPs and the establishment, maintenance and implementation of a cyber security program required by 10 CFR 73.55(a)(1) and 10 CFR 73.55(b)(8) and reporting requirements in 10 CFR Part 73, Appendix G. The implementation milestones for this program are included in VEGP COL FSAR Table 13.4-201.*
>
> *As detailed in the remainder of this SER section, the CSP has been reviewed by the NRC staff for format and content utilizing the NRC CSP template in RG 5.71, and found to include all features considered essential for such a program, and is acceptable. In particular, it has been found to comply with the Commission's regulations including 10 CFR 73.54, 10 CFR 73.55(a)(1), 10 CFR 73.55(b)(8),*

10 CFR 73.55(m), and 10 CFR Part 73, Appendix G and conforms to the NRC CSP template set forth in RG 5.71.

*The applicant has committed to incorporate this CSP into a future revision of the VEGP COL application to address NRC requirements in 10 CFR 73.54. This action will be tracked as **Confirmatory Item 13.8-1**.*

Resolution of VEGP Site-Specific Confirmatory Item 13.8-1

Confirmatory Item 13.8-1 is an applicant commitment to include the CSP into a future revision of the VEGP COL application. The staff verified that the VEGP COL application was appropriately revised. As a result, Confirmatory Item 13.8-1 is now closed.

13.8.4.1 Establishment of Cyber Security Program

The VEGP CSP describes how SNC will establish a cyber security program to achieve high assurance that the VEGP digital computer and communication systems and networks associated with safety, security, and emergency preparedness, including offsite communications and support systems and equipment which if compromised would adversely impact safety, security and/or emergency preparedness (SSEP) functions, and their digital assets, hereafter defined as CDAs, are adequately protected against cyber attacks up to and including the DBT. RG 5.71 provides a method that the staff considers acceptable for complying with this regulation. SNC complies with the requirements of 10 CFR 73.54 by providing a CSP that follows the template in Appendix A of RG 5.71, except as noted in Attachment A, "Vogtle Electric Generating Plant Units 3 and 4 Cyber Security Plan Deviations from Regulatory Guide RG 5.71." The VEGP CSP included:

> *Within the scope of the NRC's cyber security rule at 10 CFR 73.54, systems or equipment that perform important to safety functions include structures, systems, and components (SSCs) in the balance of plant (BOP) that could directly or indirectly affect reactivity at a nuclear power plant and could result in an unplanned reactor shutdown or transient. Additionally, these SSCs are under the licensee's control and include electrical distribution equipment out to the first inter-tie with the offsite distribution system.*

The VEGP CSP included a deviation from the guidance to clarify that systems or equipment that perform important to safety functions include SSCs in the balance of plant (BOP) that could directly or indirectly affect reactivity and could result in an unplanned reactor shutdown or transient. This deviation is consistent with Commission policy.

The NRC staff reviewed the VEGP CSP against the template in RG 5.71 and the staff requirements memorandum (SRM), CMWCO-10-0001, "Regulation of Cyber Security at Nuclear Power Plants," dated October 21, 2010.

The applicant states in the VEGP CSP that its security program complies with 10 CFR 73.54 by:

1) *establishing and implementing defensive strategies consistent with the defensive model, described in Section 3.1.5, including the security controls described in Sections 3.1, 3.2, and 3.3.*

2) *maintaining the program, as described in Section 4.*

Based on the above review, the NRC staff finds that establishment of a cyber security program described in Section 1 of the VEGP CSP is acceptable.

The following SER Sections 13.8.4.2 through 13.8.4.23 correlate to specific sections in Appendix A to RG 5.71. These SER sections use the same headings as the corresponding Appendix A sections, and include the Appendix A numbering system in the titles. SER Section 13.8.4.24 addresses each of the deviations identified in the applicant's CSP.

13.8.4.2 Security Assessment and Authorization (Section A.3.1.1 of Appendix A to RG 5.71)

Section 3.1.1 of the VEGP CSP states that the following will be reviewed every 24 months:

- *A formal documented security planning, assessment, and authorization policy that describes the purpose, scope, roles, responsibilities, management commitments, and coordination among departments and the implementation of the security program and the controls applied in accordance with Section 3.1.6*

- *A formal documented procedure to facilitate the implementation of the cyber security program and the security assessment*

The NRC staff reviewed the above and found that evaluation of the program elements every 24 months is not consistent with Section C.3.1.1 of RG 5.71. The time period between evaluations is 12 months longer than the time period provided in brackets in RG 5.71. However, this 24-month time period conforms to 10 CFR 73.54(g), requiring the applicant to review the cyber security program as a component of the physical security program in accordance with the requirements of 10 CFR 73.55(m), including the periodicity requirements. The requirement of 10 CFR 73.55(m) is that at minimum the applicant review each element of the physical protection program at least every 24 months.

Based on the above review, the NRC staff finds that the security assessment and authorization described in Section 3.1.1 of the VEGP CSP is acceptable.

13.8.4.3 Cyber Security Team (Section A.3.1.2 of Appendix A to RG 5.71)

Section 3.1.2 of the VEGP CSP states that a cyber security team, composed of individuals with broad knowledge, will be established and maintained and that the broad knowledge of the team will include the following areas:

- *Information and digital system technology; this includes cyber security, software development, offsite communications, computer system administration, computer engineering, and computer networking.*

- *Nuclear facility operations, engineering, and safety; this includes overall facility operations and plant technical specification compliance.*

- *Physical security and emergency preparedness; this includes the site's physical security and emergency preparedness systems and programs.*

This section of the VEGP CSP also enumerates the roles and responsibilities of the cyber security team. Aside from the deviations discussed below, this section of the VEGP CSP conforms to the CSP template wording provided in Section A.3.1.2 of RG 5.71.

The VEGP CSP includes several deviations from the text of RG 5.71:

1) *The first deviation clarifies that the cyber security team (CST) will be responsible for “overseeing” preparation of documentation of cyber security controls and that, in fact, non-team members (such as vendor personnel) may perform some of these actions, under the supervision of the CST. This clarification is acceptable to the staff since the responsibility to ensure compliance with 10 CFR 73.54 remains with the CST.*

2) *The second deviation changes the CST responsibility from “assuring the retention” of assessment documentation to “establishing the retention policy” for assessment documentation. Again, the deviation is acceptable to the staff since the responsibility to ensure compliance with 10 CFR 73.54 remains with the CST.*

3) *The third and final deviation seeks to change the basis for CST determinations being made in a free and objective manner. The RG 5.71 wording states that the CST should be free to make determinations that are not constrained by “operational goals.” The deviation changes the respective sentence to say “…by business goals.” Again, the deviation is acceptable to the staff since it maintains the same objective of keeping financial considerations out of decision making regarding cyber security.*

Based on the above review, the NRC staff finds that the CST described in Section 3.1.2 of the VEGP CSP is acceptable.

13.8.4.4 Identification of Critical Digital Assets (Section A.3.1.3 of Appendix A to RG 5.71)

Section 3.1.3 of the VEGP CSP states that to identify the critical systems (CSs) at VEGP, the CST identified and documented plant systems, equipment, communication systems, and networks that are associated with the SSEP functions described in 10 CFR 73.54(a)(1), as well as the support systems associated with these SSEP functions in accordance with the approved plant licensing basis.

The VEGP CSP also states that the CST identified and documented CDAs that have a direct, supporting, or indirect role in the proper functioning of CSs.

The steps outlined in the VEGP CSP essentially match the corresponding steps described in RG 5.71 for this same activity. The only difference between the corresponding section in RG 5.71 and the VEGP CSP is the addition of the modifying phrase: "…and defined in the approved plant licensing basis."

10 CFR 73.54(a)(1) requires that the licensee protect digital computer and communication systems and networks associated with: (i) safety-related and important-to-safety functions; (ii) security functions; (iii) emergency preparedness functions, including offsite communications; and (iv) support systems and equipment which, if compromised, would adversely impact SSEP functions.

This deviation is acceptable because SNC proposes to use its licensing basis to identify CSs that are associated with SSEP functions, as 10 CFR 73.54 requires. This statement includes the first step in RG 5.71 to analyze digital computer and communication systems and networks to determine if they include CDAs.

Based on the above review, the NRC staff finds the applicant's proposal, described in Section 3.1.3 of the VEGP CSP, to use 10 CFR 73.54(a)(1) and its licensing basis to identify CDAs to be acceptable.

13.8.4.5 Reviews and Validation Testing (Section A.3.1.4 of Appendix A to RG 5.71)

Section 3.1.4 of the VEGP CSP states that the VEGP CST will be responsible for conducting a review, performing validation activities, and for each CDA, the CST determined:

- *its direct and indirect connectivity pathways*
- *infrastructure interdependencies*
- *the application of defensive strategies, including defensive models, security controls, and other defensive measures*

- *The CSP also requires that the CST validate the above activities through comprehensive walkdowns, which include a range of activities that conform to those activities specified in RG 5.71 for this purpose.*

The CSP also requires that the CST validate the above activities through comprehensive walkdowns, which include a range of activities that conform to those activities specified in RG 5.71 for this purpose.

The requirements, processes and procedures described in this section of the VEGP CSP conform to, and encompass all of the same specifications, outlined in the comparable section of RG 5.71.

Based on the above review, the NRC staff finds that reviews and validation testing described in Section 3.1.4 of the VEGP CSP is acceptable.

13.8.4.6 Defense-In-Depth Protective Strategies (Section A.3.1.5 of Appendix A to RG 5.71)

Section 3.1.5 of the VEGP CSP states that the defensive strategy consists of the defensive model described in Section C.3.2 of RG 5.71, and the detailed defensive architecture of Appendix C, Section 6, defense-in-depth controls in Appendix C, Section 7, and security controls applied in accordance with Section 3.1.6 of the VEGP CSP with one deviation to its defensive architecture. The VEGP defensive architecture, including the deviation is consistent with the security model described in RG 5.71, which provides for isolation of safety-related and security CDAs.

Based on the above review, the NRC staff finds that the defense-in-depth protective strategies described in Section 3.1.5 of the VEGP CSP are acceptable.

13.8.4.7 Application of Security Controls (Section A.3.1.6 of Appendix A to RG 5.71)

Section 3.1.6 of the VEGP CSP states that VEGP Units 3 and 4 established defense-in-depth protective strategies by applying and documenting the following:

- *the defensive model described in Section 3.2 of RG 5.71 (discussed in SER Section 13.8.4.6)*

- *the physical and administrative security controls established by the VEGP Units 3 and 4 Physical Security Program and physical barriers, such as locked doors, locked cabinets, and locating CDAs in the VEGP Units 3 and 4 protected area or vital areas, which are part of the overall security controls used to protect CDAs from attacks*

- *verification of the effectiveness of the implemented operational and management controls described in Appendix C to RG 5.71 and implemented alternatives to the Appendix C controls for each CDA*

- *the technical controls described in Appendix B to RG 5.71 and the operational and management controls described in Appendix C to RG 5.71, consistent with the process described below*

The VEGP CSP deviates from RG 5.71, Section C.3.3 Security Controls and Appendix A.3.1.6, by stating that when a control from Appendices B and C of RG 5.71 is not implemented, the licensee will implement alternate control(s) that "do not provide less protection than the corresponding" control in the appendix. This deviation is consistent with the method used in RG 5.71, which states that controls should provide equal or better protection,

The VEGP CSP also deviates from RG 5.71 by stating that when a control can be proved to be unnecessary, the applicant will perform an analysis demonstrating that the control is not necessary, and will provide a documented justification. Although RG 5.71 specifically calls for an attack vector analysis, and the VEGP CSP does not specifically commit to performing an attack vector analysis, the VEGP CSP does commit to justifying the non-applicability of a control by demonstrating that the attack vector does not exist. This provides for the same outcome as RG 5.71.

Based on the above review, the NRC staff finds that the application of security controls described in Section 3.1.6 of the VEGP CSP is acceptable.

13.8.4.8 Incorporating the Cyber Security Program into the Physical Protection Program (Section A.3.2 of Appendix A to RG 5.71)

Section 3.2 of the VEGP CSP states that the licensee will provide the management interfaces necessary to appropriately coordinate physical and cyber security activities, as follows:

- *establish an organization that is responsible for cyber security and is independent from operations*

- *document physical and cyber security interdependencies*

- *develop policies and procedures to coordinate management of physical and cyber security controls*

- *incorporate unified policies and procedures to secure CDAs from attacks up to and including the DBT*

- *coordinate acquisition of physical or cyber security services, training, devices, and equipment*

- *coordinate interdependent physical and cyber security activities and training with physical and cyber security personnel*
- *integrate and coordinate incident response capabilities with physical and cyber incident response personnel*
- *train senior management regarding the needs of both disciplines*
- *periodically exercise the entire security organization using realistic scenarios combining both physical and cyber simulated attacks*

The VEGP CSP deviates from RG 5.71 by not creating a unified security organization. The commitment to provide for appropriate management interfaces to coordinate the physical and cyber security organizations provides for a level of integration equivalent to a unified organization.

Based on the above review, the NRC staff finds that the incorporation of the cyber security program into the physical protection program described in Section 3.2 of the VEGP CSP is acceptable.

13.8.4.9 Policies and Implementing Procedures (Section A.3.3 of Appendix A to RG 5.71)

Section 3.3 of the VEGP CSP states that the licensee will develop policies and procedures to address the security controls in Appendices B and C to RG 5.71 and review and approve issues and uses, and revise the same according to Section 4 of the CSP. The CSP will also establish specific responsibilities for the positions described in Section 10.10 of Appendix C to RG 5.71, with the following deviation.

The CSP states that this will occur "in accordance with the security control application process in Section 3.1.6 of this Plan." This process requires the applicant to justify and demonstrate that any deviation from the controls in RG 5.71 provide no less protection than the corresponding control in Appendices B and C; therefore, the VEGP CSP will require the same level of protection as the corresponding commitment in RG 5.71.

Based on the above review, the NRC staff finds that the policies and implementing procedures described in Section 3.3 of the VEGP CSP are acceptable.

13.8.4.10 Maintaining the Cyber Security Program (Section A.4 of Appendix A to RG 5.71)

Section 4 of the VEGP CSP states that the applicant will establish the programmatic elements necessary to maintain security throughout the life cycle of the CDAs, and that the applicant has implemented these elements. For new assets, SNC commits to follow the process described in Section 4.2.

Section 4 of the VEGP CSP is nearly identical to Section C.4 of RG 5.71, with the deviation of replacing the bracketed text [Licensee/Applicant] *with VEGP Units 3 and 4, and by including the caveat that the operational and management controls are applied following the process described in Section 3.1.6. The process described in Section 3.1.6 allows the licensee/applicant to not apply a control if it can demonstrate that the control is not necessary by justifying that the attack vector associated with the control does not exist. This approach is consistent with the method used in RG 5.71, and does not reduce the protection to the plant.*

Based on the above review, the NRC staff finds that the maintenance of the cyber security program described in Section 4 of the VEGP CSP is acceptable.

13.8.4.11 Continuous Monitoring and Assessment (Section A.4.1 of Appendix A to RG 5.71)

Section 4.1 of the VEGP CSP states that the licensee will continue to monitor security controls for effectiveness; will ensure that they remain in place throughout the life cycle of the CDA; and will verify that rogue assets are not connected to the infrastructure.

The VEGP CSP includes a single deviation from Section A.4.1 of RG 5.71. The RG states that "[Licensee/Applicant] *continuously monitors security controls consistent with Appendix C to RG 5.71," whereas the VEGP CSP states that "VEGP Units 3 and 4 continues to monitor security controls consistent with Appendix C to RG 5.71."*

This deviation is consistent with the method in RG 5.71, which calls for periodic assessments, which is consistent with the statement "continues to monitor."

Based on the above review, the NRC staff finds that the ongoing monitoring and assessment described in Section 4.1 of the VEGP CSP is acceptable.

13.8.4.12 Periodic Assessment of Security Controls (Section A.4.1.1 of Appendix A to RG 5.71)

Section 4.1.1 of the VEGP CSP states that the licensee will periodically assess that security controls implemented for each CDA remain robust, resilient, and effective in place throughout the life cycle, at least every 24 months.

The NRC staff reviewed the above and found that this period of assessment is not consistent with RG 5.71. The time period between evaluations is 12 months longer than the time period provided in RG 5.71. However, this 24-month time period conforms to 10 CFR 73.54(g) requiring the licensee/applicant to review the cyber security program as a component of the physical security program in accordance with the requirements of 10 CFR 73.55(m), including the periodicity requirements. The requirements of 10 CFR 73.55(m) are that, at a minimum, the licensee/applicant review each element of the physical protection program, which includes the cyber security program, at least every 24 months.

Furthermore, the VEGP CSP states that controls will be reviewed according to the requirements of the security controls if that period of review occurs more often. This is also consistent with the method provided in RG 5.71.

Based on the above review, the NRC staff finds that the periodic assessment of security controls described in Section 4.1.1 of the VEGP CSP is acceptable.

13.8.4.13 Effectiveness Analysis (Section A.4.1.2 of Appendix A to RG 5.71)

Section 4.1.2 of the VEGP CSP states that the licensee will monitor and measure the effectiveness of the cyber security program and its security controls to ensure that both are implemented correctly, operating as intended, and continuing to provide high assurance that CDAs are protected against cyber attacks. The licensee commits to verifying the effectiveness of the security controls every 24 months, or in accordance with the specific requirements of the implemented security controls, whichever is more frequent.

The NRC staff reviewed the above and found that this period of verification is inconsistent with RG 5.71. The time period between evaluations is 12 months longer than the time period provided in RG 5.71. However, this 24-month time period conforms to 10 CFR 73.54(g) requiring the applicant to review the cyber security program as a component of the physical security program in accordance with the requirements of 10 CFR 73.55(m), including the periodicity requirements. The requirements of 10 CFR 73.55(m) are that, at a minimum, the applicant review each element of the physical protection program, which includes the cyber security program, at least every 24 months.

Furthermore, the VEGP CSP states that verification will also occur according to the requirements of the security controls if that period of verification occurs more often. This is also consistent with the method provided in RG 5.71.

Based on the above review, the NRC staff finds that the effectiveness analysis described in Section 4.1.2 of the VEGP CSP is acceptable.

13.8.4.14 Vulnerability Assessments and Scans (Section A.4.1.3 of Appendix A to RG 5.71)

Section 4.1.3 of the VEGP CSP states vulnerability assessments will be performed as specified in the security controls in Appendices B and C of RG 5.71 to identify new vulnerabilities that have the potential to impact the effectiveness of the cyber security program and the security of the CDAs. The applicant also commits to address vulnerabilities that could cause CDAs to become compromised or could have an adverse impact on SSEP functions. Section 13.1 of Appendix C of RG 5.71 provides that vulnerability assessments should occur no less frequently than once a quarter, at random intervals, and when new potential vulnerabilities are reported and identified.

Section A.4.1.3 of RG 5.71 states that vulnerability assessments will occur no less frequently than quarterly, whereas the VEGP CSP states that this will occur, *"as specified in the implemented security controls in Appendices B and C to RG 5.71 and implemented alternatives to the Appendices B and C controls." The process SNC has committed to in Section 3.1.6 of the VEGP CSP requires SNC, if it does not implement the controls in Appendices B and C, to demonstrate that an alternate control does not provide less protection than the corresponding control in Appendices B and C.*

Therefore, if SNC does not implement the security control in Section 13.1, or deviates from the requirement for a quarterly vulnerability assessment, it will ensure that this deviation does not provide less protection than performing quarterly vulnerability assessments, and will provide an analysis that demonstrates that the attack vector does not exist and will document this justification for inspection.

Based on the above review, the NRC staff finds that the vulnerability assessments and scans described in Section 4.1.3 of the VEGP CSP are acceptable.

13.8.4.15 Change Control (Section A.4.2 of Appendix A to RG 5.71)

Section 4.2 of the VEGP CSP states that the licensee will systematically plan, approve, test, and document changes to the environment of the CDAs, the addition of CDAs to the environment, and changes to existing CDAs in a manner that provides a high level of assurance that the SSEP functions are protected from cyber attacks. The CSP also commits that the program establish that changes made to CDAs use the design control and configuration management procedures or other procedural processes to ensure that the existing security controls are effective and that any pathway that can be exploited to compromise a CDA is protected from cyber attacks.

The VEGP CSP does not deviate from Section A.4.2 of RG 5.71.

Based on the above review, the NRC staff finds that the change control process described in Section 4.2 of the VEGP CSP is acceptable.

13.8.4.16 Configuration Management (Section A.4.2.1 of Appendix A to RG 5.71)

Section 4.2.1 of the VEGP CSP states that the licensee will implement and document a change management process as described in Section 4.2 of the VEGP CSP. Further, it commits to implement and document the applied configuration management controls described in Appendix C, Section 11 to RG 5.71 following the process described in Section 3.1.6 of the CSP.

The VEGP CSP does not specifically commit to apply the security controls in Section 11 of Appendix C of RG 5.71; however, it does commit to apply the process in Section 3.1.6 of the CSP. The commitment in Section 4.2.1 is consistent with Section A.4.2.2 of RG 5.71 as the applicant has committed, if it does not implement the security controls in Section 11 of RG 5.71, either to *implement alternative controls that do not provide less protection than what is in Section 11, or to demonstrate that this control is unnecessary by demonstrating that the attack vectors associated with Section 11 to Appendix C of RG 5.71 do not exist for VEGP.*

Based on the above review, the NRC staff finds that the configuration management process described in Section 4.2.1 of the VEGP CSP is acceptable.

13.8.4.17 Security Impact Analysis of Changes and Environment (Section A.4.2.2 of Appendix A to RG 5.71)

Section 4.2.2 of the VEGP CSP states that the applicant will perform a security impact analysis in accordance with Section 4.1.2 before implementing a design or configuration change to a CDA or, when changes to the environment occur, to manage potential risks introduced by the changes. The CSP also commits to evaluate, document, and incorporate into the security impact analysis safety and security interdependencies of other CDAs or systems, as well as updates, and documents the following:

- *the location of the CDA and connected assets*
- *connectivity pathways (direct and indirect)*
- *infrastructure interdependencies*
- *application of defensive strategies, including defensive models, security controls, and others*
- *defensive strategy measures*
- *plant-wide physical and cyber security policies and procedures that secure CDAs from a cyber attack, including attack mitigation and incident response and recovery*

The VEGP CSP commits to perform these impact analyses as part of the change approval process to assess the impacts of the changes on the security posture of CDAs and security controls, as described in Section 4.1.2 of the VEGP CSP, and to address any identified gaps to protect CDAs from cyber attack, up to and including the DBT as described in Section 4.2.6.

Finally, Section 4.2.2 states that the licensee will manage CDAs for the cyber security of SSEP functions through an ongoing evaluation of threats and vulnerabilities and implementation of each of the applied security controls provided in Appendix B or C of RG 5.71 and implement alternatives to the Appendices B and C controls during all phases of the life cycle. Additionally, SNC has established and documented procedures for screening, evaluating, mitigating, and dispositioning threat and vulnerability notifications received from credible sources. Dispositioning includes implementation of security controls to mitigate newly reported or discovered threats and vulnerabilities.

The language in Section 4.2.2 of the VEGP CSP is identical to that in Section A.4.2.2 of RG 5.71 and includes no deviations.

Based on the above review, the NRC staff finds that the security impact analysis of changes and environment described in Section 4.2.2 of the VEGP CSP is acceptable.

13.8.4.18 Security Reassessment and Authorization (Section A.4.2.3 of Appendix A to RG 5.71)

Section 4.2.3 of the VEGP CSP states that the licensee will have implemented, documented, and maintained a process that ensures that modifications to CDAs are evaluated before implementation so that security controls remain effective and that any pathway that can be exploited to compromise the modified CDA is addressed to protect CDAs and SSEP functions from cyber attacks. This section further states that the VEGP cyber security program establishes that additions and modifications are evaluated, using a proven and accepted method, before implementation to provide high assurance of adequate protection against cyber attacks, up to and including DBTs, using the process described in Section 4.1.2 of the VEGP CSP.

The licensee also commits to disseminate, review, and update the following when a CDA modification is conducted:

- *a formal, documented security assessment and authorization policy, which addresses the purpose, scope, roles, responsibilities, management commitment, coordination among entities, and compliance to reflect all modifications or additions*
- *a formal, documented procedure to facilitate the implementation of the security reassessment and authorization policy and associated controls*

The VEGP CSP does not deviate from Section A.4.2.3 of RG 5.71.

Based on the above review, the NRC staff finds that the security reassessment and authorization described in Section 4.2.3 of the VEGP CSP is acceptable.

13.8.4.19 Updating Cyber Security Practices (Section A.4.2.4 of Appendix A to RG 5.71)

Section 4.2.4 of the VEGP CSP states that the licensee reviews, updates and modifies cyber security policies, procedures, practices, existing cyber security controls, detailed descriptions of network architecture (including logical and physical diagrams), information on security devices, and any other information associated with the state of the cyber security program or the applied security controls provided in Appendices B and C to RG 5.71 and implemented alternatives to the Appendices B and C controls when changes occur to CDAs or the environment.

This information includes the following:

- *plant- and corporate-wide information on the policies, procedures, and current practices related to cyber security*
- *detailed network architectures and diagrams*
- *configuration information on security devices or CDAs*
- *new plant- or corporate-wide cyber security defensive strategies or security controls being developed and policies, procedures, practices, and technologies related to their deployment*
- *the site's physical and operational security program*
- *cyber security requirements for vendors and contractors*
- *identified potential pathways for attacks*
- *recent cyber security studies or audits (to gain insight into areas of potential vulnerabilities); and identified infrastructure support systems (e.g., electrical power; heating, ventilation, and air conditioning; communications; fire suppression) whose failure or manipulation could impact the proper functioning of CSs*

The VEGP CSP does not deviate from Section A.4.2.4 of RG 5.71.

Based on the above review, the NRC staff finds that updating of cyber security practices described in Section 4.2.4 of the VEGP CSP is acceptable

13.8.4.20 Review and Validation Testing of a Modification or Addition of a Critical Digital Asset (Section A.4.2.5 of Appendix A to RG 5.71)

The VEGP CSP Section 4.2.5 states the licensee will conduct and document the results of reviews and validation tests of each CDA modification and addition using the process described in Section 3.1.4 of the VEGP CSP.

The VEGP CSP does not deviate from Section A.4.2.5 of RG 5.71.

Based on the above review, the NRC staff finds that the Review and Validation Testing of Modifications or Additions of a Critical Digital Asset described in Section 4.2.5 of VEGP CSP is acceptable.

13.8.4.21 Application of Security Controls Associated with a Modification or Addition (Section A.4.2.6 of Appendix A to RG 5.71)

Section 4.2.6 of the VEGP CSP states that when new CDAs are introduced into the environment of VEGP, the licensee:

- *deploys the CDA into the appropriate level of the defensive model described in Section 3.1.5 of this plan;*

- *applies the technical controls identified in Appendix B to RG 5.71 and the operational and management controls described in Appendix C to RG 5.71 in a manner consistent with the process described in Section 3.1.6 of this plan*

- *confirms that the implemented operational and management controls described in Appendix C to RG 5.71, and implemented alternatives to the Appendix C controls, are effective for the CDA*

The plan also commits that when CDAs are modified, the licensee:

- *verifies that the CDA is deployed into the proper level of the defensive model described in Section 3.1.5 of this plan*

- *performs a security impact analysis, as described in Section 4.2.2 of this plan*

- *verifies that the technical controls identified in Appendix B to RG 5.71 and the operational and management controls described in Appendix C to RG 5.71 are addressed in a manner consistent with the process described in Section 3.1.6 of this plan*

- *verifies that the applied security controls discussed above are implemented effectively, consistent with the process described in Section 4.1.2 of this plan*

- *confirms that the implemented operational and management controls discussed in Appendix C to RG 5.71 and implemented alternatives to the Appendix C controls are effective for the CDA*

The VEGP CSP deviates from Section 4.2.6 of RG 5.71 by modifying the phrase "applies the technical controls identified in Appendix B to RG 5.71 in a manner consistent with the process described in Section 3.2 of RG 5.71," to read "applies the technical controls identified in Appendix B to RG 5.71 and the operational and management controls described in Appendix C to RG 5.71 in a manner consistent with the process described in Section 3.1.6 of this plan." This is consistent with RG 5.71 as the VEGP CSP commits to following the process in Section 3.1.6 of the VEGP CSP, which requires that controls are applied, an alternative that provides equivalent protection is provided, or the licensee demonstrates that the control is not necessary.

The VEGP CSP also deviates from Section A.4.2.6 of RG 5.71 with the modification of this phrase, "verifies that the security controls discussed above are implemented effectively, consistent with the process described in Section 4.1.2 of this plan" to read "verifies that the applied security controls *discussed above are implemented effectively, consistent with the process described in Section 4.1.2 of this plan."*

This deviation is consistent with the method used in RG 5.71. RG 5.71 assumes that all the controls in Appendices B and C will be applied; whereas, the VEGP CSP commits that if a control is not applied, there will be no reduction in protection as compared to the corresponding control. This method is also captured in RG 5.71 and, therefore, the VEGP CSP is consistent with RG 5.71.

Based on the above review, the NRC staff finds that the application of security controls associated with a modification or addition described in Section 4.2.6 of the VEGP CSP is acceptable.

13.8.4.22 Cyber Security Program Review (Section A.4.3 of Appendix A to RG 5.71)

Section 4.3 of the VEGP CSP states that the applicant has established the necessary measures and governing procedures to implement periodic reviews of applicable program elements, in accordance with the requirements of 10 CFR 73.55(m). Specifically, the VEGP CSP calls for a review of the program's effectiveness at least every 24 months. In addition, reviews are to be conducted as follows:

- *within 12 months following initial implementation of the program*
- *as necessary, based upon site-specific analyses, assessments, or other performance indicators*
- *as soon as reasonably practical, but no longer than 12 months after changes occur in personnel, procedures, equipment, or facilities that potentially could adversely affect cyber security*
- *by individuals independent of those personnel responsible for program management, and any individual who has direct responsibility for implementing the program*

This deviates from RG 5.71 in the specific wording, but includes the same commitments. Specifically, RG 5.71 states that the licensee reviews the program's effectiveness at least every 24 months. In addition, reviews are conducted as follows:

- *within 12 months of the initial implementation of the program*
- *within 12 months of a change to personnel, procedures, equipment, or facilities that potentially could adversely affect security*
- *as necessary based upon site-specific analyses, assessments, or other performance indicators*

- *by individuals independent of those personnel responsible for program implementation and management*

Based on the above review, the NRC staff finds that the cyber security program review described in Section 4.3 of the VEGP CSP is acceptable.

13.8.4.23 Document Control and Records Retention and Handling (Section A.5 of Appendix A to RG 5.71)

Section 5 of the VEGP CSP states the necessary measures and governing procedures to ensure that sufficient records of items and activities affecting cyber security are developed, reviewed, approved, issued, used, and revised to reflect completed work. VEGP will retain records and supporting technical documentation required to satisfy the requirements of 10 CFR 73.54 and 10 CFR 73.55, "Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors against Radiological Sabotage," until the NRC terminates the facility's operating license. Records are retained to document access history, as well as to discover the source of cyber attacks or other security-related incidents affecting CDAs or SSEP functions, or both. VEGP Units 3 and 4 will retain superseded portions of these records for at least three years after the record is superseded, unless otherwise specified by the NRC.

This deviates from RG 5.71 by not specifically detailing the types of records, but instead describes that records will be retained to document access history and information needed to discover the source of cyber attacks and incidents. This is consistent with what is included in RG 5.71, Section 5, and includes all the performance-based characteristics and commitments of that section.

Based on the above review, the NRC staff finds that the document control and records retention handling described in Section 5 of the VEGP CSP is acceptable.

13.8.4.24 Deviations Taken to RG 5.71, Sections C.1 Through C.5

The VEGP CSP states that the plan deviates from Regulatory Positions C.1 through C.5 of RG 5.71, as noted in Attachment A to the CSP. It also deviates from Section A.1 of Appendix A of RG 5.71. For that reason, the staff considers that the full evaluation of the CSP must include a review of the deviations taken to those sections of RG 5.71 as listed in the VEGP CSP. This section of the SER lists those 69 specific deviations and their evaluated security impact. The following deviations were provided in a table, as part of Attachment A to the CSP.

13.8.4.24.1 RG 5.71, Section C.2, fourth paragraph, first sentence (page 8)

SNC added the term "adequately" to the phrase "…systems and equipment are protected from cyber attack." Since 10 CFR 73.54 specifically makes that same statement, the staff found no reason to object to that clarification. The objective is to provide adequate protection to the identified CDAs.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.2 RG 5.71, Section C.2, fourth paragraph, twelfth bullet, third sub-bullet (page 8)

SNC clarifies that its overall design is based on the Westinghouse AP1000 design and states that the AP1000 DCD commits to Revision 1 of RG 1.152, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants." Since the applicant is required to have a cyber security program that meets the performance objectives outlined in 10 CFR 73.54 and is not obliged to achieve that requirement exclusively through the example provided by RG 5.71, this clarification, in and of itself, was not considered by the staff as deviating from the requirements established by the rule,

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.3 RG 5.71, Section C.2, fifteenth bullet (page 8)

The deviation states that the required policies and procedures have not yet been written, reviewed, and approved, and, thus, are not currently available for inspection and review.

The NRC requires that these policies and procedures be completed and available for review by the completion of the CSP implementation schedule proposed by the applicant, since CSP inspections would not occur until that time. The requirements of 10 CFR 73.55(a)(4) and proposed License Condition 6 provide the necessary controls associated with developing the required policies and procedures of the CSP.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.4 RG 5.71, Section C.3, Figure 1 (Page 10)

The deviation changes the arrows on the left side of Figure 1 from "Continuous Monitoring" to "Ongoing Monitoring."

The NRC intended monitoring to occur periodically, and when required, based on certain inputs into the process. SNC states that "continuous" might imply that monitoring was perpetual and not event driven. This was not the staff's intent with the term "continuous." The staff accepts the use of the term "ongoing" to better reflect the intent of this diagram.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.5 RG 5.71, Section C.3, third paragraph, first sentence (Page 10)

The VEGP CSP changes the statement, "An acceptable method to establish a cyber security program at a facility is by performing the following, (1) analyze the digital computer and communication systems and networks, …" to "An acceptable method to establish a cyber security program at a facility is by performing the following: (1) identify critical systems and critical digital assets as described in Section C.3.1.3, (2) analyze the digital computer and communication systems and networks..."

This deviation is acceptable because SNC proposes to use its licensing basis to identify CSs that are associated with SSEP functions, as 10 CFR 73.54 requires. This statement includes the first step in RG 5.71 to analyze digital computer and communication systems and networks to determine if they include CDAs.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.6 RG 5.71, Section C.3.1, first paragraph, first sentence (page 11)

The VEGP CSP changes the statement, "Consistent with the requirements of 10 CFR 73.54(b)(1), a licensee must conduct a site-specific analysis of digital computer and communication systems and networks to identify CDAs, which are those assets that, if compromised, could adversely impact the SSEP functions of nuclear facilities." to "Consistent with the requirements of 10 CFR 73.54(b)(1), a licensee must conduct a site-specific analysis of digital computer and communication systems and networks to identify CDAs, which are those assets that, if compromised, could adversely impact the CSs of nuclear facilities."

SNC defines a CS as:

> *An analog or digital technology-based system in or outside of the plant that performs or is associated with a safety-related, important-to-safety, security, or emergency preparedness function. These critical systems include, but are not limited to, plant systems, equipment, communication systems, networks, offsite communications, or support systems or equipment, that perform or are associated with a safety-related, important-to-safety, security, or emergency preparedness function as defined by the approved plant licensing basis.*

This definition ties CSs to SSEP functions; therefore, the change is consistent with the method used in RG 5.71, as this means that CSs are all those assets associated with SSEP functions, and, therefore, could adversely impact those SSEP functions.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.7 RG 5.71, Section C.3.1, first paragraph, second bullet (page 11)

The VEGP CSP includes a deviation to correct an editorial omission in RG 5.71. Page 11 of RG 5.71 states that:

An acceptable method for identifying and documenting CDAs is as follows:

- *obtain authorization for security assessment*
- *define roles and responsibilities cyber personnel and form the cyber security team*
- *identify and document CDAs at the facility*
- *review and validate configurations of CDAs*

The VEGP CSP corrects the second bullet to read:

- *define roles and responsibilities of cyber personnel and form the cyber security team*

This deviation which supplies the omitted "of" is consistent with the intent of the referenced bullet.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.8 RG 5.71, Section C.3.1.2, third paragraph, second bullet (page 13)

The VEGP CSP changes the second bullet on Page 13 of RG 5.71 from:

> *documenting all key observations, analyses, and findings during the assessment process so that this information can be used as a basis for applying security controls;*

to:

> *documenting all key observations, analyses, and findings during the assessment process so that this information can be used as a basis for addressing security controls;*

This deviation is acceptable because RG 5.71 allows a licensee to address, as opposed to apply, security controls if it follows the process in Appendix A, Section 3.1.6 of RG 5.71, which is to apply the control, apply an alternative that provides no less protection than the corresponding security control, or to demonstrate that the control is not necessary because the attack vector, root cause, or vulnerability associated with the control does not exist.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.9 RG 5.71, Section C.3.1.2, third paragraph, sixth bullet (page 13)

The VEGP CSP changes the sixth bullet on Page 13 from:

- *preparing documentation and overseeing implementation of the cyber security controls provided in Appendices B and C to this guide, documenting the basis for not implementing certain cyber security controls provided in Appendix B, or documenting the basis for the implementation of alternate or compensating measures in lieu of any cyber security controls provided in Appendix B; and*

to:

- *overseeing documentation and implementation of the cyber security controls provided in Appendices B and C to this guide, documenting the basis for not implementing certain cyber security controls provided in Appendix B and C, or documenting the basis for the implementation of alternate or compensating measures in lieu of any cyber security controls provided in Appendix B and C; and*

This deviation is acceptable because overseeing the documentation and implementation of security controls by qualified personnel is an approved method. Further, the extension of this method in Appendix C is also acceptable as the licensee has committed to follow the process in Appendix A, Section 3.1.6 of RG 5.71.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.10 RG 5.71, Section C.3.1.2, third paragraph, seventh bullet (page 13)

The VEGP CSP includes a deviation from RG 5.71 that changes bullet 7 from:

> *assuring the retention of all assessment documentation, including notes and supporting information, in accordance with 10 CFR 73.54(h) and the record retention and handling requirements specified in Section C.5 of this guide.*

to:

> *establishing the retention policy of all assessment documentation, including notes and supporting information, in accordance with 10 CFR 73.54(h) and the record retention and handling requirements specified in Section C.5 of this guide.*

This deviation is acceptable as the licensee has committed to establish the retention policy. Although this may be done by a different team, and not the CST, it is consistent with the intent of RG 5.71.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.11 RG 5.71, Section C.3.1.2, fourth paragraph, first sentence (page 13)

The VEGP CSP deviates from RG 5.71 by changing this sentence:

> *The licensee's CST needs to have the authority to conduct an objective assessment, make determinations that are not constrained by operational goals (e.g., cost),*

to:

> *The licensee's CST needs to have the authority to conduct an objective assessment, make determinations that are not constrained by business goals (e.g., cost),*

This deviation is acceptable because the intent of this statement in RG 5.71 is to ensure that cost is not used as a factor in making determinations about the adequacy of security controls, vulnerabilities, identifying CSs and CDAs, and carrying out other assessment functions of the CST.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.12 RG 5.71, Section C.3.1.3, second paragraph (page 14)

The VEGP CSP deviates from RG 5.71 by changing the identification process from CDAs to CSs. This deviation is acceptable because the VEGP CSP commits to continue identifying CSs by identifying digital computers, networks, communication systems and support systems that perform and are associated with SSEP functions, as well as support systems and equipment that, if compromised, would adversely impact the plant's SSEP functions.

This is consistent with the process in RG 5.71, which identifies CDAs through the same process. The licensee further describes CDAs as a CS or part of a CS; therefore, the use of the term CS as opposed to CDA is also consistent with the method used in RG 5.71.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.13 RG 5.71, Section C.3.1.3, fifth paragraph, first sentence (page 15)

The VEGP CSP deviates from RG 5.71 by making an editorial correction to RG 5.71. This involves changing:

> *With the identification of the all the CSs ...*

to:

> *With the identification of all the CSs ...*

This change is acceptable because it accomplishes the intent of this phrase in RG 5.71 eliminating the unnecessary "the."

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.14 RG 5.71, Section C.3.1.3, fifth paragraph, second sentence (page 15)

The VEGP CSP deviates from RG 5.71 by changing the following statement from:

> *A CDA may be a component of a CS ...*

to:

> *A CDA may be a complete CS or component of a CS, ...*

This deviation is acceptable because this statement is factually true. A CDA may be a complete CS and the deviation does not change the level of protection provided by the method outlined in RG 5.71.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.15 RG 5.71, Section C.3.1.3, fifth paragraph, fifth sentence (page 15)

The VEGP CSP deviates from RG 5.71 by including additional documentation to help identify CSs and CDAs. Specifically VEGP includes "other licensing basis" documents to identify CSs and CDAs.

This deviation is in line with the intent of using existing documentation to identify CSs and CDAs. This section of RG 5.71 describes "helpful information sources for identifying CSs and CDAs" and is not an exhaustive list, nor is it the only method SNC has committed to use to identify CSs and CDAs. Specifically, SNC has committed to identify all digital computers, networks and communication systems associated with SSEP functions, which is what 10 CFR 73.54 requires.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.16 RG 5.71, Section C.3.1.3, eighth paragraph, first bullet (page 16)

The VEGP CSP deviates from RG 5.71 by stating that CDAs may be an entire CS. As previously discussed in Section 13.8.4.24.14 of this SER, it is true that a CDA may be an entire CS; therefore, this definition does not adversely impact either the method used in RG 5.71 or the protection that RG 5.71 provides.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.17 RG 5.71, Section C.3.1.3, eighth paragraph, second bullet (page 16)

The VEGP CSP deviates from RG 5.71 by stating that CDAs may be an entire CS. As previously discussed in Sections 13.8.4.24.14 and 13.8.4.24.16 of this SER, it is true that a CDA may be an entire CS; therefore, this definition does not adversely impact either the method used in RG 5.71 or the protection that RG 5.71 provides.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.18 RG 5.71, Section C.3.2, first paragraph, first sentence (page 18)

The VEGP CSP deviates from RG 5.71 by providing an editorial correction to RG 5.71. Specifically, the VEGP CSP changes the following sentence from:

> *As stated in 10 CFR 73.54(c)(2), the licensee must design its cyber security program to apply and maintain integrate defense-in-depth protective strategies to ensure the capability to detect, prevent, respond to, mitigate, and recover from cyber attacks.*

to:

> *As stated in 10 CFR 73.54(c)(2), the licensee must design its cyber security program to apply and maintain integrated defense-in-depth protective strategies to ensure the capability to detect, prevent, respond to, mitigate, and recover from cyber attacks.*

This deviation captures the intent of this sentence in RG 5.71 by correcting "integrate" to "integrated."

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.19 RG 5.71, Section C.3.2, second paragraph, fourth sentence (page 18)

The VEGP CSP deviates from RG 5.71 by pointing to an editorial error in RG 5.71. Specifically, the VEGP CSP changes the following sentence from:

> *Therefore, defense-in-depth is achieved not only by implementing multiple security boundaries, but also by instituting and maintaining a robust program of security controls that assess, protect, respond, prevent, detect, and mitigates an attack on a CDA and with recovery.*

to:

> *Therefore, defense-in-depth is achieved not only by implementing multiple security boundaries, but also by instituting and maintaining a robust program of security controls that assess, protect, respond, prevent, detect, and mitigate an attack on a CDA and with recovery.*

This deviation captures the intent of this sentence in RG 5.71 by correcting "mitigates" to "mitigate." Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.20 RG 5.71, Section C.3.2, third paragraph, first sentence (page 18)

The VEGP CSP deviates from RG 5.71 by pointing to an editorial error in RG 5.71. Specifically, the VEGP CSP changes the following sentence from:

> *For example, if a failure in prevention were to occur (e.g., a violation of policy) or if protection mechanisms were to be bypassed (e.g., by a new virus that is not yet identified as a cyber attack), mechanisms would still in place to detect and respond to an unauthorized alteration in an impacted CDA, mitigate the impacts of this alteration, and recover normal operations of the impacted CDA before an adverse impact.*

to:

> *For example, if a failure in prevention were to occur (e.g., a violation of policy) or if protection mechanisms were to be bypassed (e.g., by a new virus that is not yet identified as a cyber attack), mechanisms would still be in place to detect and respond to an unauthorized alteration in an impacted CDA, mitigate the impacts of this alteration, and recover normal operations of the impacted CDA before an adverse impact.*

This is acceptable because the change to add the word "be" to the phrase "would still be in place to detect" captures the intent of this sentence by supplying the "be" omitted from RG 5.71.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.21 RG 5.71, Section C.3.2.1, Figure 5 (Page 19)

The VEGP CSP includes a defensive architecture, which deviates from the example provided in RG 5.71. The proposed architecture is acceptable because it provides defense-in-depth, communication isolation for safety and security systems, and multiple nondeterministic boundaries for nonsafety/nonsecurity CDAs. This provides adequate protection for CDAs and ensures that appropriate isolation and boundary protection exists for all CDAs where appropriate.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.22 RG 5.71, Section C.3.2.1, third paragraph (page 19)

The VEGP CSP deviates from RG 5.71 by modifying the characteristics of an acceptable defensive architecture by stating that the architecture includes CSs and CDAs configured in accordance with Section 5 of Appendix B, and Sections 6 and 7 of Appendix C in accordance with the security control application process described in Section 3.3. As previously discussed in Section 13.8.4.24.9 of this SER, the use of the security control application process to address controls is consistent with RG 5.71.

SNC has committed to apply the security control, demonstrate that alternative controls provide no less protection than the corresponding control, or demonstrate through analysis that the attack vector the control addresses does not exist; therefore, the control is not necessary.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.23 RG 5.71, Section C.3.2.1, third paragraph, first bullet (page 19)

The VEGP CSP deviates from RG 5.71 by modifying the example defensive architecture to match the architecture to be used in the AP1000. This deviation is acceptable because it provides the appropriate isolation of safety and security CDAs, and adequate boundaries for nonsafety/nonsecurity CDAs.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.24 RG 5.71, Section C.3.2.1, third paragraph, second bullet (page 19)

The VEGP CSP deviates from RG 5.71 by modifying the example defensive architecture to match the architecture to be used in the AP1000. As previously discussed in Section 13.8.4.6, this deviation is acceptable because it provides the appropriate isolation of safety and security CDAs, and adequate boundaries for nonsafety/nonsecurity CDAs. This is consistent with the defensive model in RG 5.71, as the VEGP defensive architecture provides boundaries for safety systems that are deterministic.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.25 RG 5.71, Section C.3.2.1, third paragraph, third bullet (page 19)

The VEGP CSP deviates from RG 5.71 regarding communications from digital assets at lower security levels to digital assets at higher security levels. This deviation is acceptable because the defensive architecture prevents specific *communication from lower security levels to specific higher security levels. This is consistent with the defensive model in RG 5.71.*

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.26 RG 5.71, Section C.3.2.1, third paragraph, new second bullet (page 19)

The VEGP CSP deviates from RG 5.71 regarding remote access. This is consistent with the guidance in Section C.7 of RG 5.71, which also states that remote access to CDAs at the highest level be prevented.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.27 RG 5.71, Section C.3.2.1, third paragraph, new sixth bullet (page 19)

The VEGP CSP deviates from RG 5.71 by including in its defensive architecture a statement from Section C.7 of RG 5.71 for validating data (software updates, new firmware, etc.) using a method at or above the level of security the CDA that will have data transferred to it. This concept is already acceptable in RG 5.71 and is also included in the defensive architecture, although in a different section of the document. This is consistent with the method used in RG 5.71 and does not adversely impact the protection provided.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable

13.8.4.24.28 RG 5.71, Section C.3.2.1, third paragraph, seventh bullet (page 19)

The VEGP CSP deviates from RG 5.71 by changing the commitment to eliminate applications, services and protocols not necessary to support the design-basis function of the CDAs to eliminate, disable, or render these inoperable. This is consistent with the method in RG 5.71, because in some cases these elements cannot be eliminated, but rather may have to be disabled or otherwise rendered inoperable. In each case, the result is the same. The asset is only configured to perform its design-based function and nothing more, which produces no less protection than the method in RG 5.71.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable. RG 5.71, Section C.3.2.1, third paragraph, new sixth bullet (page 19)

13.8.4.24.29 RG 5.71, Section C.3.2.1, third paragraph, eighth bullet (page 19)

The VEGP CSP deviates from RG 5.71 by eliminating the requirement to configure CDAs and boundary protection systems in accordance with Section 5 of Appendix B and Sections 6 and 7 of Appendix C. However, the VEGP CSP does commit to this in the preamble statement as described in

Section 13.8.4.24.22 of this SER. Therefore, the VEGP CSP provides the same commitment to perform this as does RG 5.71, albeit in a different part of the same section.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.30 RG 5.71, Section C.3.2.1, fourth paragraph (page 19

The VEGP CSP deviates from RG 5.71 by deleting the paragraph that commits to applying the security controls. However, the VEGP security plan commits, in Section 3.1.6, to address these controls and is, therefore, consistent with the method used in RG 5.71. The deleted paragraph is, therefore, unnecessary in the VEGP CSP to achieve the same commitment.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.31 RG 5.71, Section C.3.2.1, Prior to fifth paragraph (page 19)

The VEGP CSP deviates from the RG 5.71 defensive architecture. The VEGP architecture is described in Section 13.8.4.6 of this SER.

Based on the review and assessment in Section 13.8.4.6, the NRC staff finds that this deviation is acceptable.

13.8.4.24.32 RG 5.71, Section C.3.3, first paragraph, second sentence (page 20)

The VEGP CSP deviates from RG 5.71 by changing the following sentence:

> *A cyber compromise of CDAs would adversely impact nuclear facilities' SSEP functions that are necessary for protecting public health and safety.*

to:

> *A cyber compromise of CDAs could adversely impact nuclear facilities' SSEP functions that are necessary for protecting public health and safety.*

This deviation is consistent with the intent of RG 5.71, which implies that a compromise could lead to adverse impact and possible radiological sabotage. The intent of the paragraph is to establish the impact that could occur if a CDA were compromised. The security controls are designed around worst case scenarios, and the change in the VEGP CSP from "would" to "could" maintains this logic.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.33 RG 5.71, Section C.3.3, third paragraph, fourth sentence (page 20)

The VEGP CSP deviates from RG 5.71 by making an editorial correction to RG 5.71. This involves changing the statement:

> *Thus to provide high assurance that CDAs are protected from cyber attacks, potential cyber risks of these CDAs must be addressed known potential cyber risks.*

to:

> *Thus to provide high assurance that CDAs are protected from cyber attacks, potential cyber risks of these CDAs must be addressed for known potential cyber risks.*

This is acceptable because the change captures the intent of this sentence by supplying the "for" omitted from RG 5.71.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.34 RG 5.71, Section C.3.3, third paragraph, first sentence (page 20)

The VEGP CSP deviates from RG 5.71 by adding Appendix C to the list of controls that may be addressed using the method in Section 3.1.6 of Appendix A. This is consistent with the intent of RG 5.71, which assumes that all the controls in Appendix C can be implemented as written. However, if the controls can be addressed to demonstrate that an alternative control provides no less protection than the comparable control in Appendix C, or that the control is not necessary by demonstrating that the attack vector does not exist, this would meet the intent of RG 5.71.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.35 RG 5.71, Section C.3.3, third paragraph, first bullet (page 20)

The VEGP CSP deviates from RG 5.71 by adding Appendix C to the list of controls that may be addressed using the method in Section 3.1.6 of Appendix A. This is consistent with the intent of RG 5.71, which assumes that all the controls in Appendix C can be implemented as written. However, if the controls can be addressed to demonstrate that an alternative control provides no less protection than the comparable control in Appendix C, or that the control is not necessary by demonstrating that the attack vector does not exist, this would meet the intent of RG 5.71.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.36 RG 5.71, Section C.3.3, third paragraph, second bullet (page 20)

The VEGP CSP deviates from RG 5.71 by stating that alternative controls will not provide equal or better protection to the corresponding control, but rather that they will not provide less protection than the corresponding control. This is consistent with the method used in RG 5.71; providing an alternative that does not provide less protection, and does not adversely impact the security program. Therefore, this change in commitment will provide an adequate level of protection and is consistent with the method used in RG 5.71.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.37 RG 5.71, Section C.3.3, third paragraph, second bullet, second sub-bullet (page 20)

The VEGP CSP deviates from RG 5.71 by changing the statement:

> *performing and documenting the attack vector and attack tree analyses of the CDA and alternative countermeasures to confirm that the countermeasures provide the same or greater protection as the corresponding security control in Appendix B.*

to:

> *performing and documenting an attack vector and attack tree analysis of the CDA and alternative countermeasures to confirm countermeasures provide no decrease in the effectiveness of protection as compared to the corresponding security control identified in Appendix B or C.*

This deviation is acceptable because whether the licensee performs a single analysis or multiple analyses, the method is comparable provided that it will demonstrate that there is no decrease in protection. Further, the modification of the second part of the sentence is also acceptable because the intent of this method in RG 5.71 is to ensure that alternative controls do not provide less protection than the corresponding control. Therefore, a commitment to ensure that alternatives do not provide less protection produces a comparable level of protection as stating that the alternatives provide equal or better protection. Finally, the addition of the Appendix C controls to this method is acceptable because the licensee has committed to apply the control, apply an alternative that provides no less protection than the comparable control or not to apply the control and demonstrate that the attack vector does not exist.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.38 RG 5.71, Section C.3.3, third paragraph, second bullet, third sub-bullet (page 20)

The VEGP CSP deviates from RG 5.71 in a similar manner to deviations in Section 13.8.4.24.37 of this SER by changing the commitment to implement alternative countermeasures that provide at least the same degree of protection as the corresponding security control in Appendix B, to implementing alternative controls to provide no decrease in the effectiveness of protection as compared to the corresponding security control identified in Appendices B and C of RG 5.71.

This method is consistent with the method in RG 5.71 as it also meets the criteria for the performance based characteristics of 10 CFR 73.54. As long as the implemented alternative control does not provide less protection than the corresponding control in RG 5.71, the intent of this section of RG 5.71 has been met. Alternative controls are considered to be adequate only if they provide equivalent protection, and the VEGP CSP commits to that minimum standard.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.39 RG 5.71, Section C.3.3, third paragraph, third bullet (page 20)

The VEGP CSP deviates from RG 5.71 by not stating that SNC will specifically perform an attack vector and attack tree analysis to demonstrate that one of the specific security controls is not necessary. SNC does commit to performing an analysis to demonstrate that the attack vector does not exist (i.e., is not applicable), thereby obviating the need for a specific security control.

This method is consistent with the method in RG 5.71 as it commits to demonstrating a conclusion, specifically, that the attack vector does not exist. If the licensee can demonstrate this, and not use an attack vector or attack tree analysis, the results are still the same and, therefore, the method would produce a result that does not provide less protection than the method in RG 5.71.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.40 RG 5.71, Section C.3.3, fourth paragraph, second sentence (page 20)

The VEGP CSP deviates from RG 5.71 by making an editorial correction to RG 5.71. This involves changing the statement:

> *When a security control is determined to have an adverse affect, alternate controls should be used by the licensee to protect the CDA from cyber attack up to and including the DBT consistent with the process described above.*

to:

> *When a security control is determined to have an adverse effect, alternate controls should be used by the licensee to protect the CDA from cyber attack up to and including the DBT consistent with the process described above.*

This is acceptable because the change captures the intent of this sentence in RG 5.71, by correcting "affect" to "effect."

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.41 RG 5.71, Section C.3.3, fifth paragraph, second sentence (page 21)

The VEGP CSP deviates from RG 5.71 by making an editorial correction to RG 5.71. This involves changing the statement:

> *If these effectiveness or vulnerability analyses identify a gap in the cyber security program, the licensee may need to implement additional security measures and controls not provided in Appendixes B and C.*

to:

> *If these effectiveness or vulnerability analyses identify a gap in the cyber security program, the licensee may need to implement additional security measures and controls not provided in Appendices B and C.*

This change is acceptable because it captures the intent of this sentence in RG 5.71, by correcting "Appendixes" to "Appendices."

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.42 RG 5.71, Sections C.3.3.1.1 through C.3.3.1.5, first paragraph and last bullet (pages 21 and 22)

The VEGP CSP deviates from RG 5.71 by stating that it will not apply all of the security controls in RG 5.71, but rather will address them. The VEGP CSP already commits to the RG 5.71 process, which is:

a) applying controls;

b) applying an alternative control that does not provide less protection than the corresponding control; or

c) not applying a control, but demonstrating that the corresponding attack vector does not exist.

The intent of RG 5.71 is to address the controls in Appendices B and C. This can be accomplished in accordance with Section 3.1.6 of Appendix A, to which SNC has committed.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.43 RG 5.71, Section C.3.3.1.1, first paragraph, second bullet, fourth sub-bullet (page 21)

The VEGP CSP deviates from RG 5.71 by committing to audit CDAs at an interval defined for the CDA, or within 5 days following revocation of an individual's unescorted access, due to a lack of trustworthiness or reliability, or as soon as reasonably practical upon changes in personnel. Although this method uses a different frequency than the method in RG 5.71, which calls for annual assessments, or assessments immediately upon changes in personnel, this frequency does meet the requirements of 10 CFR 73.55(m), which allows the licensee to define these intervals based on its own assessments of need.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.44 RG 5.71, Sections C.3.3.2.1 through C.3.3.2.5, first paragraph and last bullet (pages 23 and 24)

The VEGP CSP deviates from RG 5.71 in a fashion similar to the deviation cited in Section 13.8.4.24.42 of this SER by committing not to apply the controls, but rather to address them. As previously stated, this deviation is consistent with the method in RG 5.71, and also meets the intent of the RG, provided that the licensee follows the process in Section 3.1.6 of Appendix A, to which SNC has committed.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.45 RG 5.71, Sections C.3.3.2.6 through C.3.3.2.9, first paragraph and last bullet (pages 24-26)

The VEGP CSP deviates from RG 5.71 in a fashion similar to the deviation cited in Sections 13.8.4.24.42 and 13.8.4.24.44 of this SER by committing to apply the controls, but rather to address them. As previously stated, this deviation is consistent with the method in RG 5.71, and also meets the intent of the RG, provided that the licensee follows the process in Section 3.1.6 of Appendix A, to which SNC has committed.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.46 RG 5.71, Section C.3.3.2.9, first paragraph, first bullet (page 25)

The VEGP CSP deviates from RG 5.71 by making an editorial correction to RG 5.71. This involves changing the first bullet:

- *develop, disseminate, and annually review and update the configuration management policy and program which defines the purpose of the nuclear facility's configuration management policy, scope, roles, requirements, responsibilities, and management commitments necessary to provide, with high assurance, that (1) when a modification to a CDA does not reduce the existing security and (2) any unauthorized or inadvertent modification of a CDA is prevented.*

to:

- *develop, disseminate, and annually review and update the configuration management policy and program which defines the purpose of the nuclear facility's configuration management policy, scope, roles, requirements, responsibilities, and management commitments necessary to provide, with high assurance, that (1) a modification to a CDA does not reduce the existing security and (2) any unauthorized or inadvertent modification of a CDA is prevented.*

This is acceptable because it captures the intent of this sentence in RG 5.71, by striking the word "when" after "(1)." This editorial mistake will be corrected in a future revision.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.47 RG 5.71, Section C.3.3.3.1, first paragraph and last bullet (page 26)

The VEGP CSP deviates from RG 5.71 in a fashion similar to the deviations cited in Sections 13.8.4.24.42, 13.8.4.24.44 and 13.8.4.24.45 of this SER, and by committing not to apply the controls, but rather to address them. As previously stated, this deviation is consistent with the method in RG 5.71, and also meets the intent of RG 5.71, provided that the licensee follows the process in Section 3.1.6 of Appendix A, to which SNC has committed.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.48 RG 5.71, Section C.3.3.3.1, second paragraph (page 26)

The VEGP CSP deviates from RG 5.71 by committing to Revision 1 of RG 1.152 and not Revision 2 of RG 1.152 as stated in RG 5.71. The results of the NRC *staff's technical evaluation of the digital instrumentation and controls design of the AP1000 are documented in Chapter 7 of NUREG-1793 and its supplements. SNC's use of the defensive architecture as discussed in Section 13.8.4.6 is acceptable to the staff.*

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.49 RG 5.71, Section C.3.3.3.2, first paragraph, second sentence (page 26)

The VEGP CSP deviates from RG 5.71 by committing to provide adequate protection of high assurance against cyber attacks. Although this commitment is worded differently than the commitment provided in RG 5.71, it does meet the requirement of 10 CFR 73.54(a), which states that licensees "shall provide high assurance that digital computer and communication systems and networks are adequately protected against cyber attacks, up to and including the design basis threat as described in 10 CFR 73.1."

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.50 RG 5.71, Section C.3.4, second paragraph, first sentence (page 26)

The VEGP CSP deviates from RG 5.71 as described in Section 13.8.4.8 of this SER by committing not to integrate management of physical and cyber security, but rather to provide the management interfaces necessary to appropriately coordinate the physical and cyber security activities. The VEGP CSP includes a commitment to establish an organization that is responsible for cyber security and is independent of operations. The combination of an independent organization responsible for cyber security, and management coordination between physical and cyber security meets the requirements of the rule and does not provide less protection than the method described in RG 5.71.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.51 RG 5.71, Section C.3.4, second paragraph, first bullet (page 27)

The VEGP CSP deviates from RG 5.71 as also described in Section 13.8.4.8 of this SER by committing not to form a unified security organization, but rather to establish a cyber security organization that is responsible for cyber security and is independent from operations. The combination of an independent organization responsible for cyber security, and management coordination as described in Section 13.8.4.24.50 of this SER between physical and cyber security meets the requirements of the rule, and does not provide less protection than the method described in RG 5.71.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.52 RG 5.71, Section C.4, first paragraph, first sentence (page 27)

The VEGP CSP deviates from RG 5.71 by changing the phrase:

> *Once the security program is in place...*

to:

> *Once the cyber security program is in place...*

This deviation is acceptable because the CSP only applies to the applicant's cyber security program.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.53 RG 5.71, Section C.4, first paragraph, first bullet (page 28)

The VEGP CSP deviates from RG 5.71 as previously described in Section 13.8.4.11 of this SER by changing the phrase "continuous monitoring and assessment" to "ongoing monitoring and assessment." This description is consistent with the method in RG 5.71 by establishing intervals for these assessments, which include the same elements as in RG 5.71, and meeting the periodicity requirements of 10 CFR 73.55(m).

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.54 RG 5.71, Section C.4.1, section heading and first paragraph, first sentence (page 28)

The VEGP CSP deviates from RG 5.71 as previously described in Sections 13.8.4.11 and 13.8.4.24.53 of this SER by changing the phrase "continuous monitoring and assessment" to "ongoing monitoring and assessment." This description is consistent with the method in RG 5.71 by establishing intervals for these assessments, which include the same elements in RG 5.71 and meeting the periodicity requirements of 10 CFR 73.55(m).

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.55 RG 5.71, Section C.4.1, second paragraph, first sentence (page 28)

The VEGP CSP deviates from RG 5.71 as previously described in Sections 13.8.4.11, 13.8.4.24.53 and 13.8.4.24.54 of this SER by changing the phrase "continuous monitoring and assessment" to "ongoing monitoring and assessment." This description is consistent with the method in RG 5.71 by establishing intervals for these assessments, which include the same elements as in RG 5.71 and meeting the periodicity requirements of 10 CFR 73.55(m).

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.56 RG 5.71, Section C.4.1, second paragraph, first bullet (page 28)

The VEGP CSP deviates from RG 5.71 by making an editorial correction to RG 5.71. This involves changing the phrase:

> *ongoing assessments of verify that the security controls...*

to:

> *ongoing assessments to verify that the security controls."*

This change is acceptable because it captures the intent of this sentence in RG 5.71, by substituting "to" for "of."

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.57 RG 5.71, Section C.4.1, third paragraph, first and second sentences (page 28)

The VEGP CSP deviates from RG 5.71 as previously described in Sections 13.8.4.11, 13.8.4.24.53, 13.8.4.24.54 and 13.8.4.24.55 of this SER by changing the phrase "continuous monitoring and assessment" to "ongoing monitoring and assessment." This description is consistent with the method in RG 5.71 by establishing intervals for these assessments, which include the same elements as in RG 5.71, and meeting the periodicity requirements of 10 CFR 73.55(m).

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.58 RG 5.71, Section C.4.1.1, first paragraph, second sentence (page 28)

Section 3.1.1 of the VEGP CSP states that status of security controls will be verified in accordance with the requirements of 10 CFR 73.55(m).

The NRC staff reviewed the above and found that reviewing security controls in accordance with 10 CFR 73.55(m) is in accordance with RG 5.71. The time period between evaluations may be longer than the time period provided in RG 5.71. However, this period cannot exceed 24 months, which conforms to 10 CFR 73.54(g), requiring the applicant to review the cyber security program as a component of the physical security program in accordance with the requirements of 10 CFR 73.55(m), including the periodicity requirements. The requirements of 10 CFR 73.55(m) are that, at minimum, the applicant review each element of the physical protection program at least every 24 months.

The licensee has also committed to address C.13 of Appendix C to RG 5.71, "Security Assessment and Risk Management," which calls for vulnerability assessments on a quarterly basis. SNC commits to apply this control, apply an alternative that provides no less protection than C.13, or demonstrate that any attack vectors associated with vulnerabilities that may be discovered through quarterly assessments do not exist. The VEGP CSP also includes addressing controls that specifically include defined verification periods and that detect when some controls are not working correctly.

This, coupled with the CSP conforming to requirements of 10 CFR 73.55(m), which includes an initial assessment within 12 months of the program inception, and as necessary based on site-specific analyses, assessments, or other performance indicators, provides a level of protection consistent with the method in RG 5.71.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.59 RG 5.71, Section C.4.1.2, first paragraph, third sentence (page 29)

Section 3.1.1 of the VEGP CSP states that effectiveness of security controls will be verified in accordance with the requirements of 10 CFR 73.55(m). As previously discussed in Section 13.8.4.12 of this SER, the NRC staff reviewed the above and found that the period of effectiveness analysis is comparable with that of RG 5.71.

The time period between evaluations is 12 months longer than the time period provided in RG 5.71. However, this 24-month time period conforms to 10 CFR 73.54(g) requiring the applicant to review the cyber security program as a component of the physical security program in accordance with the requirements of 10 CFR 73.55(m), including the periodicity requirements. The requirements of 10 CFR 73.55(m) are that, at minimum, the applicant review each element of the physical protection program, which includes the cyber security program, at least every 24 months and within 12 months of the implementation of the program, or within 12 months when changes that may adversely impact the security program occur.

Furthermore, the VEGP CSP states that controls will be reviewed according to the requirements of the security controls if that period of review occurs more often. This is also consistent with the method provided in RG 5.71.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.60 RG 5.71, Section C.4.1.3, first paragraph, second sentence (page 29)

VEGP CSP Section 4.1.3 deviates from RG 5.71 by stating that vulnerability assessments will occur periodically. RG 5.71, Section C.4.1.3 states that vulnerability assessments will occur no less frequently than on a quarterly basis.

As previously described in Section 13.8.4.14 of this SER, the VEGP CSP states vulnerability assessments will be performed as specified in the security controls in Appendices B and C of RG 5.71, and when new vulnerabilities that could affect the effectiveness of the cyber security program and the security of the CDAs are identified. The licensee also commits to addressing vulnerabilities that could cause CDAs to become compromised or could have an adverse impact on SSEP functions. Section 13.1 of Appendix C of RG 5.71, which VEGP commits to address in accordance with the process in Section 3.1.6 of Appendix A, provides that vulnerability assessments should occur no less frequently than once a quarter, at random intervals, and when new potential vulnerabilities are reported and identified. SNC has not deviated from the interval.

The process the applicant has committed to in Section 3.1.6 of the VEGP CSP requires SNC, if it does not implement Section 13.1 of Appendix C, to implement an alternate control that does not provide less protection than the corresponding control in Appendices B and C, or to demonstrate that any attack vectors associated with vulnerabilities that may be discovered through quarterly assessments do not exist.

Therefore, if SNC does not implement the security control in Appendix C, Section 13.1 of RG 5.71, or deviates from the guidance for a quarterly vulnerability assessment, it will ensure that this deviation does not provide less protection than performing quarterly vulnerability assessments, and will provide an analysis that demonstrates that the attack vector does not exist and will document this justification for inspection.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.61 RG 5.71, Section C.4.2, first paragraph, second sentence (page 30)

The VEGP CSP deviates from RG 5.71 by committing not to implement the security controls in Section 11 of Appendix C of RG 5.71, but rather to address those controls in accordance with Section C.3.3 of RG 5.71.

As previously described in Section 13.8.4.7 of this SER, the VEGP CSP deviates from RG 5.71 by committing to address security controls rather than committing to apply them. The VEGP CSP states that when a control from Appendices B and C of RG 5.71, such as Section 11 of Appendix C, is not implemented that the licensee will implement alternate control(s) that "do not provide less protection that the corresponding" control in the appendix. This *deviation is consistent with the method used in RG 5.71, which states that controls should provide equal or better protection.*

As also previously discussed in Section 13.8.4.7 of this SER, the VEGP CSP deviates from RG 5.71 by stating that when a control can be proven to be unnecessary, the applicant will perform an analysis demonstrating that the control is not necessary, and will provide a documented justification. Therefore, SNC commits that in addressing the security controls in Appendix C, Section 11 of RG 5.71 that it will either apply the control, apply an alternative that does not provide less protection or will demonstrate that the control is not necessary because the attack vectors do not exist. This method is consistent with the method used in RG 5.71, which also allows for controls to be addressed.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.62 RG 5.71, Section C.4.2.1, first paragraph, third sentence (page 30)

The VEGP CSP deviates from RG 5.71 in a manner similar to the previous deviation in Section 13.8.4.24.61 of this SER. Specifically, that configuration management will be used to ensure that each of the controls is addressed in Appendices B and C of RG 5.71, as opposed to implemented. This method is consistent with the method in RG 5.71, as the applicant commits to follow the process in Section C.3.3 of RG 5.71, which requires that the applicant implement the control, apply an alternative control that does not provide less protection than the corresponding control in RG 5.71, or demonstrate that the attack vector associated with the control does not exist. Therefore, the VEGP CSP method will provide no less protection than the method provided for in RG 5.71.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.63 RG 5.71, Section C.4.2.1, second paragraph, third sentence (page 30)

The VEGP CSP deviates from RG 5.71 by including the statement, "in accordance with the process described in Section C.3.3 of this guide." As previously discussed in Section 13.8.4.14 of this SER, the method in Section C.3.3 is consistent with the method in RG 5.71, which requires that the licensee either implement the control, apply an alternative control that does not provide less protection than the corresponding control in RG 5.71, or demonstrate that the attack vector associated with the control does not exist. Therefore, the VEGP CSP method will provide no less protection than the method provided for in RG 5.71.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.64 RG 5.71, Section C.4.3, second paragraph (page 31)

The VEGP CSP deviates from RG 5.71, as previously discussed in Section 13.8.4.22 of this SER, by stating that the applicant has established the necessary measures and governing procedures to implement periodic reviews of applicable program elements, in accordance with the requirements of 10 CFR 73.55(m). Specifically, the VEGP CSP calls for a review of the program's effectiveness at least every 24 months. In addition, reviews are to be conducted as follows:

- *within 12 months following initial implementation of the program*
- *as necessary based upon site-specific analyses, assessments, or other performance indicators*
- *as soon as reasonably practical, but no longer than 12 months, after changes occur in personnel, procedures, equipment, or facilities that potentially could adversely affect cyber security*
- *by individuals independent of those personnel responsible for program management and any individual who has direct responsibility for implementing the program*

This deviates from RG 5.71 in the specific wording, but includes the same commitments as RG 5.71. Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.65 RG 5.71, Section C.5, second paragraph, second and third sentences (page 32)

As previously discussed in Section 13.8.4.23, the VEGP CSP deviates from RG 5.71 documentation retention commitments. Specifically, VEGP CSP Section 5 states the records are retained to document access history and information needed to discover the source of cyber attacks and incidents. The VEGP CSP deletes the phrase:

> *Records required for retention include, but are not limited to, digital records, log files, audit files, and nondigital records that capture, record, and analyze network and CDA events.*

The VEGP CSP commits to retaining all access history records, records to discover the source of cyber attacks or other security-related incidents affecting CDAs or SSEP functions, or both. This is consistent with what is included in RG 5.71 Section 5, as it includes all the performance-based characteristics and commitments of that section.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.66 RG 5.71, Glossary (Page 35)

The VEGP CSP's definition of a CDA deviates from the definition provided in RG 5.71. Specifically, the VEGP CSP deviates by stating that a CDA can be a CS or a subcomponent of a CS. This definition does not materially change the use of the term, and is correct: A CDA can be a CS. This definition is consistent with the definition in RG 5.71. The VEGP CSP, by the use of this definition, does not provide for less protection than RG 5.71, nor does this reduce the scope of the assets required to be protected under the rule.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.67 RG 5.71, Glossary (Page 35)

The VEGP CSP deviates from the definition of a CS in RG 5.71 by adding the caveat "as defined by the plant licensing basis." RG 5.71 states that a CS is an analog or digital technology based system in or outside the plant that performs or is associated with a safety-related, important-to-safety, security, or emergency preparedness function. These CSs include, but are not limited to, plant systems, equipment, communication systems, networks, offsite communications, or support systems or equipment, that perform or are associated with safety-related, important-to-safety, security, or emergency preparedness functions.

The addition of the phrase "as defined by the plants' licensing basis," limits the scope of the functions to those that are defined by the licensing basis. As previously discussed in Section 13.8.4.4 of this SER, the staff ~~was concerned that this modifier might cause the licensee to exclude CSs, which ought to be included, according to the rule~~ [found this modification acceptable]. *10 CFR 73.51(a)(1) requires that the licensee protect digital computer and communication systems and networks associated with: (i) safety-related and important-to-safety functions; (ii) security functions; (iii) emergency preparedness functions, including offsite communications; and (iv) support systems and equipment, which if compromised would adversely impact SSEP functions. However, further reviews resulted in the staff finding that the VEGP CSP scoping discussion adequately described a process to include all CDAs within the scope of 10 CFR 73.54(a)(1).*

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.68 RG 5.71, Glossary (Page 35)

The VEGP CSP deviates from the RG 5.71 definition of cyber attack by replacing the phrase "conducted by threat agents having either malicious or non-malicious intent" with the phrase "conducted by threat agents." The NRC staff finds this deviation to be acceptable because deletion of the intent of a threat agent, be it malicious or non-malicious, still provides a commitment to protect against threats by threat agents.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

13.8.4.24.69 RG 5.71, Appendix A, Introduction (Page A-1)

The VEGP CSP deviates from the RG 5.71 scope discussion by including within scope systems or equipment that perform important to safety functions including SSCs in the BOP that could directly or indirectly affect reactivity at a nuclear power plant and could result in an unplanned reactor shutdown or transient. Additionally, these SSCs are under the licensee's control and include electrical distribution equipment out to the first inter tie with the offsite distribution system. The NRC staff finds this deviation to be acceptable because it is consistent with Commission policy.

Based on the above review and assessment, the NRC staff finds that this deviation is acceptable.

License Conditions

- *Part 10, License Condition 2, COL Item 13.6-5 and License Condition 3, Item G.10*

The applicant proposed two license conditions in Part 10 of the VEGP COL application, which will require the applicant to implement the cyber security program prior to initial fuel load.

In a letter dated October 22, 2010, the applicant provided supplemental information which proposed to amend the milestone included in Part 2, FSAR Table 13.4-201 to implement the cyber security program prior to receipt of fuel onsite (protected area.) The NRC staff finds the proposed implementation milestone for the cyber security program (security prior to receipt of fuel onsite (protected area)) appropriate and in accordance with the requirement in 10 CFR 73.55(a)(4). Therefore the staff finds that the proposed License Conditions 2 and 3 are not necessary.

- *Part 10, License Condition 6*

The applicant proposed a license condition in Part 10 of the VEGP COL application to provide a schedule to support the NRC's inspection of operational programs, including the cyber security program. Although the CSP is not identified as an operational program in SECY-05-0197, the proposed license condition is consistent with the policy established in SECY-05-0197 for operational programs in general, and is acceptable.

13.8.5 Post Combined License Activities

For the reasons discussed in the technical evaluation section above, the staff finds the following license condition proposed by the applicant acceptable:

- License Condition (13-14) – No later than 12 months after issuance of the COL, the licensee shall submit to the Director of NRO a schedule that supports planning for and conduct of NRC inspection of the cyber security program implementation. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until the cyber security program has been fully implemented.

13.8.6 Conclusion

The staff reviewed the WLS COL application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to cyber security, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the CSP for format and content using the NRC CSP template in RG 5.71, and noted that it includes all features considered essential to such a program. In particular, the staff noted it complies with applicable NRC regulations including 10 CFR 73.1, 10 CFR 73.54, 10 CFR 73.55(a)(1), 10 CFR 73.55(b)(8), 10 CFR 73.55(m), and 10 CFR Part 73, Appendix G.

14.0 INITIAL TEST PROGRAM

The initial test program covers structures, systems, and components (SSCs) and design features for both the nuclear portion of the facility and the balance of plant. The information provided addresses the major phases of the test program, including preoperational tests, initial fuel loading and initial criticality, low-power tests, and power ascension tests. The scope of the initial test program, as well as general plans for accomplishing the test program, are described in sufficient detail to demonstrate that due consideration has been given to matters that normally require advance planning.

The technical aspects of the initial test program are described in sufficient detail to show that: (1) the test program adequately verifies the functional requirements of plant SSCs; and (2) the sequence of testing is such that the safety of the plant does not depend on untested SSCs. In addition, measures are described to ensure that: (1) the initial test program is accomplished with adequate numbers of qualified personnel; (2) adequate administrative controls will be established to govern the initial test program; (3) the test program is used, to the extent practicable, to train and familiarize the plant's operating and technical staff in the operation of the facility; and (4) the adequacy of plant operating and emergency procedures is verified, to the extent practicable, during the period of the initial test program.

This chapter also provides information on the inspections, tests, analyses and acceptance criteria (ITAAC) that are proposed to demonstrate that, when the ITAAC are performed and the acceptance criteria met, the facility has been constructed and will operate in conformance with the combined license (COL), the Atomic Energy Act, and Nuclear Regulatory Commission (NRC) regulations.

14.1 Specific Information to be Included in Preliminary/Final Safety Analysis Reports (Related to RG 1.206, Section C.III.1, Chapter 14, C.I.14.1, "Specific Information To Be Addressed for the Initial Plant Test Program")

Section 14.1 of the William States Lee III Nuclear Station (WLS) COL Final Safety Analysis Report (FSAR), Revision 11, incorporates by reference, with no departures or supplements, Section 14.1, "Specific Information to be Included in Preliminary/Final Safety Analysis Reports," of Revision 19 of the AP1000 Design Control Document (DCD). The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," and its supplements.

[1] See Section 1.2.2 for a discussion of the staff's review related to verification of the scope of information to be included in a COL application that references a design certification (DC).

14.2 Specific Information to be Included in Standard Safety Analysis Reports (Related to RG 1.206, Section C.III.1, Chapter 14, C.I.14.2, “Initial Plant Test Program”)

14.2.1 Summary of Test Program and Objectives

14.2.1.1 *Introduction*

This section describes the major phases of the initial test program as well as the general prerequisites and specific objectives to be achieved for each phase.

14.2.1.2 *Summary of Application*

Section 14.2 of the WLS COL FSAR, Revision 11, incorporates by reference Section 14.2 of the AP1000 DCD, Revision 19. Section 14.2 of the DCD includes Section 14.2.1.

In addition, in WLS COL FSAR Section 14.2.1, the applicant provided the following:

AP1000 COL Information Item

- STD COL 14.4-3

The applicant provided additional information in standard (STD) COL 14.4-3 to address the COL holder’s responsibility for development of a site-specific startup administrative manual (procedure) that will include the administrative procedures and requirements that will govern the activities associated with the plant’s initial test program. Also added was information related to first of a kind testing features.

Additionally, the applicant described how the initial test program is applied to the facility. This information was provided to supplement the information incorporated by reference from the AP1000 DCD.

14.2.1.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the test program summary and objectives are given in Section 14.2 of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition).”

The applicable regulatory requirements for the information being reviewed in this section are Title 10 of the *Code of Federal Regulations* (10 CFR) 52.79(a)(28) and Criterion XI of Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants” to 10 CFR Part 50, “Domestic licensing of production and utilization facilities.”

Regulatory Guide (RG) 1.68, Revision 3, "Initial Test Program for Water-Cooled Nuclear Power Plants," provides guidance on how to comply with Criterion XI of Appendix B to 10 CFR Part 50.

14.2.1.4 *Technical Evaluation*

The NRC staff reviewed Section 14.2.1 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the initial test program summary and objectives. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this safety evaluation report (SER) provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and to use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (Vogtle Electric Generating Plant (VEGP), Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from requests for additional information (RAIs).

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the Bellefonte Nuclear Plant (BLN), Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from Section 14.2.1.4 of the VEGP SER:

> *AP1000 COL Information Item*
>
> - *STD COL 14.4-3*
>
> *The NRC staff reviewed STD COL 14.4-3 related to COL Information Item 14.4-3 included in the VEGP COL FSAR. The applicant provided additional information*

to address COL Information Item 14.4-3 and to supplement the information addressed in the AP1000 DCD.

COL Information Item 14.4-3 states:

> *The Combined License holder is responsible for a site-specific startup administration manual (procedure), which contains the administration procedures and requirements that govern the activities associated with the plant initial test program, as identified in Subsection 14.2.3.*

This commitment was also captured as COL Action Item 14.4-3 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:

> *The COL applicant is responsible for preparing a startup administrative manual which contains the administrative procedures and standards that govern the activities associated with the plant initial test program.*

STD COL 14.4-3 was not explicitly evaluated in Section 14.2.1.4 of the BLN SER. However, portions of the evaluation material in Section 14.2.1.4 of the BLN SER are directly applicable to this COL item. Therefore, the NRC staff used this evaluation material, identified below as standard content material, in the disposition of STD COL 14.4-3, as it relates to the initial test program summary and objectives.

The staff reviewed Section 14.2.1 and requested that as part of RAI 14.2-12, dated December 8, 2008, the applicant describe how the BLN test program meets the objectives in Section 14.2.1 of the AP1000 DCD, Revision 17. In its January 22, 2009, response to this RAI, the applicant proposed to revise Section 14.2.1 of the BLN COL FSAR to supplement Section 14.2.1 of the AP1000 DCD, Revision 17. The applicant stated in its response that Section 14.2 of the BLN COL FSAR describes the controls that will be implemented in the site-specific startup administrative manual (procedure). The applicant also described the testing of first-of-a-kind design features and the use of operating experience (OE) from previous first-of-a-kind tests performed on other AP1000 plants. Additionally, the applicant proposed to develop administrative controls for crediting previously performed testing of first-of-a-kind AP1000 design features.

The staff determined that the proposed changes adequately clarify the objectives of the initial test program, consistent with the guidance in RG 1.68. Therefore, the staff finds this change acceptable. The applicant will revise the BLN COL FSAR to include the proposed administrative controls. This item is identified as ***Confirmatory Item 14.2-1****, pending NRC review and approval of the revised BLN COL FSAR.*

Resolution of Standard Content Confirmatory Item 14.2-1

The staff verified that the VEGP applicant has incorporated into its FSAR the proposed administrative controls identified as Confirmatory Item 14.2-1 in the staff's SER for the BLN COL. On this basis, Confirmatory Item 14.2-1 is resolved.

14.2.1.5 *Post Combined License Activities*

There are no post-COL activities related to this section.

14.2.1.6 *Conclusion*

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the application addressed the required information relating to the initial test program summary and objectives and there is no outstanding information to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the information presented in the WLS COL FSAR is acceptable because it meets the requirements of 10 CFR 52.79(a)(28) and Criterion XI of Appendix B to 10 CFR Part 50. The staff based its conclusions on the following:

- STD COL 14.4-3 is acceptable because it provides an adequate description of the administrative requirements associated with the test program objectives that will be implemented during the conduct of the initial test program.

14.2.2 Organization, Staffing, and Responsibilities (Related to RG 1.206, Section C.III.1, Chapter 14, C.I.14.2.2, "Organization and Staffing")

14.2.2.1 *Introduction*

The organization used to manage, supervise, or execute all phases of the initial test program is described. This description includes the organizational responsibilities and authorities, the degree of participation of each organizational unit in the implementation of the initial test program, and personnel training, experience, and qualification requirements.

14.2.2.2 *Summary of Application*

Section 14.2 of the WLS COL FSAR, Revision 11, incorporates by reference Section 14.2 of the AP1000 DCD, Revision 19. Section 14.2 of the DCD includes Section 14.2.2.

WLS COL FSAR Section 14.2.2 addresses the plant test and operations organization (PT&O) and other organizations that will participate in the implementation of the initial test program.

In addition, in WLS COL FSAR Section 14.2.2, the applicant provided the following:

AP1000 COL Information Item

- STD COL 14.4-1

The applicant provided additional information in STD COL 14.4-1 to provide a description of the organization, staffing, and responsibilities related to the initial test program.

14.2.2.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the organization, staffing, and responsibilities are given in Section 14.2 of NUREG-0800.

The applicable regulatory requirements for the information being reviewed in this section are 10 CFR 52.79(a)(28) and Criterion XI of Appendix B to 10 CFR Part 50. RG 1.68 provides guidance on how to comply with Criterion XI of Appendix B to 10 CFR Part 50.

14.2.2.4 *Technical Evaluation*

The NRC staff reviewed Section 14.2.2 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the initial test program organization, staffing, and responsibilities. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the BLN Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from Section 14.2.2.4 of the VEGP SER:

> *<u>AP1000 COL Information Item</u>*
>
> - *STD COL 14.4-1*
>
> *The NRC staff reviewed STD COL 14.4-1 related to COL Information Item 14.4-1 included under Section 14.2.2 of the BLN COL FSAR. The applicant provided information to replace the existing information in AP1000 DCD Section 14.2.2 with a description of the organization, staffing, and responsibilities related to the initial test program. This information was provided to address COL Information Item 14.4-1 in the AP1000 DCD, Revision 17. COL Information Item 14.4-1 states:*
>
> > *The specific staff, staff responsibilities, authorities, and personnel qualifications for performing the AP1000 initial test program are the responsibility of the Combined License applicant. This test organization is responsible for the planning, executing, and documenting of the plant initial testing and related activities that occur between the completion of plant/system/component construction and commencement of plant commercial operation. Transfer and retention of experience and knowledge gained during initial testing for the subsequent commercial operation of the plant is an objective of the test program.*
>
> *This commitment was also captured as COL Action Item 14.4-1 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:*
>
> > *The COL applicant will establish the specific staff, staff responsibilities, authorities, and personnel qualifications for performing the AP1000 initial test program.*
>
> *To address STD COL 14.4-1, the applicant described the PT&O organization in Section 14.2.2 of the BLN COL FSAR. The applicant stated that the PT&O organization will be responsible for the implementation of the initial test program, including the construction and installation, preoperational, and startup testing phases. In addition, the applicant described the responsibilities, interfaces, and authorities of the positions in the PT&O organization, including the following:*

- *Manager in charge of the PT&O organization, responsible for staffing the PT&O organization, developing procedures for the preoperational and startup test phases, managing the initial test program, implementing the initial test program schedule, and manage contracts associated with the initial test program.*

- *Functional Manager in charge of the PT&O support, responsible for the implementation of plans, schedules, and development and approval of test procedures.*

- *PT&O Engineers, responsible for the development of system test procedures.*

- *Functional manager in charge of startup, responsible for the management of preoperational and startup testing. Activities include participation in the Joint Test Working Group (JTWG), preparation of the detailed schedule for preoperational and startup test activities, coordination of vendor participation in the initial test program, supervising and directing startup engineers, and developing periodic progress reports.*

- *Startup Engineers, responsible for coordinating testing activities, identifying special or temporary equipment or services needed to support testing, ensuring compliance with administrative controls, and reviewing and evaluating test results.*

- *PT&O organization personnel qualifications and training program description.*

The staff reviewed the applicant's proposed resolution to COL Information Item 14.4-1 addressing organizational and staffing responsibilities for the initial test program. In its review, the staff identified areas where additional information was needed.

In RAIs 14.2-5 and 14.2-6, dated May 15, 2008, the staff requested that the applicant supplement the information incorporated by reference from Section 14.2.2 of the AP1000 DCD, Revision 17, and provide a description of the responsibilities, authorities, interfaces, and qualifications requirements of the organizations responsible for the overall administration of the initial test program, consistent with the guidance in RG 1.206 and Section 14.2 of NUREG-0800. In its response to RAIs 14.2-5 and 14.2-6, dated June 26, 2008, the applicant stated that Section 14.4 of the BLN COL FSAR incorporated by reference Section 14.4.3 of the AP1000 DCD and no further changes to the BLN COL FSAR were needed. However, the staff determined that the information included in BLN COL FSAR was insufficient. Therefore, the staff asked the applicant in RAI 14.2-12, dated December 8, 2008, to provide information regarding the organization(s) that will be in charge of the overall administration, technical direction, coordination, and implementation of the initial test program.

Specifically, the staff requested that the applicant provide organizational descriptions of the principal management positions (including any augmenting organizations) responsible for planning, executing, and documenting preoperational and startup testing activities. RAI 14.2-12 stated that this description should include the authorities, responsibilities and interfaces, and the degree of participation of each identified organizational unit. Additionally, the staff requested that the applicant describe training and qualification requirements for organizations responsible for implementing the initial test program.

In its response to RAI 14.2-12 dated January 22, 2009, the applicant proposed to include in Section 14.2.2 of the BLN COL FSAR, a description of the following organizational groups that will participate in the implementation of the initial test program:

- *The JTWG, including details of the key responsibilities, authorities, and interfaces*
- *The Site Construction Group (Architect-Engineer), including participating organizations, authorities, interfaces, and functional responsibilities*
- *The Site Preoperational Test Group, including participating organizations, authorities, interfaces, and functional responsibilities*
- *The Site Startup Test Group, including participating organizations, authorities, interfaces, and functional responsibilities*

In addition, the applicant proposed to include information related to the education, training, experience, and qualification requirements of supervisory personnel, test personnel, and other major participating organizations responsible for implementing the initial test program and developing testing, operating, and emergency procedures. This description would include administrative provisions for the establishment of a training program consistent with the criteria described in Three Mile Island (TMI) Action Plan Item I.G.1, (NUREG-0737, "Clarification of TMI Action Plan Requirements") and considerations for staffing effects that could result from overlapping initial test programs at multi-unit sites.

The staff reviewed the proposed organizational description provided by the applicant as part of the response to RAI 14.2-12. The applicant proposed to describe its overall responsibility for the conduct of the initial test program and also proposed to include a description of the major organizations that will be responsible for the administration and technical direction of the initial test program. To this end, the applicant proposed to include in Section 14.2.2.3 of the BLN COL FSAR the functions, responsibilities, and composition of the JTWG. Specifically, the JTWG will be composed of representatives from the plant's operations group, Westinghouse, the Architect-Engineer, and representatives from the test support groups. The applicant proposed to include a description of *the responsibilities, authorities, and interfaces of these organizations. The JTWG will provide oversight of the implementation of the initial test program, including planning, scheduling, and performance of preoperational and startup testing. Also, the JTWG will review, evaluate, and approve administrative and test procedures, and will review and evaluate construction, preoperational, and startup test results and test turnover packages. The applicant proposed to revise the BLN COL FSAR to include the proposed organizational description.*

Additionally, the applicant proposed to include a description of the responsibilities, authorities, and interfaces of supporting organizations including the Site Construction Group (Architect-Engineer), the Site Preoperational Test Group, and the Site Startup Test Group. A description of each proposed test group follows.

Section 14.2.2.4 of the BLN COL FSAR would be revised to describe the Site Construction Group (Architect-Engineer). The Site Construction Group will be composed, as necessary, of members from the construction group, the construction services group, the construction services procurement group, and the construction services quality group. The Site Construction Group will provide oversight of construction installation and testing, vendor interface and procurement associated with support testing activities, and turnover of tested equipment, systems, and testing documentation to the Site Preoperational Test Group.

Section 14.2.2.5 of the BLN COL FSAR would be revised to describe the Site Preoperational Test Group. The Site Preoperational Test Group will consist of engineering leads and preoperational test teams, and will accept turnover of systems and equipment from the construction organization, and plan, scope, schedule, and oversee testing of plant systems. Additionally, the Site Preoperational Test Group will coordinate tagging and maintenance of systems, will provide coordination with other participating organizations, and will resolve open items and exceptions identified during the implementation of the preoperational test program.

Section 14.2.2.6 of the BLN COL FSAR would be revised to describe the Site Startup Test Group. The Site Startup Test Group will include engineering leads and startup test teams, and will be responsible for the acceptance of SSCs for integrated testing. In addition, the Site Startup Test Group will manage and oversee the testing of plant SSCs to support the plant power ascension test program, and will accept and turn over startup test packages to the site licensee.

The applicant also proposed to include information in Section 14.2.2.2 of the BLN COL FSAR to address training and qualification requirements for individuals and organizations implementing the initial test program. The response stated that the training organization will develop procedures to implement a training and qualification program in accordance with the requirements of the licensee quality assurance program and in coordination with Westinghouse. This training and *qualification program will be used to confirm that test personnel have adequate training, qualification, and certification. In addition, the proposed training and qualification program will confirm that experienced and qualified personnel are available to develop testing, operating, and emergency procedures. The proposed training and qualification program will also provide supplemental operator training in accordance with TMI Action Plan Item I.G.1. The response stated that the site-specific startup administrative manual will contain measures to verify that personnel formulating and conducting test activities are not the same personnel who designed or are responsible for satisfactory performance of systems or design features under test. In addition, the startup administrative manual will provide controls for the consideration of staffing effects that could result from overlapping initial test programs at multi-unit sites.*

The staff determined that the proposed changes adequately define the organizations that will carry out the initial test program, describe the authorities, responsibilities, and interfaces, and delineate training and qualification requirements for organizations participating in the implementation of the initial test program, consistent with the guidance in RG 1.68. Additionally, Section 1.0, Table 1.9-201 of the BLN COL FSAR includes a commitment to RG 1.8, Revision 3, "Qualification and Training of Personnel for Nuclear Power Plants," which provides training and qualification requirements for nuclear power plant personnel, including personnel participating in initial test program activities. The applicant will revise the BLN COL FSAR to include the proposed administrative controls. Therefore, the staff finds this change acceptable. This is identified as ***Confirmatory Item 14.2-2****, pending NRC review and approval of the revised BLN COL FSAR.*

Resolution of Standard Content Confirmatory Item 14.2-2

The staff verified that the VEGP applicant has incorporated into its FSAR the proposed administrative controls identified as Confirmatory Item 14.2-2 in the staff's SER for the BLN COL. On this basis, Confirmatory Item 14.2-2 is resolved.

Evaluation of Additional Information

In its letter dated November 11, 2010, the VEGP applicant provided additional information on the training and qualification requirements for nonsupervisory test engineers participating in initial test program activities. In the standard content evaluation presented above for STD COL 14.4-1, the staff notes that RG 1.8 is referenced by the applicant as providing the training and qualification requirements for nuclear power plant personnel, including personnel participating in initial test program activities. In the November 11, 2010, letter, the applicant stated that VEGP COL FSAR Section 14.2.2.2 would be revised to state that acceptable qualifications for nonsupervisory test engineers will follow the guidance provided in RG 1.28 as discussed in VEGP COL FSAR Appendix 1AA, i.e., Appendix 2A-1 of American Society of Mechanical Engineers

(ASME) NQA-1-1994, "Quality Assurance Requirements for Nuclear Facility Applications."

The use of ASME NQA-1-1994 is endorsed in Section 17.5 of NUREG-0800 as providing an acceptable means for complying with 10 CFR Part 50, Appendix B, Criterion II, "Quality Assurance Program." Specifically, Item T of Part II of Section 17.5 of NUREG-0800 references ASME NQA-1-1994 in its guidance on training and qualification for personnel associated with inspection and testing activities. Therefore, the staff finds acceptable the proposed changes to VEGP COL FSAR Section 14.2.2.2, as stated in the applicant's November 11, 2010, letter. The planned VEGP COL application changes will be tracked as ***VEGP Confirmatory Item 14.2-1.***

<u>*Resolution of VEGP Standard Content Confirmatory Item 14.2-1*</u>

VEGP Confirmatory Item 14.2-1 is an applicant commitment to revise its FSAR to specify the qualifications for test engineers. The staff verified that VEGP COL FSAR Section 14.2.2.2 was appropriately updated. As a result, VEGP Confirmatory Item 14.2-1 is now closed. The applicant indicated that the proposed changes to its FSAR Section 14.2.2.2 is expected to be standard for the subsequent COL applicants. Since Confirmatory Item 14.2-1 already exists as a standard confirmatory item in this SER, the staff designated this standard confirmatory item as VEGP Confirmatory Item 14.2-1.

14.2.2.5 *Post Combined License Activities*

There are no post-COL activities related to this section.

14.2.2.6 *Conclusion*

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the application addressed the required information relating to the initial test program organization, staffing, and responsibilities and there is no outstanding information to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that, the information presented in the WLS COL FSAR is acceptable because it meets the requirements of 10 CFR 52.79(a)(28) and Criterion XI of Appendix B to 10 CFR Part 50. The staff based its conclusions on the following:

- STD COL 14.4-1 is acceptable because it provides an adequate description of the organizational responsibilities and authorities, the degree of participation of each organizational unit in the implementation of the initial test program, and personnel training, experience, and qualification requirements and meets the guidance in RG 1.68.

14.2.3 Test Specifications and Test Procedures (Related to RG 1.206, Section C.III.1, Chapter 14, C.I.14.2.3, “Test Procedures,” C.I.14.2.4, “Conduct of Test Program,” C.I.14.2.5, “Review, Evaluation, and Approval of Test Results,” and C.I.14.2.6, “Test Records”)

14.2.3.1 *Introduction*

Test specifications and test procedures address the process used to develop, review, and approve individual test procedures, including the organizational units or personnel that are involved in performing these activities and their respective responsibilities.

“Conduct of Test Program” describes the administrative controls that govern the conduct of each major phase of the test program. This description includes the administrative controls used to ensure that the necessary prerequisites are satisfied for each major phase and for individual tests. Controls to be followed during plant modifications or maintenance tasks that are determined to be necessary to conduct the test program are also described, as well as the methods used to ensure retesting following such modifications or maintenance.

“Review of Test Results” describes the specific controls to be established for the review, evaluation, and approval of test results by appropriate personnel and/or organizations. This description includes specific controls to be established to ensure notification of affected and responsible organizations or personnel when test acceptance criteria are not met, as well as the controls established to resolve such matters.

In addition, administrative controls to identify and cross-reference each test (or portion thereof) required to be completed before initial fuel loading to satisfy ITAAC in accordance with 10 CFR 52.99(a) are discussed.

14.2.3.2 *Summary of Application*

Section 14.2 of the WLS COL FSAR, Revision 11, incorporates by reference Section 14.2 of the AP1000 DCD, Revision 19. Section 14.2 of the DCD includes Section 14.2.3.

In addition, in WLS COL FSAR Sections 14.2 and 14.4, the applicant provided the following:

AP1000 COL Information Items

- STD COL 14.4-2

The applicant provided additional information in STD COL 14.4-2 to address COL holder responsibility for the development of test specifications and test procedures.

- STD COL 14.4-3

The applicant provided additional information in STD COL 14.4-3 to address COL holder responsibility for the development of a site-specific startup administrative manual (procedure) that will include the administrative procedures and requirements that will govern the activities associated with the plant's initial test program.

- STD COL 14.4-4

The applicant provided additional information in STD COL 14.4-4 to address COL holder responsibility for the review and evaluation of test results.

Supplemental Information

- STD SUP 14.2-5

The applicant provided additional information in STD Supplement (SUP) 14.2-5 to address administrative requirements for the preparation of work requests.

- STD SUP 14.2-6

The applicant provided additional information in STD SUP 14.2-6 to address administrative requirements for turnover of systems and components during the construction phase.

- STD SUP 14.2-7

The applicant provided additional information in STD SUP 14.2-7 to address administrative controls for the conduct of modifications during the initial test program.

- STD SUP 14.2-8

The applicant provided additional information in STD SUP 14.2-8 to address administrative controls for the conduct of maintenance during the initial test program.

In addition, in Part 10 of the WLS COL application, the applicant provided the following information:

License Conditions

- Part 10, License Condition 2, Items 14.4-2, 14.4-3 and 14.4-4

The proposed license conditions will require the licensee to complete the actions described in STD COL 14.4-2 and STD COL 14.4-4 prior to fuel loading and STD COL 14.4-3 prior to initiation of the test program.

- Part 10, License Condition 6

The proposed license condition will require the licensee to provide a schedule to support NRC inspections of operational programs including a submittal for approved preoperational and startup test procedures.

- Part 10, License Condition 8

The proposed license condition will require the licensee to report any changes to the initial test program (ITP) within one month of such a change.

14.2.3.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the test specifications and test procedures, conduct of test program, and review and evaluation of test results are given in Section 14.2 of NUREG-0800.

The applicable regulatory requirements for the information being reviewed in this section are 10 CFR 52.79(a)(28) and Criterion XI of Appendix B to 10 CFR Part 50. RG 1.68 provides guidance on how to comply with Criterion XI of Appendix B to 10 CFR Part 50.

14.2.3.4 *Technical Evaluation*

The NRC staff reviewed Section 14.2.3 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the test specifications and procedures, conduct of test program, and review and evaluation of test results. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is

identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the BLN Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from Section 14.2.3.4 of the VEGP SER:

> *<u>AP1000 COL Information Items</u>*
>
> - *STD COL 14.4-2, addressing test specifications and test procedures.*
>
> *The NRC staff reviewed STD COL 14.4-2 related to COL Information Item 14.4-2 included in the BLN COL FSAR. The applicant provided information to address COL Information Item 14.4-2 and to supplement the information addressed in the AP1000 DCD, Revision 17. COL Information Item 14.4-2 states:*
>
>> *The Combined License holder will provide the Preoperational and Startup Procedures to the NRC prior to each planned test in accordance with the requirements of DCD Subsection 14.2.3.*
>>
>> *The following words represent the original Combined License Information Item commitment:*
>>
>>> *The Combined License applicant is responsible for providing test specifications and test procedures for the preoperational and startup tests, as identified in Subsection 14.2.3, for review by the NRC.*
>
> *The commitment was also captured as COL Action Item 14.4-2 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:*
>
>> *The COL applicant will develop test specifications and procedures for the preoperational and startup tests for review by the NRC.*
>
> *The staff reviewed the applicant's proposed resolution of COL Information Item 14.4-2.*
>
> *In reviewing Section 14.2 of the BLN COL FSAR, Revision 0, the applicant did not provide a description of the methodology used to develop test specifications and procedures; did not provide a description of the controls to ensure the participation of the design organization(s), the COL applicant, architect-engineer(s), and other major contractors, subcontractors, and vendors, as applicable; and did not discuss the qualification or experience requirements for personnel participating in the development of test specifications and test procedures. In RAI 14.2-8, the staff requested that the applicant provide information regarding the methodology that will be used for the generation,*

review, and approval of preoperational and startup test procedures. Additionally, the staff requested that the applicant explain which provisions in the application ensure the availability of approved test procedures for review by NRC inspectors at least 60 days before their intended use, and ensure timely notification to the NRC of changes in approved test procedures that have been made available for NRC review.

In its response to RAI 14.2-8 dated June 26, 2008, the applicant stated that Section 14.2.3 of the AP1000 DCD provided administrative controls to ensure that approved test procedures will be provided to the NRC about 60 days prior to the scheduled performance of preoperational tests, such as test for systems and components that perform safety-related functions, and tests of systems and components that are non-safety-related but perform defense-in-depth functions. The staff found this response acceptable. However, the applicant did not provide a description of the administrative controls to be used to develop, review, and approve preoperational and startup test procedures. In RAI 14.2-12, dated December 8, 2008, the staff requested that the applicant provide such a description in the BLN COL FSAR.

In its response to RAI 14.2-12 dated January 22, 2009, the applicant proposed to include in Section 14.2.3 of the BLN COL FSAR the following administrative controls that will be prescribed in the site-specific startup administrative manual for the development, review, and approval of test specifications and test procedures:

- *Provisions to ensure that the appropriate technical information required for the preparation of test procedures is included, including prerequisites, format and content, objectives, test conditions, and acceptance criteria*

- *Provisions to ensure the participation of the design organization in the development of detailed test procedures*

- *Provisions to ensure that personnel developing and reviewing test procedures have the appropriate technical background and experience*

- *Provisions to ensure the availability of test procedures to the NRC onsite inspectors approximately 60 days prior to their intended use*

*The staff reviewed the applicant's response to this RAI and determined that the proposed changes provide the general methods and administrative provisions to control procedure development, review, and approval, including the responsibilities of the various organizations participating in this process, consistent with the guidance in RG 1.68. The applicant will revise the BLN COL FSAR to include the proposed administrative controls. Therefore, the staff finds the proposed change acceptable. This is identified as **Confirmatory Item 14.2-3**, pending NRC review and approval of the revised BLN COL FSAR.*

The applicant identified COL Information Item 14.4-2 as an activity that cannot be fully resolved prior to issuance of the COL. In BLN COL FSAR, Part 10, "License Conditions and ITAAC," License Condition 2, "COL Holder Items," the applicant proposed Item 14.4-2 to address the development of test specifications and test procedures. Additionally, the applicant proposed License Condition 6, "Operational Program Readiness," addressing implementation schedules to support planning for and conduct of NRC staff inspections of operational programs. Because the initial test program is identified as an operational program, the applicant provided implementation milestones consistent with the guidance contained in RG 1.206. To address the availability of test specifications and test procedures, Item d. of License Condition 6 requires a submittal schedule for preoperational and startup test procedures.

Since development of test specifications and test procedures will require detailed plant-specific design information and close coordination with design organizations, the staff determined that it is acceptable to develop detailed preoperational and startup test specifications and test procedures during the post-COL phase (See Section 14.2.3.5). Therefore, the staff finds acceptable proposed License Condition 2, Item 14.4-2. Concerns remain regarding the adequacy of administrative controls in License Condition 6, Item d., for the development of test specifications and test procedures. This is identified as ***Open Item 14.2-1.***

In RAI 14.2-11, the NRC staff requested that the applicant provide additional information regarding the provisions that will identify and cross-reference all or part of each test that is required to be completed before initial fuel loading and that is designed to satisfy ITAAC. The staff requested that the applicant revise Section 14.2 of the BLN COL FSAR to address this issue. In its September 3, 2008, response to RAI 14.2-11, the applicant stated that test procedures (or sections thereof) will be cross-referenced to ITAACs. In addition, activities related to ITAAC closure will include references to test procedures in order to facilitate NRC review and acceptance. The applicant stated that Chapter 14 of the BLN COL FSAR would be revised to include development of a cross-reference list between ITAACs and test procedures and/or sections of procedures. The staff confirmed that this change was incorporated in Revision 1 of the BLN COL FSAR. Section 14.4.2 of the BLN COL FSAR states that a cross-reference list will be developed between ITAACs and test procedures and/or sections of test procedures. The staff finds this change acceptable. This resolves RAI 14.2-11.

<u>Resolution of Standard Content Confirmatory Item 14.2-3</u>

The staff verified that the VEGP applicant has incorporated into its FSAR the proposed administrative controls identified as Confirmatory Item 14.2-3 in the staff's SER for the BLN COL. On this basis, Confirmatory Item 14.2-3 is resolved.

<u>Resolution of Standard Content Open Item 14.2-1</u>

Part 10 of the VEGP COL application, proposed License Condition 6, "Operational Program Readiness," describes the process for submitting to the appropriate Director of the NRC a schedule that will support planning for and conduct of NRC inspections of operational programs. The applicant also included, in Item c. of License Condition 6 (which corresponds to Item d. of License Condition 6 in the BLN COL application), administrative provisions for the submittal of approved preoperational and startup test procedures to NRC onsite inspectors in accordance with Section 14.2.3 of the FSAR. Following the evaluation of Item d. of License Condition 6 in the BLN COL application, as documented in the BLN SER, the staff has determined on closer examination that proposed License Condition 2, Item 14.4-2, will result in adequate administrative controls for the development of detailed test specifications and test procedures. On this basis, the staff finds that Item c. in proposed License Condition 6 of Part 10 of the VEGP COL application is acceptable and Open Item 14.2-1 is therefore resolved.

The following portion of this technical evaluation section is reproduced from Section 14.2.3.4 of the BLN SER:

- *STD COL 14.4-3, addressing the conduct of test program*

The NRC staff reviewed STD COL 14.4-3 related to COL Information Item 14.4-3 included in the BLN COL FSAR. The applicant provided additional information to address COL Information Item 14.4-3 and to supplement the information addressed in the AP1000 DCD, Revision 17. COL Information Item 14.4-3 states:

> *The Combined License holder is responsible for a site-specific startup administration manual (procedure), which contains the administration procedures and requirements that govern the activities associated with the plant initial test program, as identified in Subsection 14.2.3.*
>
> *The following words represent the original COL information item commitment:*
>
> > *The Combined License applicant is responsible for a startup administration manual (procedure), which contains the administration procedures and requirements that govern the activities associated with the plant initial test program, as identified in Subsection 14.2.3.*

This commitment was also captured as COL Action Item 14.4-3 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:

> *The COL applicant is responsible for preparing a startup administrative manual which contains the administrative procedures and standards that govern the activities associated with the plant initial test program.*

In Section 14.4 of the BLN COL FSAR, the applicant incorporated by reference Section 14.4.3 of the AP1000 DCD, Revision 17. This section provided a summary overview of the administrative process and program controls to be utilized in the conduct of the AP1000 Startup Test Program at a licensed AP1000 operational plant site. It also provided a general description of responsibilities and activities related to the testing of plant equipment in the period between system turnover until plant acceptance.

The staff reviewed the information provided to address COL Information Item 14.4-3 related to the conduct of the initial test program in the BLN COL FSAR. In its review, the staff identified areas where additional information was needed. A description of the specific issues follows.

In RAI 14.2-4, the staff requested that the applicant supplement the information incorporated by reference from Section 14.4.3 of the AP1000 DCD, Revision 17, and to provide a description of the administrative controls that will be implemented during the conduct of the initial test program, consistent with the guidance in RG 1.206 and Section 14.2 of NUREG-0800. In its response to RAI 14.2-4 dated June 26, 2008, the applicant stated that Section 14.4 of the BLN COL FSAR incorporated by reference Section 14.4.3 of the AP1000 DCD and no further changes to the BLN COL FSAR were needed. However, the staff determined that the information included in BLN COL FSAR was insufficient. Therefore, in RAI 14.2-12 dated December 8, 2008[SIC], the staff requested the applicant include a set of administrative controls for the conduct of the initial test program in Section 14.2 of the BLN COL FSAR.

In its response to RAI 14.2-12 dated January 22, 2009 and March 26, 2009, the applicant proposed to include in Section 14.2.3.1 of the BLN COL FSAR a description of the administrative controls for the control of testing activities. The proposed controls will include measures for procedure verification, work control, system turnover, conduct of modifications, and conduct of maintenance activities during the initial test program.

Section 14.2.3.1.1 would be revised to provide administrative controls for the verification of approved test procedures. The response stated that this section will include measures to consider design and licensing changes made after the development of test procedures to ensure that these changes are incorporated in approved test procedures. In addition, the applicant stated that available information regarding operating experience (OE) will be factored in the development of individual test procedures. Test deficiencies, nonconformances, exceptions, and failures will be tracked using the applicant's corrective action *program. The applicant also proposed controls to involve design organizations in the resolution of design-related problems that result in, or contribute to, a failure to meet test acceptance criteria. In its description, the applicant assigned responsibilities for the review of test procedures, test execution, data collection and recording, and for the review and evaluation of test results prior to commencing each major phase of the initial test program.*

The following supplemental items were not in Revision 1 of the BLN FSAR and are addressed for the first time in this SER for the VEGP COL application. However, portions of the standard evaluation material in the BLN SER under the evaluation of STD COL 14.4-3 are directly applicable to the new STD SUP items identified in the VEGP FSAR. Therefore, the NRC staff used this evaluation material, identified below as standard content material, in the disposition of these supplemental items.

<u>*Supplemental Information*</u>

- *STD SUP 14.2-5*

The applicant provided additional information in STD SUP 14.2-5 to address administrative requirements for the preparation of work requests.

The following portion of this technical evaluation section is reproduced from Section 14.2.3.4 of the BLN SER:

Section 14.2.3.1.2 would be revised to provide administrative measures for the control of work requests and controls for the control of tagging requests. Specifically, the response stated that the applicant will be responsible for the preparation of work requests and for supervising minor repairs and modifications, changes to equipment settings, and disconnecting and reconnecting of electrical terminations. Additionally, the Startup Group will provide for the coordination of construction-related work requests. The applicant also stated that the Startup Test Engineers may perform independent verification of work requests. These activities will be controlled by administrative procedures.

- *STD SUP 14.2-6*

The applicant provided additional information in STD SUP 14.2-6 to address administrative requirements for turnover of systems and components during the construction phase.

The following portion of this technical evaluation section is reproduced from Section 14.2.3.4 of the BLN SER.

Section 14.2.3.1.3 would be revised to provide controls for system turnover during the conduct of the test program. The response proposed guidelines that will be used to define the boundary and interfaces between related *systems/subsystems and to generate boundary scope documents. The response also proposed a systematic turnover process that includes requirements for the following:*

- *Documenting inspections performed by the construction organization (e.g., highlighted drawings showing areas inspected)*
- *Documenting results of construction testing*
- *Determining the construction related inspections and tests that need to be completed before preoperational testing begins. Any open items are evaluated for acceptability before commencing preoperational testing.*
- *Developing and implementing plans for correcting adverse conditions and open items, and means for tracking such conditions and items*
- *Verifying completeness of construction and documentation of incomplete items*

- *STD SUP 14.2-7*

The applicant provided additional information in STD SUP 14.2-7 to address administrative controls for the conduct of modifications during the initial test program.

The following portion of this technical evaluation section is reproduced from Section 14.2.3.4 of the BLN SER:

Section 14.2.3.1.4 would be revised to include controls for modifications during the conduct of the test program. The response also proposed measures for retesting activities following such modifications. In its description, the applicant stated that modifications will be documented in test procedures and will contain restoration steps to confirm satisfactory restoration to the required configuration. Additionally, modifications will be reviewed to determine the scope of post-modification testing activities. Finally, the response stated that retesting for modifications will be documented and verified to ensure the validity of preoperational testing and ITAAC.

- *STD SUP 14.2-8*

The applicant provided additional information in STD SUP 14.2-8 to address administrative controls for the conduct of maintenance during the initial test program.

The following portion of this technical evaluation section is reproduced from Section 14.2.3.4 of the BLN SER:

Section 14.2.3.1.5 would be revised to include controls for corrective or preventive maintenance during the conduct of the initial test program. The response proposed that the applicant will review maintenance activities to determine post-maintenance testing to be performed. Additionally, post-maintenance testing will be conducted and documented, and its results verified to maintain the validity of preoperational testing and ITAAC.

The following portion of this technical evaluation section is reproduced from Section 14.2.3.4 of the BLN SER, and is applicable to all four STD SUP items discussed above.

The staff reviewed the applicant's response to this RAI and determined that this change provides an adequate set of administrative measures to control the conduct of the initial test program, consistent with the guidance in RG 1.68, RG 1.206, and Section 14.2 of NUREG-0800. The applicant will revise the BLN COL FSAR to include the proposed administrative controls. Therefore, the staff finds this change acceptable. This is identified as ***Confirmatory Item 14.2-4****, pending NRC review and approval of the revised BLN COL FSAR.*

In addition to the administrative controls for the conduct of the initial test program, the applicant identified COL Information Item 14.4-3 as an activity that cannot be fully resolved prior to issuance of the COL. In BLN COL FSAR, Part 10, "License Conditions and ITAAC," License Condition 2, "COL Holder Items," the applicant proposed Item 14.4-3 to address the development of a site-specific startup administrative manual. This site-specific startup administrative manual will contain the administration procedures and requirements that govern the activities associated with the plant initial test program, as described in Section 14.2 of the BLN COL FSAR. The applicant stated that the startup administrative manual will be provided to the NRC prior to initiating the initial test program. Additionally, in Part 10 of the BLN COL FSAR, proposed License Condition 8, "Startup Testing," the applicant discussed the process for making changes to the initial test program described in Chapter 14 of the Bellefonte COL FSAR. The applicant stated that any changes to the initial startup test program made in accordance with the provisions of 10 CFR 50.59 or Section VIII of Appendix D to 10 CFR Part 52 shall be reported in accordance with 50.59(d) within one month of such change.

The staff determined that it is acceptable to develop a site-specific startup administrative manual, which will contain the administrative procedures and standards that govern the activities associated with the plant initial test program, during the post-COL phase (see Section 14.2.3.5). Therefore, the staff finds acceptable proposed License Condition 2, Item 14.4-3. Concerns remain regarding the adequacy of administrative controls for changing the test program as described in License Condition 8. This is identified as ***Open Item 14.2-2.***

Resolution of Standard Content Confirmatory Item 14.2-4

The staff verified that the VEGP applicant has incorporated into its FSAR, as STD SUP 14.2-5 through STD SUP 14.2-8, the proposed administrative controls identified as Confirmatory Item 14.2-4 in the staff's SER for the BLN COL. On this basis, Confirmatory Item 14.2-4 is resolved.

Resolution of Standard Content Open Item 14.2-2

Part 10 of the VEGP COL application, proposed License Condition 8, "Startup Testing," describes the process for initiating changes to the initial test program. The applicant proposed to notify the NRC of any change made to the startup test program described in Chapter 14 of the VEGP COL FSAR in accordance with provisions of 10 CFR 50.59(d) or Section VIII of Appendix D, "Design Certification Rule for the AP1000 Design," *to 10 CFR Part 52, "Licenses, certifications and approvals for nuclear power plants," within one month of such change. Following the evaluation of License Condition 8 in the BLN COL application, as documented in the BLN SER, the staff has determined, based on closer examination, that proposed License Condition 8 provides adequate administrative controls for notifying the NRC of changes to the test program, consistent with regulatory requirements in 10 CFR 50.59(d) and Section VIII of Appendix D to 10 CFR Part 52. On this basis, the staff determined that the applicant adequately addressed Open Item 14.2-2, and it is, therefore, resolved.*

The following portion of this technical evaluation section is reproduced from Section 14.2.3.4 of the BLN SER:

AP1000 COL Information Item

- *STD COL 14.4-4, addressing the review and evaluation of test results*

The NRC staff reviewed STD COL 14.4-4 related to COL Information Item 14.4-4 included under Section 14.2.3.2 of the BLN COL FSAR. The applicant provided additional information to address COL Information Item 14.4-4 as described in the AP1000 DCD, Revision 17. COL Information Item 14.4-4 states:

> *The combined license holder is responsible for review and evaluation of individual test results as well as final review of overall test results and for review of selected milestones or hold points within the test phases. Test exceptions or results which do not meet acceptance criteria are identified to the affected and responsible design organizations, and corrective actions and retests, as required, are performed.*

The commitment was also captured as COL Action Item 14.4-4 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:

> *The COL applicant or holder is responsible for review and evaluation of individual test results.*

In Section 14.2.3.2 of the BLN COL FSAR, the applicant provided specific administrative controls for the review and evaluation of test results. The applicant stated that the startup engineer is responsible for reviewing and evaluating the test data, test results, and verifying that the acceptance criteria have been met. The applicant also stated that test results will be reviewed and approved by the JTWG. The applicant included provisions to identify and notify the responsible design organizations when test exceptions or results do not meet acceptance criteria. The applicant also discussed the utilization of the corrective action program for tracking test results that do not meet the acceptance criteria, and for providing corrective action and retests, as required. Additionally, the applicant provided controls for the review of preoperational and startup test results, and for the retention of test reports.

While reviewing Section 14.2.3.2, the staff was unable to find provisions to ensure that retesting required for modification or maintenance remains in compliance with ITAAC. In RAI 14.2-10, the staff requested that the applicant provide additional information regarding the provisions to ensure that retesting remains in compliance with ITAAC. The staff requested that the applicant revise Section 14.2.3.2 of the BLN COL FSAR to include such provisions. In its September 3*, 2008, response to the staff's RAI, the applicant stated that normal maintenance, repairs, and design changes are controlled by the configuration control process in conjunction with the quality assurance and corrective action programs. These processes will provide for the review of changes that could have an impact on ITAAC. The staff confirmed that Section 14.2.3.2 of the BLN COL FSAR, Revision 1, was amended to include provisions to verify that the results of retesting do not invalidate ITAAC. The staff finds this change acceptable. This resolves RAI 14.2-10.*

In RAI 14.2-12, dated December 8, 2008, the staff requested that the applicant supplement Section 14.2.3.2 of the BLN COL FSAR by adding additional administrative controls to be implemented for the review, evaluation, and approval of test results, consistent with the guidance in RG 1.206. In its January 22, 2009, response to the staff's RAI, the applicant proposed controls and assigned responsibilities for the review of each major phase of the initial test program. Specifically, the applicant proposed to develop controls to assure that results of the preoperational and startup test phases will be reviewed and evaluated by qualified personnel from the PT&O and the JTWG organizations and approved by the plant manager. Also, the review of test results will include participation from design and construction organizations. Following each major phase of the initial test program, and before proceeding to the next stage of testing, the applicant will review test results to ensure that all required tests have been completed and that testing for the next major phase will be conducted in a safe manner. Additionally, the applicant proposed to develop controls to prepare *startup test results in accordance with RG 1.16, “Reporting of Operating Information – Appendix A Technical Specifications.”*

The staff reviewed the applicant’s response to RAI 14.2-12 and determined that the proposed changes provide administrative provisions to control the review, evaluation, and approval of test results, consistent with the guidance in RG 1.68, RG 1.206, and Section 14.2 of NUREG-0800. Therefore, the staff finds this change acceptable. The applicant will revise the BLN COL FSAR to include the proposed administrative controls. This is identified as ***Confirmatory Item 14.2-5****, pending NRC review and approval of the revised BLN COL FSAR.*

In addition to the administrative controls for the review, evaluation, and approval of test results, the applicant identified COL Information Item 14.4-4 as an activity that cannot be fully resolved prior to issuance of the COL. In BLN COL FSAR, Part 10, “License Conditions and ITAAC,” proposed License Condition 2, “COL Holder Items,” the applicant proposed Item 14.4-4 to address the review and evaluation of test results. The applicant stated that the COL holder will be responsible for the review and evaluation of test results, as well as the final review of overall test results and for the review of selected milestones or hold points within the test phases. In addition, the applicant stated that test exceptions or results which do not meet acceptance criteria will be identified to the affected and responsible design organizations, and corrective actions and retests, as required, will be performed.

Since test results will not be available until a facility is built, the staff determined that it is appropriate and acceptable for the COL holder to review and evaluate individual test results during the post-COL phase (see Section 14.2.3.5). The staff reviewed the proposed license condition and determined that the applicant provided sufficient administrative controls for the review and evaluation of test results, consistent with the guidance contained in RG 1.68, RG 1.206, and Section 14.2 of NUREG-0800.

Test Records

In its response to RAI 14.2-12, the applicant proposed to supplement the information incorporated by reference from Section 14.2.3.3 of the AP1000 DCD, Revision 17. The applicant stated that startup test reports will be generated and will describe and summarize the completion of tests during the initial test program. These proposed reports will address each test described in the BLN COL FSAR, describe measured values of operating conditions or characteristics from the initial test program as compared to design or specification values, and describe corrective actions and information required by license conditions. The applicant also described the frequency of such reports. Specifically, these proposed reports will be submitted 9 months following initial criticality, 90 days after completion of the test program, or 90 days after the start of commercial operations. The applicant also stated that in the event that one report does not cover these three events (i.e., initial criticality, completion of the *test program, and start of commercial operations), supplemental reports will be submitted every three months until all three events are completed.*

The staff reviewed the applicant's response to RAI 14.2-12 and determined that the proposed changes provide a set of administrative provisions to generate test reports, consistent with the guidance in RG 1.68, RG 1.206, and Section 14.2 of NUREG-0800. Therefore, the staff finds this change acceptable. The applicant will revise the BLN COL FSAR to include the proposed administrative controls. This is identified as ***Confirmatory Item 14.2-6****, pending NRC review and approval of the revised BLN COL FSAR.*

The staff determined that the supplemental information provided by the applicant described an acceptable method for activities related to test specifications and test procedures, conduct of the initial test program, and review, evaluation, and approval of test results, consistent with the guidance in RG 1.68 and RG 1.206. Therefore, the staff finds this change to be acceptable.

<u>*Resolution of Standard Content Confirmatory Items 14.2-5 and 14.2-6*</u>

The staff verified that the VEGP applicant has incorporated into its FSAR the proposed administrative controls identified as Confirmatory Items 14.2-5 and 14.2-6 in the staff's SER for the BLN COL. On this basis, Confirmatory Items 14.2-5 and 14.2-6 are resolved.

<u>*Evaluation of Revised License Condition 2, Items 14.4-3 and 14.4-4*</u>

In a letter dated October 15, 2010, the applicant proposed revisions to Items 14.4-3 and 14.4-4 of License Condition 2. Item 14.4-3 (evaluated above as part of the four SUP items) and Item 14.4-4 (evaluated above as part of STD COL 14.4-4) are considered unnecessary by the applicant as they can be adequately addressed by other proposed license conditions. The applicant proposed to replace the current text for Item 14.4-3 with, "Note - addressed by proposed License Conditions #3 and #6," and proposed to replace the current text for Item 14.4-4 with, "Note - addressed by proposed License Condition #9."

The text of Item 14.4-3 of License Condition 2 proposed to be deleted by the applicant's October 15, 2010, letter states that a site-specific startup administration manual (procedure), which includes the administration procedures and requirements that govern the activities associated with the plant's initial test program, would be provided prior to initiating the plant initial test program. Proposed License Condition 3 requires the operational program that addresses startup testing to be implemented prior to beginning the testing, and the proposed revision to Item c of License Condition 6 (evaluated above) would add the site-specific startup administrative manual to the items for which a schedule of availability would be provided to the NRC. The staff agrees that the combination of proposed License Condition 3 and proposed License Condition 6 *(as revised) will accomplish the goal of the text that is currently in Item 14.4-3 of License Condition 2.*

The text of Item 14.4-4 of License Condition 2 that is proposed to be deleted by the applicant's October 15, 2010, letter states that prior to initial fuel load, the licensee is responsible for review and evaluation of individual test results, as well as final review of overall test results and for review of selected milestones or hold points within the test phases. Test exceptions or results that do not meet acceptance criteria are identified to the affected and responsible design organizations, and corrective actions and retests are performed. The applicant stated that the proposed revision to License Condition 9 (which was initially proposed by the applicant in a letter dated June 18, 2010) also requires review and evaluation of individual test results, and that test exceptions or results that do not meet acceptance criteria are identified to the affected and responsible organizations, and corrective actions and retests, as required, are performed. The proposed revision would specifically add the review and evaluation of test results for those tests conducted during preoperational testing and for those conducted during power ascension (i.e., above low-power testing (defined as less than 5 percent rated thermal power [RTP])) up to and including testing at 100 percent RTP. This condition would then cover the entire startup testing program and would be retitled as "Startup Program Test Results." The staff agrees that the proposed revisions to License Condition 9 will accomplish the goal of the text that is currently in Item 14.4-4 of License Condition 2. Proposed License Condition 9 is evaluated by the staff in Section 14.2.8 of this SER.

14.2.3.5 *Post Combined License Activities*

The license condition language in this section has been clarified from previously considered language. In a letter dated March 22, 2016 (ADAMS Accession No. ML16084A099), the applicant did not identify any concerns with the clarified license condition language. The changes do not affect the staff's above analysis of the conditions, and therefore, for the reasons discussed in the technical evaluation section above, the staff finds the following license conditions acceptable:

- License Condition (14-1) – No later than 12 months after issuance of the COL, the licensee shall submit to the Director of NRO, or the Director's designee, a schedule for implementation of the approved preoperational and startup procedures (including the site-specific startup administration manual). The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until this license condition has been fully implemented. The schedule shall identify the completion of or implementation of the pre-operational and startup procedures (including the site-specific startup administration manual) identified in FSAR Section 14.2.3 (before initiating the initial test program).

- License Condition (14-2) – Within one month of change, any changes to the Initial Startup Test Program described in Chapter 14 of the WLS COL FSAR made in accordance with the provisions of 10 CFR 50.59 or Section VIII of Appendix D to 10 CFR Part 52 shall be reported in accordance with 10 CFR 50.59(d).

14.2.3.6 *Conclusion*

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to the test specifications and procedures, and there is no outstanding information to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the information presented in the WLS COL FSAR is acceptable because it meets the requirements of 10 CFR 52.79(a)(28) and Criterion XI of Appendix B to 10 CFR Part 50. The staff based its conclusions on the following:

- STD COL 14.4-2 is acceptable because it provides an adequate description of the administrative controls for the development, review, and approval of individual test specifications and test procedures that will be implemented during the conduct of the initial test program and meets the guidance in NUREG-0800, Section 14.2.

- STD COL 14.4-3 is acceptable because it provides an adequate description of the administrative controls for the development of a site-specific administrative manual (procedure) that will be implemented during the conduct of each major phase of the initial test program and meets the guidance in NUREG-0800, Section 14.2.

- STD COL 14.4-4 is acceptable because it provides an adequate description of the administrative controls for the review, evaluation, and approval of test results by qualified personnel, and the resolution of test exceptions or tests that do not meet the acceptance criteria during each major phase of the initial test program and meets the guidance in NUREG-0800, Section 14.2 and RG 1.68.

- STD SUP 14.2-5 is acceptable because it provides an adequate description of the administrative controls for work and tagging requests that will be implemented during the conduct of the initial test program and meets the guidance in NUREG-0800, Section 14.2.

- STD SUP 14.2-6 is acceptable because it provides an adequate description of the administrative controls for system turnover in an orderly and well-coordinated manner during the conduct of the initial test program and meets the guidance in NUREG-0800, Section 14.2.

- STD SUP 14.2-7 is acceptable because it provides an adequate description of the administrative controls for plant modifications and repairs identified as a result of plant testing and meets the guidance in NUREG-0800, Section 14.2.

- STD SUP 14.2-8 is acceptable because it provides an adequate description of the administrative controls for corrective or preventive maintenance that will be implemented during the conduct of the initial test program and meets the guidance in NUREG-0800, Section 14.2.

14.2.4 Compliance of Test Program with Regulatory Guides

Section 14.2 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 14.2.4, “Compliance of Test Program with Regulatory Guides,” of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff’s review confirmed that there is no outstanding issue related to this section. The results of the NRC staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

14.2.5 Utilization of Operating Experience (Related to RG 1.206, Section C.III.1, Chapter 14, C.I.14.2.8, “Utilization of Reactor Operating and Testing Experiences in Development of Test Program”)

14.2.5.1 *Introduction*

The design, testing, startup, and OE from previous pressurized water reactor plants is utilized in the development of the initial preoperational and startup test program for the AP1000 plant. It is also the responsibility of the COL applicant to utilize the reactor operating and testing experience in different aspects of the testing program.

14.2.5.2 *Summary of Application*

Section 14.2 of the WLS COL FSAR, Revision 11, incorporates by reference Section 14.2 of the AP1000 DCD, Revision 19. Section 14.2 of the DCD includes Section 14.2.5.

In addition, in WLS COL FSAR Section 14.2.5 and in Part 10 of the application, the applicant provided the following:

Supplemental Information

- STD SUP 14.2-4

The applicant provided supplemental information to describe the utilization of operating experience in the development of plant administrative procedures.

License Conditions

The applicant provided license conditions in Part 10 of its application that endorsed the standard content license conditions of the RCOL. The specific number of a proposed WLS license condition in some instances is different from the RCOL number. For these instances the corresponding WLS Part 10, license condition number is provided in square brackets. For example, RCOL Part 10, License Condition 7 equals WLS Part 10, License Condition 9 (noted below).

- Part 10, License Condition 2, Item 14.4-6

The proposed license condition addresses first-plant-only and three-plant-only tests.

- Part 10, License Condition 7 [WLS Part 10, License Condition 9]

The proposed license condition addresses first-plant-only and three-plant-only tests.

14.2.5.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the utilization of operating and testing experience are given in Section 14.2 of NUREG-0800.

The applicable regulatory requirements for the information being reviewed in this section are 10 CFR 52.79(a)(28) and Criterion XI of Appendix B to 10 CFR Part 50. RG 1.68 provides guidance on how to comply with Criterion XI of Appendix B to 10 CFR Part 50.

14.2.5.4 *Technical Evaluation*

The NRC staff reviewed Section 14.2.5 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the utilization of operating and testing experience. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the BLN Units 3 and 4 COL application. Any confirmatory items in the standard content material retain the numbers assigned in the VEGP SER.

The following portion of this technical evaluation section is reproduced from Section 14.2.5.4 of the VEGP SER:

> *Supplemental Information*
>
> - *STD SUP 14.2-4*
>
> *The applicant provided supplemental information to describe the utilization of operating experience in the development of plant administrative procedures.*
>
> *STD SUP 14.2-4 was not in Revision 1 of the BLN FSAR and is addressed for the first time in this SER for the VEGP COL application. However, portions of the standard evaluation material in Section 14.2.5.4 of the BLN SER are directly applicable to the new STD SUP item identified in the VEGP FSAR. Therefore, the NRC staff used this evaluation material, identified below as standard content material, in the disposition of STD SUP 14.2-4.*
>
> *Section 14.2.5 of the AP1000 DCD provided a summary overview of the administrative controls to be utilized for the development of preoperational and startup test programs for the AP1000 plant. As part of RAI 14.2-12, dated December 8, 2008, the NRC staff requested that the applicant supplement the BLN COL FSAR to describe how OE information will be used in developing and executing test procedures. In its January 22, 2009, response to the staff's RAI, the applicant proposed to revise the information in Section 14.2.5 of the BLN COL FSAR. The response stated that administrative procedures will be used for the control and evaluation of OE information. Specifically, the response proposed the use of OE during test procedure preparation, including the sources and types of information reviewed. Sources of OE reported and described include NRC reports, Institute of Nuclear Power Operations reports, and Significant Operating Event Reports. The response stated that Section 14.2.5 of the BLN COL FSAR would include a summary of the principal conclusions from a review of operating and testing experiences at other reactor facilities and their effect on the applicant's test program.*
>
> *The staff determined that the information proposed by the applicant describes an acceptable method for the consideration of reactor operating and testing*

*experience, and discussed the principal conclusions from a review of operating and testing experience and its inclusion into the initial test program description, consistent with the guidance in RG 1.68 and RG 1.206. Therefore, the staff finds this change acceptable. The applicant will revise the BLN COL FSAR to include the proposed administrative controls. This is identified as **Confirmatory Item 14.2-7**, pending NRC review and approval of the revised BLN COL FSAR.*

Resolution of Standard Content Confirmatory Item 14.2-7

The staff verified that the VEGP applicant has incorporated into its FSAR, in response to STD SUP 14.2-4, the proposed administrative controls identified as Confirmatory Item 14.2-7 in the staff's SER for the BLN COL. On this basis, Confirmatory Item 14.2-7 is resolved.

License Conditions

- *Part 10, License Condition 2, Item 14.4-6*

The following portion of this technical evaluation section is reproduced from Section 14.2.5.4 of the BLN SER:

In BLN COL FSAR, Part 10, "License Conditions and ITAAC," proposed License Condition 2, "COL Holder Items," the applicant proposed Item 14.4-6 to address first-plant-only and three-plant-only tests. The applicant stated that the COL holder for the first plant and the first three plants will perform the tests listed in Section 14.2.5 of the BLN COL FSAR. For subsequent plants, the COL applicant shall provide a justification that the results of the first-plant only tests or first-three-plant tests are applicable to the subsequent plant. In addition, COL holders referencing the results of the tests will provide the report prior to preoperational testing.

The staff reviewed the proposed license condition and determined that the applicant provided sufficient administrative controls for the performance of first-plant-only and three-plant-only tests, consistent with the guidance contained in RG 1.68, RG 1.206, and Section 14.2 of NUREG-0800. In addition, since test activities will not start until a facility is built, the staff determined that it is appropriate and acceptable for the COL holder to conduct these first-plant-only and three-plant-only tests during the post-COL phase (see Section 14.2.5.5).

Evaluation of Revised License Condition 2, Item 14.4-6

In a letter dated October 15, 2010, the VEGP applicant proposed a revision to License Condition 2, Item 14.4-6. Item 14.4-6 is considered unnecessary by the applicant as it can be adequately addressed by other proposed license conditions. The applicant proposed to replace the current text for Item 14.4-6 with, "Note - addressed by proposed License Conditions #7 and #9."

The text of Item 14.4-6 proposed to be deleted by the applicant's October 15, 2010, letter states the licensee(s) for the first plant and the first three plants will perform the tests listed in Section 14.2.5 of the VEGP COL FSAR. For subsequent plants, either tests listed in Section 14.2.5 shall be performed or the licensee shall provide a justification to the NRC, prior to fuel load, that the results of the first-plant-only tests or first-three-plant tests are applicable to the subsequent plant. The licensee(s) for the first AP1000 plant (or first-three-plants) will perform the tests defined during preoperational and startup testing as identified in Sections 14.2.9 and 14.2.10 of the VEGP COL FSAR.

The applicant stated that the October 15, 2010, proposed revisions to License Conditions 7 and 9 (both license conditions were initially proposed by the applicant in a letter dated June 18, 2010) adequately address the 3 parts of Item 14.4-6. Proposed License Condition 7 provides details on first-plant-only and three-plant-only tests and proposed License Condition 9 requires review and evaluation of individual test results, and that test exceptions or results that do not meet acceptance criteria are identified to the affected and responsible organizations, and corrective actions and retests, as required, are performed. The October 15, 2010, proposed revision to License Condition 9 would specifically add the review and evaluation of test results for those tests conducted during preoperational testing and for those conducted during power ascension (i.e., above low-power testing (<5 percent RTP) up to and including testing at 100 percent RTP). The October 15, 2010, proposed revision to License Condition 7 will address the written notifications for the pertinent testing.

The staff agrees that the proposed revisions to License Conditions 7 and 9 will accomplish the goal of the text that is currently in Item 14.4-6 of License Condition 2. Proposed License Condition 7 is evaluated by the staff later in this SER section. Proposed License Condition 9 is evaluated by the staff in Section 14.2.8 of this SER.

- *Part 10, License Condition 7 [WLS Part 10, License Condition 9]*

In its letter dated June 18, 2010, as revised by letter dated October 15, 2010, the applicant proposed License Condition 7, providing additional details on first-plant-only and three-plant-only tests. Certain design features of the AP1000 plant will be subjected to special tests to establish unique phenomenological performance parameters of the AP1000 design. Because of the standardization of the AP1000 design, these special tests (designated as first-plant-only tests and first-three-plant-only tests) are not required on subsequent plants. These tests will be controlled through license conditions to ensure that relevant test results are reviewed, evaluated, and approved by the designated licensee management before proceeding with the next testing phase. Accordingly, the applicant proposed the following license condition:

First-Plant-Only and First-Three-Plant-Only Testing

A licensee shall provide written identification of the applicable references for documentation for the completion of the testing to the Director of the Office of New Reactors (or equivalent NRC management) within thirty (30) calendar days of the licensee confirmation of acceptable test results.

Subsequent plant licensees crediting completion of testing by the first-plant or by the first-three plants shall provide a report referencing the applicable documentation identified by the first (or first three) plant(s) confirming the testing to the Director of the Office of New Reactors (or equivalent NRC management). This report shall be provided to NRC either prior to initiation of pre-operational testing, or within sixty (60) days of the identification of the documentation for the completion of the testing by the first plant (or third plant, as appropriate), whichever is later.

The NRC staff reviewed the proposed license condition and concludes that it contains some of the necessary attributes to achieve sufficient oversight by licensee management and assure adequate and timely notification to the NRC. However, the NRC staff plans to impose additional conditions in the areas addressed by proposed License Condition 7 to ensure that the relevant requirements in Section 14.2 of the AP1000 DCD are met.

14.2.5.5 *Post Combined License Activities*

The license condition language in this section has been clarified from previously considered language. In a letter dated March 22, 2016 (ADAMS Accession No. ML16084A099), the applicant did not identify any concerns with the clarified license condition language. The changes do not affect the staff's above analysis of the conditions, and therefore, for the reasons discussed in the technical evaluation section above, the staff finds the following license condition acceptable:

- License Condition (14-3) – The licensee shall perform the design-specific pre-operational tests identified below:
    1. In-Containment Refueling Water Storage Tank (IRWST) Heatup Test (first plant test as identified in AP1000 Design Control Document (DCD), Rev. 19, Section 14.2.9.1.3 Item (h));
    2. Pressurizer Surge Line Stratification Evaluation (first plant test as identified in AP1000 DCD, Rev. 19, Section 14.2.9.1.7 Item (d));
    3. Reactor Vessel Internals Vibration Testing (first plant test as identified in AP1000 DCD, Rev. 19, Section 14.2.9.1.9);
    4. Core Makeup Tank Heated Recirculation Tests (first three plants test as identified in AP1000 DCD, Rev. 19, Section 14.2.9.1.3 Items (k) and (w)); and
    5. Automatic Depressurization System Blowdown Test (first three plants test as identified in AP1000 DCD, Rev. 19, Section 14.2.9.1.3 Item (s)).

  The licensee shall notify the Director of NRO, or the Director's designee, in writing, upon successful completion of the design specific pre-operational tests.

14.2.5.6 *Conclusion*

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to the utilization of operating and testing experience, and there is no outstanding information to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the information presented in the WLS COL FSAR is acceptable, because it meets the requirements of 10 CFR 52.79(a)(28) and Criterion XI of Appendix B to 10 CFR Part 50. The staff based its conclusions on the following:

- STD SUP 14.2-4 is acceptable because it provides an adequate description of the administrative procedures that will be implemented to utilize operating experience in the development of plant administrative procedures during the conduct of the initial test program and meets the guidance in NUREG-0800, Section 14.2.

14.2.6 Use of Plant Operating and Emergency Procedures (Related to RG 1.206, Section C.III.1, Chapter 14, C.I.14.2.9, "Trial Use of Plant Operating and Emergency Procedures")

14.2.6.1 *Introduction*

To the extent practicable throughout the preoperational and initial startup test program, test procedures utilize operating, emergency, and abnormal procedures where applicable in the performance of tests. The use of these procedures is intended to do the following:

1. Provide the specific procedure or illustrate changes that may be required.
2. Provide training of plant personnel in the use of these procedures.

3. Increase the level of knowledge of plant personnel on the systems being tested.

A testing procedure utilizing an operating, emergency, or abnormal procedure references the procedure directly, or extracts a series of steps from the procedure in a way that is optimal to accomplishing the above goals while efficiently performing the specified testing.

14.2.6.2 *Summary of Application*

Section 14.2 of the WLS COL FSAR, Revision 11, incorporates by reference Section 14.2 of the AP1000 DCD, Revision 19. Section 14.2 of the DCD includes Section 14.2.6.

In addition, in WLS COL FSAR Section 14.2.6, the applicant provided the following:

AP1000 COL Information Item

- STD COL 14.4-3

The applicant provided additional information in STD COL 14.4-3 to address COL holder responsibility for the development of a site-specific startup administrative manual (procedure) that will include the administrative procedures and requirements that will govern the activities associated with the plant's initial test program.

14.2.6.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the use of plant operating and emergency procedures are given in Section 14.2 of NUREG-0800.

The applicable regulatory requirements for the information being reviewed in this section are 10 CFR 52.79(a)(28) and Criterion XI of Appendix B to 10 CFR Part 50. RG 1.68 provides guidance on how to comply with Criterion XI of Appendix B to 10 CFR Part 50.

14.2.6.4 *Technical Evaluation*

The NRC staff reviewed Section 14.2.6 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to plant operating and emergency procedures. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the BLN Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from Section 14.2.6.4 of the VEGP SER:

> *<u>AP1000 COL Information Item</u>*
>
> - *STD COL 14.4-3*
>
> *STD COL 14.4-3 was not explicitly evaluated in Section 14.2.6.4 of the BLN SER. However, the standard evaluation material in Section 14.2.6.4 of the BLN SER is directly applicable to this COL item. Therefore, the NRC staff used this evaluation material, identified below as standard content material, in the disposition of STD COL 14.4-3, as it relates to plant operating and emergency procedures.*
>
> *Section 14.2.6 of the AP1000 DCD stated that plant normal, abnormal, and emergency operating procedures will be used when performing preoperational and startup tests. As part of RAI 14.2-12, dated December 8, 2008, the staff requested that the applicant supplement the information incorporated by reference and describe how, and to what extent, the plant operating, emergency, and surveillance procedures will be trial-tested during the initial test program. In its January 22, 2009, response to the staff's RAI, the applicant proposed a method to develop, trial-test, and correct plant operating and emergency procedures during the initial test program. The response stated that preoperational and start up test procedures, normal, abnormal, and emergency procedures, and alarm response procedures, will be verified, validated, and implemented. The response proposed to describe administrative measures for*

the trial use of procedures in human machine interface testing as part of the control room design finalization. The response also proposed that controls would include the development of operating and emergency procedures to support human factors engineering, operational task analysis, training simulator development, and verification and validation of procedures and training material.

The response also proposed to include Section 14.2.6.1, "Operator Training and Participation during Certain Initial Tests," in the BLN COL FSAR. The response proposed administrative controls that will provide for the participation of plant operators and shift crews in plant changes, off-normal events, test program schedule, and selected startup tests. The response also proposed measures to ensure that unexpected plant or system responses will be reviewed, evaluated, and their results factored into the operator training program. The response stated that the operator training program will satisfy the criteria described in TMI Action Plan Item I.G.1 of NUREG-0737.

The staff determined that the information proposed by the applicant describe an acceptable method for the trial use of plant operating, emergency, and surveillance procedures, consistent with the guidance in RG 1.68 and RG 1.206. Therefore, the staff finds this change acceptable. The applicant will revise the BLN COL FSAR to include the proposed administrative controls. This is identified as ***Confirmatory Item 14.2-8****, pending NRC review and approval of the revised BLN COL FSAR.*

Resolution of Standard Content Confirmatory Item 14.2-8

The staff verified that the VEGP applicant has incorporated into its FSAR the proposed administrative controls identified as Confirmatory Item 14.2-8 in the staff's SER for the BLN COL. On this basis, Confirmatory Item 14.2-8 is resolved.

14.2.6.5 *Post Combined License Activities*

There are no post-COL activities related to this section.

14.2.6.6 *Conclusion*

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to the use of plant operating and emergency procedures, and there is no outstanding information to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the information presented in the WLS COL FSAR is acceptable because it meets the requirements of 10 CFR 52.79(a)(28) and Criterion XI of Appendix B to 10 CFR Part 50. The staff based its conclusions on the following:

- STD COL 14.4-3 is acceptable because it provides an adequate description of the administrative measures for the trial use of plant operating, emergency, and surveillance procedures that will be implemented during the conduct of the initial test program and meets the guidance in NUREG-0800, Section 14.2 and RG 1.68.

14.2.7 Initial Fuel Loading and Initial Criticality

Section 14.2 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 14.2.7, "Initial Fuel Loading and Initial Criticality," of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

14.2.8 Test Program Schedule (Related to RG 1.206, Section C.III.1, Chapter 14, C.I.14.2.11, "Test Program Schedule")

14.2.8.1 *Introduction*

This section describes administrative controls for the development of a schedule, relative to the fuel loading date, for conducting each major phase of the test program. Each test required to be completed before initial fuel loading is identified.

14.2.8.2 *Summary of Application*

Section 14.2 of the WLS COL FSAR, Revision 11, incorporates by reference Section 14.2 of the AP1000 DCD, Revision 19. Section 14.2 of the DCD includes Section 14.2.8.

In addition, in WLS COL FSAR, Section 14.2.8, the applicant provided the following:

Supplemental Information

- STD SUP 14.2-1

The applicant provided supplemental information to address the site-specific initial test program schedule.

In addition, in Part 10 of the WLS COL application, the applicant provided the following:

License Conditions

The applicant provided license conditions in Part 10 of its application that endorsed the standard content license conditions of the RCOL. The specific number of a proposed WLS license condition in some instances is different from the RCOL number. For these instances the corresponding WLS Part 10, license condition number is provided in square brackets. For example, RCOL Part 10, License Condition 9 is WLS Part 10, License Condition 10 (noted below).

- Part 10, License Condition 3

The proposed license condition addresses the initial test program implementation milestones.

- Part 10, License Condition 6

The proposed license condition addresses reporting requirements to the NRC regarding the initial test program.

- Part 10, License Condition 7

The proposed license condition addresses first-plant-only and first-three-plant-only testing.

- Part 10, License Condition 9 [WLS Part 10, License Condition 10]

The proposed license condition addresses the power-ascension test phase.

14.2.8.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the test program schedule are given in Section 14.2 of NUREG-0800.

The applicable regulatory requirements for the information being reviewed in this section are 10 CFR 52.79(a)(28) and Criterion XI of Appendix B to 10 CFR Part 50. RG 1.68 provides guidance on how to comply with Criterion XI of Appendix B to 10 CFR Part 50.

14.2.8.4 *Technical Evaluation*

The NRC staff reviewed Section 14.2.8 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the test program schedule. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP

Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the BLN Units 3 and 4 COL application. Any confirmatory items in the standard content material retain the numbers assigned in the VEGP SER.

The following portion of this technical evaluation section is reproduced from Section 14.2.8.4 of the VEGP SER:

Supplemental Information

- *STD SUP 14.2-1*

The applicant provided supplemental information to address the site-specific initial test program schedule.

The following portion of this technical evaluation section is reproduced from Section 14.2.8.4 of the BLN SER:

Test Program Schedule

As part of RAI 14.2-12, dated December 8, 2008, the staff requested that the applicant supplement the information incorporated by reference and describe the methodology that will be used to develop a schedule for conducting each major phase of the initial test program and for the development of test procedures. In its January 22, 2009, response to the staff's RAI, the applicant proposed to include information that further describes the administrative controls that will be used to develop a test program schedule. The applicant proposed controls for the development of a site-specific schedule that will address each major phase of the test program and will consider the organizational impact on overlapping test program schedules for multi-unit sites. The applicant also discussed the administrative measures in the startup administrative manual related to the test procedure development schedule and the initial test program schedule. The *applicant proposed specific controls for the development of detailed plant operating and emergency procedures, the availability of approved test procedures for review by NRC inspectors, and for the notification to the NRC of changes to approved test procedures. The response also stated that schedule milestones for the development of plant operating procedures are presented in Table 13.4-201 of the BLN COL FSAR. Finally, the response stated that operating and emergency procedures will be available for use both prior to the start of licensed operator training as well as during the initial test program implementation.*

The staff determined that the information proposed by the applicant described the methodology that will be used to develop a schedule, relative to the fuel loading date, for conducting each major phase of the test program, and for the development of test procedures, consistent with the guidance in RG 1.68 and RG 1.206. Therefore, the staff finds this change acceptable. The applicant will revise the BLN COL FSAR to include the proposed administrative controls. This is identified as ***Confirmatory Item 14.2-9****, pending NRC review and approval of the revised BLN COL FSAR.*

Operational Programs Required by the Regulations

In Section 13.4, Table 13.4-201, of the BLN COL FSAR, the applicant provided information to address the implementation of operational programs. The applicant identified the initial test program as an operational program and provided implementation milestones for each major phase of the test program. Additionally, the applicant stated that the initial test program will be implemented in three phases, namely the construction test program phase, the preoperational test program phase, and the startup test program phase. The construction test program phase will start prior to the first construction test being conducted. It will be followed by the preoperational test phase, which will start prior to the first preoperational test. Finally, the startup test phase is identified, and the applicant stated that it will start prior to initial fuel load. The staff reviewed the proposed milestones and determined that they adequately describe the implementation of each major phase of the initial test program and are, therefore, acceptable.

Resolution of Standard Content Confirmatory Item 14.2-9

The staff verified that the VEGP applicant has incorporated into its FSAR, in response to STD SUP 14.2-1, the proposed administrative controls identified as Confirmatory Item 14.2-9 in the staff's SER for the BLN COL. On this basis, Confirmatory Item 14.2-9 is resolved.

<u>*License Conditions*</u>

- *Part 10, License Conditions 3 and 6*

The following portion of this technical evaluation section is reproduced from Section 14.2.8.4 of the BLN SER:

In Part 10 of the BLN COL FSAR, License Condition 3, "Operational Program Implementation," the applicant proposed a license condition for the implementation of operational programs as described in Table 13.4-201 of the FSAR. This license condition included implementation milestones for the initial test program, namely E.1, F.1, and H.1. Specifically:

- *Milestone E.1 states that for construction testing, the licensee will implement the construction testing phase of the initial test program prior to the first construction test being conducted.*

- *Milestone F.1 states that for preoperational testing, the licensee will implement the preoperational testing phase of the initial test program prior to the first preoperational test being conducted.*

- *Milestone H.1 states that for startup testing, the licensee will implement the startup testing phase prior to initial fuel load.*

In Part 10 of the BLN COL FSAR, proposed License Condition 6, "Operational Program Readiness," the applicant states:

> *The licensee shall submit to the appropriate Director of the NRC, a schedule, no later than 12 months after issuance of the COL, that supports planning for and conduct of the NRC inspection of the operational programs listed in the operation program FSAR Table 13.4-201. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until either the operation programs in the FSAR table have been fully implemented or the plant has been placed in commercial service.*

The staff reviewed the BLN COL FSAR Table 13.4-201, and notes that the initial test program is listed as an operational program.

The staff determined that the proposed license conditions adequately describe the implementation of each major phase of the initial test program, consistent with the guidance contained in RG 1.68, RG 1.206, and Section 14.2 of NUREG-0800. In addition, since test activities will not start until a facility is built; the staff determined that it is appropriate and acceptable for the COL holder to submit a schedule, which will contain implementation details of operational programs, during the post-COL phase (see Section 14.2.8.5).

- *Part 10, License Condition 9 [WLS Part 10, License Condition 10]*

Certain milestones within the startup testing phase of the initial test program (i.e., pre-critical testing, criticality testing, and low-power testing) will need to be controlled through license conditions to ensure that relevant test results are reviewed, evaluated, and approved by the designated licensee management before proceeding with the power ascension test phase.

In its second letter dated June 18, 2010[2], as revised by letter dated October 15, 2010, the applicant proposed License Condition 9, providing additional detail on the power-ascension test phase. Specifically, the applicant proposed the following license condition:

> *<u>Pre-operational Testing</u>*
>
> *Following completion of pre-operational testing, the licensee shall review and evaluate individual test results. Test exceptions or results which do not meet acceptance criteria are identified to the affected and responsible organizations, and corrective actions and retests, as required, are performed.*
>
> *<u>Pre-critical and Criticality Testing</u>*
>
> 1. *Following completion of pre-critical and criticality testing, the licensee shall review and evaluate individual test results. Test exceptions or results which do not meet acceptance criteria are identified to the affected and responsible organizations, and corrective actions and retests, as required, are performed.*
>
> 2. *The licensee shall provide written notification to the Director of the Office of New Reactors (or equivalent NRC management) within fourteen (14) calendar days of completion of the pre-critical and criticality testing.*
>
> *<u>Low-Power (<5% RTP) Testing</u>*
>
> 1. *Following completion of low-power (<5% RTP) testing, the licensee shall review and evaluate individual test results. Test exceptions or results which do not meet acceptance criteria are identified to the affected and responsible organizations, and corrective actions and retests, as required, are performed.*

[2] The first letter dated June 18, 2010, provided proposed License Condition 7, which is evaluated in Section 14.2.5 of this SER.

2. *The licensee shall provide written notification to the Director of the Office of New Reactors within fourteen (14) calendar days of completion of the low power testing.*

At-Power (5%-100% RTP) Testing

1. *Following completion of at-power testing (at or above 5% RTP up to and including testing at 100% RTP), the licensee shall review and evaluate individual test results. Test exceptions or results which do not meet acceptance criteria are identified to the affected and responsible organizations, and corrective actions and retests, as required, are performed.*

2. *The licensee shall provide written notification to the Director of the Office of New Reactors (or equivalent NRC management) within fourteen (14) calendar days of completion of the at-power testing.*

The NRC staff reviewed the proposed license condition and concludes that it contains some of the necessary attributes to achieve sufficient oversight by licensee management and assure adequate and timely notification to the NRC. However, the NRC staff plans to impose additional conditions in the areas addressed by proposed License Condition 9 to ensure that the relevant guidance of RG 1.68 and the relevant requirements of Criterion XI of Appendix B to 10 CFR Part 50 are met.

14.2.8.5 *Post Combined License Activities*

The license condition language in this section has been clarified from previously considered language. In a letter dated March 22, 2016 (ADAMS Accession No. ML16084A099), the applicant did not identify any concerns with the clarified license condition language. The changes do not affect the staff's above analysis of the conditions, and therefore, for the reasons discussed in the technical evaluation section above, the staff finds the following license conditions acceptable:

- License Condition (14-4) – The licensee shall implement the initial test program or applicable portions thereof as described in the milestones below:
  1. Construction Test Program implemented before the first construction test;
  2. Preoperational Test Program implemented before the first preoperational test; and
  3. Startup Test Program implemented before initial fuel load.

- License Condition (14-5) – No later than 12 months after issuance of the COL, the licensee shall submit to the Director of NRO, or the director's designee, a schedule that supports planning for and conduct of NRC inspections of the Initial Test Program (ITP). The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until the ITP has been fully implemented.

- License Condition (14-6) –

  Pre-operational Testing

  1. The licensee shall review and evaluate the results of the tests identified in License Condition (14-3) and confirm that these test results are within the range of acceptable values predicted or otherwise confirm that the tested systems perform their specified functions in accordance with AP1000 DCD Rev. 19, Section 14.2.9.
  2. The licensee shall notify the Director of NRO, or the Director's designee, in writing, upon successful completion of the design specific pre-operational tests identified in License Condition (14-3); and
  3. The licensee shall notify the Director of NRO, or the Director's designee, in writing, upon the successful completion of all the ITAAC.

  Nuclear Fuel Loading and Pre-critical Testing

  1. Until the submission of the notification required by "Pre-operational Testing," item 2, above, the licensee shall not load fuel into the reactor vessel;
  2. Upon submission of the notification required by "Pre-operational Testing," item 2, above, and upon a Commission finding in accordance with 10 CFR 52.103(g) that all the acceptance criteria in the ITAAC are met, the licensee is authorized to perform pre-critical tests in accordance with the conditions specified herein;
  3. The licensee shall perform the pre-critical tests identified in AP1000 DCD Rev. 19, Section 14.2.10.1;
  4. The licensee shall review and evaluate the results of the tests identified in "Nuclear Fuel Loading and Pre-critical Testing," item 3, above, and confirm that these test results are within the range of acceptable values predicted or otherwise confirm that the tested systems perform their specified functions in accordance with AP1000 DCD Rev. 19, Section 14.2.10; and
  5. The licensee shall notify the Director of NRO, or the Director's designee, in writing, upon successful completion of the pre-critical tests identified in "Nuclear Fuel Loading and Pre-critical Testing," item 3, above.

  Initial Criticality and Low-Power Testing

  1. Upon submission of the notification required by "Nuclear Fuel Loading and Pre-critical Testing," item 5, above, the licensee is authorized to operate the facility at reactor steady-state core power levels not to exceed 5-percent thermal power in accordance with the conditions specified herein;

2. The licensee shall perform the initial criticality and low-power tests identified in AP1000 DCD Rev. 19, Sections 14.2.10.2 and 14.2.10.3, respectively, the Natural Circulation (first plant test) identified in AP1000 DCD Rev. 19, Section 14.2.10.3.6, and the Passive Residual Heat Removal Heat Exchanger (first plant test) identified in AP1000 DCD Rev. 19, Section 14.2.10.3.7;
3. The licensee shall review and evaluate the results of the tests identified in "Initial Criticality and Low-Power Testing," item 2, above, and confirm that these test results are within the range of acceptable values predicted or otherwise confirm that the tested systems perform their specified functions in accordance with AP1000 DCD Rev. 19, Section 14.2.10.2 and 14.2.10.3; and
4. The licensee shall notify the Director of NRO, or the Director's designee, in writing, upon successful completion of initial criticality and low-power tests identified in "Initial Criticality and Low-Power Testing," item 2, above, including the design-specific tests identified therein.

Power Ascension Testing

1. Upon submission of the notification required by "Initial Criticality and Low-Power Testing," item 4, above, the licensee is authorized to operate the facility at reactor steady-state core power levels not to exceed 100-percent thermal power in accordance with the conditions specified herein, but only for the purpose of performing power ascension testing;
2. The licensee shall perform the power ascension tests identified in the AP1000 DCD Rev. 19, Section 14.2.10.4, the Rod Cluster Control Assembly Out of Bank Measurements (first plant test) identified in AP1000 DCD, Rev. 19, Section 14.2.10.4.6, and the Load Follow Demonstration (first plant test) identified in AP1000 DCD, Rev. 19, Section 14.2.10.4.22;
3. The licensee shall review and evaluate the results of the tests identified in "Power Ascension Testing," item 2, above, and confirm that these test results are within the range of acceptable values predicted or otherwise confirm that the tested systems perform their specified functions in accordance with AP1000 DCD Rev.19, Section 14.2.10.4; and
4. The licensee shall notify the Director of NRO, or the Director's designee, in writing, upon successful completion of power ascension tests identified in "Power Ascension Testing," item 2, above, including the design-specific tests identified therein.

Maximum Power Level

Upon submission of the notification required by "Power Ascension Testing," item 4, above, the licensee is authorized to operate the facility at steady state reactor core power levels not to exceed 3400 MW thermal (100-percent thermal power), as described in the FSAR, in accordance with the conditions specified herein.

14.2.8.6 *Conclusion*

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to the test program schedule, and there is no outstanding information to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the information presented in the WLS COL FSAR is acceptable, because it meets the requirements of 10 CFR 52.79(a)(28) and Criterion XI of Appendix B to 10 CFR Part 50. The staff based its conclusions on the following:

- STD SUP 14.2-1 is acceptable because it provides an adequate description of the administrative measures for the development of a site-specific initial test program schedule and meets the guidance in NUREG-0800, Section 14.2 and RG 1.68.

14.2.9 Preoperational Test Descriptions (Related to RG 1.206, Section C.III.1, Chapter 14, C.I.14.2.12, "Individual Test Descriptions")

14.2.9.1 *Introduction*

This section includes test abstracts for each individual test conducted during the initial test program. The abstracts: (1) identify each test by title; (2) specify the prerequisites and major plant operating conditions necessary for each test (such as power level and mode of operation of major control systems); (3) provide a summary description of the test objectives and method, significant parameters, and plant performance characteristics to be monitored; and (4) provide a summary of the acceptance criteria established for each test to ensure that the test verifies the functional adequacy of the SSCs involved in the test. The abstracts also include sufficient information to justify the specified test method if such method does not subject the SSC under test to representative design operating conditions. In addition, the abstracts identify pertinent precautions for individual tests, as necessary (e.g., minimum flow requirements or reactor power level that must be maintained).

14.2.9.2 *Summary of Application*

Section 14.2 of the WLS COL FSAR, Revision 11, incorporates by reference Section 14.2 of the AP1000 DCD, Revision 19. Section 14.2 of the DCD includes Section 14.2.9.

In addition, in WLS COL FSAR, the applicant provided the following:

Departures

- WLS DEP 6.4-2

The applicant provided additional information in Section 14.2.9 of the WLS COL FSAR about WLS DEP 6.4-2 related to design changes affecting how the temperature and humidity in the main control room are maintained within the limits for reliable human performance. This information, as well as related WLS DEP 6.4-2 information appearing in other chapters of the FSAR, is reviewed in Section 21.3 of this SER.

AP1000 COL Information Items

- STD COL 14.4-5

The applicant provided additional information in STD COL 14.4-5 to address interface requirements.

- STD COL 3.9-5

The applicant provided additional information in STD COL 3.9-5 to address initial testing of the pressurizer surge line piping.

Supplemental Information

- STD SUP 14.2-2

The applicant provided additional information in STD SUP 14.2-2 to address the development of administrative procedures that will be implemented during the preoperational testing activities.

14.2.9.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the preoperational test descriptions are given in Section 14.2 of NUREG-0800.

The applicable regulatory requirements for the information being reviewed in this section are 10 CFR 52.79(a)(28) and Criterion XI of Appendix B to 10 CFR Part 50. RG 1.68 provides guidance on how to comply with Criterion XI of Appendix B to 10 CFR Part 50.

14.2.9.4 *Technical Evaluation*

The NRC staff reviewed Section 14.2.9 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the preoperational test descriptions. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the BLN Units 3 and 4 COL application. Any confirmatory items in the standard content material retain the numbers assigned in the VEGP SER.

The following portion of this technical evaluation section is reproduced from Section 14.2.9.4 of the VEGP SER:

> *<u>AP1000 COL Information Items</u>*
>
> - *STD COL 14.4-5*
>
> *The NRC staff reviewed STD COL 14.4-5 related to COL Information Item 14.4-5, which addresses interface requirements. The applicant provided additional information in Sections 14.2.9 and 14.2.10 of the VEGP COL FSAR to address COL Information Item 14.4-5. COL Information Item 14.4-5 states:*
>
> > *The Combined License applicant is responsible for testing that may be required of structures and systems which are outside the scope of this design certification. Test Specifications and acceptance criteria are provided by the responsible design organizations as identified in subsection 14.2.3 [of the AP1000 DCD]. The interfacing systems to be considered for testing are taken from Table 1.8-1 [of the AP1000 DCD] and include as a minimum, the following:*
> >
> > - *Storm drains*
> > - *Site specific seismic sensors*
> > - *Offsite [alternating current] ac power systems*
> > - *Circulating water heat sink*

- *Raw and sanitary water systems*
- *Individual equipment associated with the fire brigade*
- *Portable personnel monitors and radiation survey instruments*
- *Equipment associated with the physical security plan*

The commitment was also captured as COL Action Item 14.4-5 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:

> *The COL applicant is responsible for testing that may be required of structures and systems that are outside the scope of the design certification.*

The following portion of this technical evaluation section is reproduced from Section 14.2.9.4 of the BLN SER. Some of the text in the BLN SER associated with STD COL 14.4-5 has been relocated to the evaluation of STD SUP 14.2-2, as discussed below.

In its review of the information provided by the applicant to address COL Information Item 14.4-5, the staff noted that the seismic monitoring system testing described in Section 14.2.9.4.15 of the AP1000 DCD also applies to the site-specific seismic sensors.

The applicant also provided information regarding the following systems:

- *storm drains (Section 14.2.9.4.22)*
- *offsite ac power systems (Section 14.2.9.4.23)*
- *raw water systems (Section 14.2.9.4.24)*
- *sanitary drainage system (Section 14.2.9.4.25)*
- *fire brigade support equipment (Section 14.2.9.4.26)*
- *portable personnel monitors and radiation survey instruments (Section 14.2.9.4.27)*
- *cooling tower(s) (Section 14.2.10.4.29)*

The staff notes that information provided relative to equipment associated with the Physical Security Plan will be reviewed in Chapter 13 of this SER.

As part of RAI 14.2-1, the staff requested that the applicant provide additional information in the test abstract related to the offsite ac power systems. Specifically, Section 14.2.9.4.23 of the BLN COL FSAR states that the offsite ac power system components undergo a series of individual component and integrated system preoperational tests to verify that the offsite ac power system performs in accordance with the associated component design specifications. The individual component and integrated tests include:

a. *Availability of ac and direct current (dc) power to the switchyard equipment is verified.*

b. Operation of high voltage (HV) circuit breakers is verified.

c. Operation of HV disconnect switches and ground switches is verified.

d. Operation of substation transformers is verified.

e. Operation of current transformers, voltage transformers, and protective relays is verified.

f. Operation of switchyard equipment controls, metering, interlocks, and alarms that affect plant offsite ac power system performance is verified.

g. Design limits of switchyard voltages and stability are verified.

h. Under simulated fault conditions, proper function of alarms and protective relaying circuits is verified.

The staff asked in its RAI that the above list should include the following items:

- *Operation of instrumentation and control alarms used to monitor switchyard equipment status*
- *Proper operation and load carrying capability of breakers, switchgear, transformers, and cables*
- *Proper operation of the automatic transfer capability of the preferred power supply to the maintenance power supply through the reserve auxiliary transformer*
- *Operation of main generator in islanding mode is verified to ensure that the onsite power system equipment including the Class 1E battery chargers and uninterruptible power supplies can withstand the voltage spike from the generator following isolation from the grid.*
- *Switchyard interface agreement and protocols are verified.*

The staff requested that the applicant revise Section 14.2.9.4.23 to include the above items, or justify their exclusion.

In its June 26, 2008, response to RAI 14.2-1, the applicant agreed to add the above tests to BLN COL FSAR Section 14.2.9.4.23, except for verifying the proper operation of the generator in islanding mode. The applicant stated that this islanding mode test does not belong to this BLN COL FSAR section. This test is specified by Westinghouse as a load rejection test from 100 percent power in AP1000 DCD Section 14.2.10.4.21. That section will verify proper operation of equipment utilized in the generator islanding mode by a combination of the *purchase specifications for the equipment and verification of satisfactory performance after the load reject test from 100 percent power. The applicant proposed to revise BLN COL FSAR Chapter 14, Section 14.2.9.4.23 by adding the following to the end of the existing Section 14.2.9.4.23 in the sequence indicated:*

i. Operation of instrumentation and control alarms used to monitor switchyard equipment status.

j. Proper operation and load carrying capability of breakers, switchgear, transformers, and cables, and verification of these items by a non-testing means such as a [quality control] QC nameplate check of as-built equipment where testing would not be practical or feasible.

k. Verification of proper operation of the automatic transfer capability of the preferred power supply to the maintenance power supply through the reserve auxiliary transformer.

l. Switchyard interface agreement and protocols are verified.

With the addition of above offsite ac power system tests to the existing Section 14.2.9.4.23, the staff finds that the offsite ac power system testing performed under BLN COL FSAR Chapter 14, Section 14.2.9.4.23 will demonstrate the energization and proper operation of the as-installed switchyard components. In addition, the staff concurs with the applicant that verification of proper operation of the generator in islanding mode is part of AP1000 DCD Section 14.2.10.4.21, "100 Percent Load Rejection." Therefore, the staff finds the applicant's response acceptable. This is ***Confirmatory Item 14.2-11****, pending NRC review and approval of the revised BLN COL FSAR.*

As part of RAI 14.2-2, the staff also requested that the applicant provide additional information to the test abstract related to the offsite ac power systems. The staff stated that the AP1000 DCD provides interface requirements for the transmission switchyard and onsite power system in accordance with 10 CFR 52.79(b). Specifically, Summary Table 1.8-1, "Plant Interfaces with the Remainder of Plant," requires the COL applicant to address offsite ac requirements (Item 8.2) for steady-state load, inrush kVA for motors, nominal voltage, allowable voltage regulation, nominal allowable frequency fluctuation, maximum frequency decay rate, and limiting under-frequency value for the reactor coolant pump (RCP). It further requires the offsite transmission system analysis (Item 8.3) for loss of the AP1000 unit or the largest unit, for voltage operating range, for maintaining transient stability, and for the RCP bus voltage to remain above the voltage required to maintain the flow assumed in Chapter 15 analyses for a minimum of three seconds following a turbine trip. The staff requested that the applicant discuss how the preoperational test performed under Section 14.2.9.4.23 (General Test Methods and Acceptance Criteria) for BLN verifies all requirements cited in Sections 8.2 and 8.3 of the AP1000 DCD.

In its June 26, 2008, response to RAI 14.2-2, the applicant stated that site interface requirements in AP1000 DCD Table 1.8-1, Items 8.2 (offsite ac requirements) and 8.3 (offsite transmission system and stability analyses) are verified not just by BLN COL FSAR Section 14.2.9.4.23 (preoperational test for offsite ac power systems) alone, but a combination of analyses and testing as described below:

- *The site interface parameters identified in AP1000 DCD Table 1.8-1, Items 8.2 and 8.3, as provided by Westinghouse, are used as input parameters or acceptance criteria in the Grid Stability Analysis performed.*

- *The Offsite AC Power Systems tests detailed in BLN COL FSAR Section 14.2.9.4.23, as modified by the applicant's response to RAI 14.2-1, require specific preoperational testing of as-installed switchyard components as described in BLN COL FSAR Section 8.2 to demonstrate proper operation of the design capabilities and protective features of those components.*

- *The tests detailed in AP1000 DCD Section 14.2.9.4.21, Main, Unit Auxiliary and Reserve Auxiliary Transformer Test, demonstrate the energization of the transformers and the proper operation of associated protective relaying, alarms, and control devices.*

- *The tests detailed in AP1000 DCD Section 14.2.9.2.15, Main AC Power System Testing, verify power availability to support proper operation of required electrical loads.*

- *The 100 percent load reject test described in AP1000 DCD Section 14.2.10.4.21 provides for an integrated plant response and verification of the demands placed on the electrical distribution system when the plant is separated from the grid.*

The staff has reviewed BLN COL FSAR Section 14.2.9.4.23 and AP1000 DCD Sections 14.2.9.4.21, 14.2.9.2.15, and 14.2.10.4.21 cited by the applicant for proper operation of components and the interface parameters required for the grid stability and offsite transmission system analyses. The staff concurs with the applicant that the site interface requirements in AP1000 DCD Table 1.8-1, Items 8.2 and 8.3 can be verified by the combination of analyses and testing described above. Therefore, the NRC staff finds the applicant's response to be acceptable. This resolves RAI 14.2-2.

In RAI 14.2-9, the staff requested that the applicant provide additional information in the test abstract related to the fire brigade support equipment test abstract in Section 14.2.9.4.26 of the BLN COL FSAR. Specifically, RG 1.189, Regulatory Position 3.4.2, Hydrants and Hose Houses, states that "threads compatible with those used by local fire departments should be provided on all hydrants, hose *couplings, and standpipe risers. Alternatively, a sufficient number of hose thread adapters may be provided." The importance of ensuring that installed plant fire equipment be compatible with the equipment used by local fire departments warrants the inclusion of installed plant fire equipment (hydrants, hoses, couplings, and standpipe risers) in the initial test program to verify either the compatibility of threads or the provision of an adequate supply of hose thread adaptors that will be readily available in the event of a fire. The staff requested that the applicant revise Section 14.2.9.4.26 to address this issue. In addition, with respect to BLN COL FSAR Section 14.2.9.4.26(c), the staff requested that the applicant specifically identify any portable "communication equipment" that is credited for fire brigade use. In a letter dated June 30, 2008, the applicant proposed to add the requirement to verify fire equipment hose thread compatibility in Section 14.2 in a future revision of the BLN COL FSAR. The staff confirmed that the applicant addressed the relevant information in Revision 1 of the BLN COL FSAR, and there is no outstanding information expected to be addressed related to this section. This resolves RAI 14.2-9.*

In RAI 12.3-12.4-5, the staff requested that the applicant provide additional information related to the portable personnel monitors and radiation survey instruments test abstract contained in Section 14.2.9.4.27 of the BLN COL FSAR. Specifically, the staff requested the applicant to provide information regarding the accuracy and overall performance of portable survey instruments addressed in standard ANSI N42.17A-1989, and information related to the calibration and maintenance of portable radiation survey instruments addressed in ANSI N323A-1997. The staff also requested that the applicant revise Section 14.2 of the BLN COL FSAR to address this issue. In a letter dated September 22, 2008, the applicant proposed to revise Section 14.2.9.4.27 by providing additional text to the general method and acceptance criteria. Specifically, the applicant proposed that the portable monitors and instrument test shall include provisions for verifying proper functioning of monitors and instruments to respond to radiation as required and proper operatability [sic] *of instrumentation controls, battery, and alarms as applicable. Further, the applicant proposed to revise Appendix 1AA to Chapter 1, to include the updated version of ANSI N323A cited in the exception to Regulatory Guide 8.6. The staff reviewed the applicant's response and found the proposed changes acceptable. Further, the staff confirmed that the applicant addressed the relevant information in Revision 1 of the BLN COL FSAR, and there is no outstanding information expected to be addressed related to this section. This resolves RAI 12.3-12.4-5.*

Resolution of Standard Content Confirmatory Item 14.2-11

The staff verified that the VEGP applicant has incorporated into its FSAR the proposed administrative controls identified as Confirmatory Item 14.2-11 in the staff's SER for the BLN COL. On this basis, Confirmatory Item 14.2-11 is resolved.

- *STD COL 3.9-5*

In a letter dated July 2, 2010 and supplemented by letter dated August 6, 2010, the VEGP applicant identified changes to be made to VEGP COL FSAR Section 14.2.9 involving the initial testing of the pressurizer surge line piping. This COL item is primarily addressed in Section 3.9.3 of the VEGP COL FSAR and that portion is reviewed by the NRC staff in Section 3.12 of this SER. The portion of STD COL 3.9-5 addressed in FSAR Section 14.2 and evaluated in this SER section, is the discussion of the test abstract to identify the standard operating conditions for surge line thermal monitoring instrumentation verification and data gathering that complies with NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification." The staff notes that this proposed testing is to be done on the first AP1000 unit placed in operation.

*The NRC staff has compared the purpose, prerequisites, and general test methods and acceptance criteria provided by the VEGP applicant in the test abstract for the pressurizer surge line piping, to the guidance in NRC Bulletin 88-11. The staff concludes that sufficient information on the test procedure has been provided to assure that the test results will quantify the extent of thermal stratification, thermal stripping and piping deflections, as recommended in Bulletin 88-11. Therefore, the staff finds that the portion of STD COL 3.9-5 relevant to the preoperational testing of the pressurizer surge line piping to be acceptable. The incorporation of the planned changes to the VEGP COL FSAR will be tracked as **VEGP Confirmatory Item 14.2-2**.*

<u>Resolution of VEGP Standard Content Confirmatory Item 14.2-2</u>

VEGP Confirmatory Item 14.2-2 is an applicant commitment to revise its FSAR to specify surge line monitoring test procedures. The staff verified that the VEGP COL FSAR was appropriately updated. As a result, VEGP Confirmatory Item 14.2-2 is now closed. The applicant indicated that the proposed changes to its FSAR are expected to be standard for the subsequent COL applicants. Since Confirmatory Item 14.2-2 already exists as a standard confirmatory item in this SER, the staff designated this standard confirmatory item as VEGP Confirmatory Item 14.2-2.

<u>Supplemental Information</u>

- *STD SUP 14.2-2*

The applicant provided additional information in STD SUP 14.2-2 to address the development of administrative procedures that will be implemented during the preoperational testing activities.

STD SUP 14.2-2 was not in Revision 1 of the BLN FSAR and is addressed for the first time in this SER for the VEGP COL application. However, portions of the standard evaluation material in Section 14.2.9.4 of the BLN SER are directly *applicable to the new STD SUP item identified in the VEGP FSAR. Therefore, the NRC staff used this evaluation material, identified below as standard content material, in the disposition of STD SUP 14.2-2.*

As part of the response to RAI 14.2-12, the applicant proposed to supplement Section 14.2.9 of the AP1000 DCD, Revision 17, with additional administrative controls that will be implemented during preoperational testing activities. The response stated that the control of systems that need to be returned to the construction organization for modifications, repairs, or to correct a new problem will be through administrative procedures. These procedures will also provide directions for the following activities:

- *Release control of systems and/or components to construction*
- *Documentation of the actual work performed and the impact on testing*
- *Identification of required testing to restore the system to an identified status (operability, functionality, availability), as well as the identification of re-performance tests based on the impact of the work performed*
- *Authorizations and tracking of operability and unavailability determinations*
- *Verification activities to ensure that retests stay in compliance with ITAAC commitments*

The staff reviewed this supplemental information related to preoperational test descriptions and determined that it provided adequate administrative controls for an orderly turnover of plant systems when these have to be returned to the construction organization. Therefore, the staff finds this information acceptable. The applicant will revise the BLN COL FSAR to include the proposed administrative controls. This is identified as ***Confirmatory Item 14.2-10****, pending NRC review and approval of the revised BLN COL FSAR.*

Resolution of Standard Content Confirmatory Item 14.2-10

The staff verified that the VEGP applicant has incorporated into its FSAR, in response to STD SUP 14.2-2, the proposed administrative controls identified as Confirmatory Item 14.2-10 in the staff's SER for the BLN COL. On this basis, Confirmatory Item 14.2-10 is resolved.

14.2.9.5 *Post Combined License Activities*

There are no post-COL activities related to this section.

14.2.9.6 *Conclusion*

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to the preoperational test descriptions, and there is no outstanding information to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the information presented in the WLS COL FSAR is acceptable because it meets the requirements of 10 CFR 52.79(a)(28) and Criterion XI of Appendix B to 10 CFR Part 50. It also meets the guidance in NUREG-0800, Section 14.2 and RG 1.68.

The staff based its conclusions on the following:

- WLS DEP 6.4-2, related to design changes affecting how the temperature and humidity in the main control room are maintained within the limits for reliable human performance, is reviewed and found acceptable by the staff in Section 21.3 of this SER.

- STD COL 14.4-5 is acceptable because it provides an adequate description of testing of structures and systems that are outside the scope of the DC.

- STD COL 3.9-5, as it applies to the test abstract for the surge line thermal monitoring, is acceptable because it provides assurance that the test results will quantify the extent of thermal stratification, thermal stripping and piping deflections, as recommended in Bulletin 88-11.

- STD SUP 14.2-2 is acceptable because it provides an adequate description for the development of administrative controls that will be implemented during the preoperational testing activities.

14.2.10 Startup Test Procedures (Related to RG 1.206, Section C.III.1, Chapter 14, C.I.14.2.12, "Individual Test Descriptions")

14.2.10.1 *Introduction*

Startup test procedures address the tests that comprise the startup phase of the test program. For each test, a general description is provided for test objective, test prerequisites, test description, and test performance criteria, where applicable. In describing a test, the operating and safety-related characteristics of the plant to be tested and evaluated are identified. Where applicable, the relevant performance criteria for the test are discussed. Some of the criteria relate to the value of process variables assigned in the design or analysis of the plant, component systems, and associated equipment. Other criteria may be associated with expectations relating to the performance of systems.

14.2.10.2 *Summary of Application*

Section 14.2 of the WLS COL FSAR, Revision 11, incorporates by reference Section 14.2 of the AP1000 DCD, Revision 19. Section 14.2 of the DCD includes Section 14.2.10.

In addition, in WLS COL FSAR Section 14.2.10, the applicant provided the following:

AP1000 COL Information Item

- STD COL 14.4-5

The applicant provided additional information in STD COL 14.4-5 to address interface requirements. This COL item is evaluated by the staff in Section 14.2.9 of this SER.

Supplemental Information

- STD SUP 14.2-3

The applicant provided additional information in STD SUP 14.2-3 to address the development of administrative controls that will be implemented during power ascension testing activities.

14.2.10.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the startup test procedures are given in Section 14.2 of NUREG-0800.

The applicable regulatory requirements for the information being reviewed in this section are 10 CFR 52.79(a)(28) and Criterion XI of Appendix B to 10 CFR Part 50. RG 1.68 provides guidance on how to comply with Criterion XI of Appendix B to 10 CFR Part 50.

14.2.10.4 *Technical Evaluation*

The NRC staff reviewed Section 14.2.10 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the startup test procedures. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the BLN Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from Section 14.2.10.4 of the VEGP SER:

Supplemental Information

- *STD SUP 14.2-3*

The applicant provided additional information in STD SUP 14.2-3 to address the development of administrative controls that will be implemented during power ascension testing activities.

STD SUP 14.2-3 was not in Revision 1 of the BLN FSAR and is addressed for the first time in this SER for the VEGP COL application. However, the standard evaluation material in Section 14.2.9.4 of the BLN SER is directly applicable to the new STD SUP item identified in the VEGP FSAR. Therefore, the NRC staff used this evaluation material, identified below as standard content material, in the disposition of STD SUP 14.2-3.

As part of the response to RAI 14.2-12, the applicant proposed supplemental information in Section 14.2.10 of the BLN COL FSAR, with additional administrative controls that will be implemented during power ascension testing activities consistent with the guidance in RG 1.68 and NUREG-0800. The applicant proposed to discuss a power ascension test plan that will provide controls for operations during the power ascension test phase, including the following:

- *Verification of core performance parameters*
- *Verification of adequate calibration of nuclear instrumentation*
- *Controls for high flux trips consistent with TS requirements*
- *Conduct of surveys of plant systems and equipment*

- *Checks for unexpected radioactivity in process systems and effluents*
- *Perform reactor coolant leak checks*
- *Controls for reviews of testing at each power plateau*

Additionally, the applicant proposed to provide controls for the extrapolation of tests at lower power levels in order to determine the acceptability of performing the test at higher power levels. The applicant proposed to describe measures for the use of surveillance test procedures to document portions of tests, and the use of initial test program tests to satisfy TS surveillance requirements.

The staff reviewed this proposed supplemental information related to the power ascension test phase and determined that it provided adequate administrative controls for activities during power ascension testing. Therefore, the staff finds this information acceptable. The applicant will revise the BLN COL FSAR to include the proposed administrative controls. This is identified as ***Confirmatory Item 14.2-12****, pending NRC review and approval of the revised BLN COL FSAR.*

Resolution of Standard Content Confirmatory Item 14.2-12

The staff verified that the VEGP applicant has incorporated into its FSAR, in response to STD SUP 14.2-3, the proposed administrative controls identified as Confirmatory Item 14.2-12 in the staff's SER for the BLN COL. On this basis, Confirmatory Item 14.2-12 is resolved.

14.2.10.5 *Post Combined License Activities*

There are no post-COL activities related to this section.

14.2.10.6 *Conclusion*

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to the startup test procedures, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the information presented in the WLS COL FSAR is acceptable because it meets the requirements of 10 CFR 52.79(a)(28) and Criterion XI of Appendix B to 10 CFR Part 50. The staff based its conclusions on the following:

- STD SUP 14.2-3 is acceptable because it provides an adequate description of the administrative controls associated with the activities that will be implemented during the power ascension testing phase of the initial test program and meets the guidance in NUREG-0800, Section 14.2 and RG 1.68.

14.3 Certified Design Material (Related to RG 1.206, Section C.III.1, Chapter 14, C.I.14.3, "Inspections, Tests, Analyses, and Acceptance Criteria")

14.3.1 Introduction

This section addresses the selection criteria and processes used to develop the WLS Certified Design Materials (CDMs). It specifically addresses the site-specific inspections, tests, analyses, and acceptance criteria (SS-ITAAC). The COL applicant provides its proposed selection methodology and criteria for establishing the ITAAC that are necessary and sufficient to provide reasonable assurance that the facility has been constructed and will be operated in conformity with the license and the Commission's rules and regulations.

The applicant proposes, in addition to the ITAAC incorporated by reference from the AP1000 DCD, SS-ITAAC to provide reasonable assurance that the facility has been constructed and will operate in conformance with the applicable regulations.

14.3.2 Summary of Application

Section 14.3 of the WLS COL FSAR, Revision 11, incorporates by reference Section 14.3 of the AP1000 DCD, Revision 19.

In addition, in WLS COL FSAR Section 14.3, the applicant provided the following:

Departures

- WLS DEP 3.2-1

The applicant revised DCD Table 14.3-2, "Design Basis Accident Analysis," Sheets 7 and 8 of 17, as new WLS COL FSAR Table 14.3-202, Sheets 1 and 2, providing additional information about WLS DEP 3.2-1 related to design modifications to and performance of the condensate return portion of the Passive Core Cooling System. This information, as well as related WLS DEP 3.2-1 information appearing in other chapters of the WLS COL FSAR, is reviewed in Section 21.1 of the SER.

- WLS DEP 6.4-1

The applicant provided additional information in Section 14.3 of the WLS COL FSAR about WLS DEP 6.4-1 related to design changes affecting habitability of the main control room and changes to the calculated doses to control room operators. This information, as well as related WLS DEP 6.4-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.2 of this SER.

- WLS DEP 6.4-2

The applicant provided additional information in Section 14.3 of the WLS COL FSAR about WLS DEP 6.4-2 related to design changes affecting how the temperature and humidity in the main control room are maintained within the limits for reliable human performance. This information, as well as related WLS DEP 6.4-2 information appearing in other chapters of the FSAR, is reviewed in Section 21.3 of this SER.

- WLS DEP 7.3-1

The applicant provided additional information in Section 14.3 of the WLS COL FSAR about WLS DEP 7.3-1 related to required design changes for the PMS source range neutron flux doubling logic to comply with the requirements of IEEE Std. 603-1991, Clause 6.6. This information, as well as related WLS DEP 7.3-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.5 of this SER.

AP1000 COL Information Items

- STD COL 3.6-1

The applicant provided additional information in STD COL 3.6-1 to provide its plan for completing the pipe rupture hazard analysis.

- STD COL 3.9-7

The applicant provided additional information in STD COL 3.9-7 to provide its plan for completing the piping design.

- WLS COL 2.5-17

The applicant provided additional information in WLS COL 2.5-17 in WLS COL FSAR Section 14.3.3.1, "Waterproof Membrane ITAAC." This section references the design of the waterproof membrane beneath the nuclear island basemat that is described in AP1000 DCD Section 3.4.1.1.1.1.

Supplemental Information

- STD SUP 14.3-1

The applicant provided supplemental information in STD SUP 14.3 1 in WLS COL FSAR Section 14.3.2.3, "Site Specific ITAAC (SS-ITAAC)," and Section 14.3.2.3.3, "Other Site-Specific Systems." Section 14.3.2.3 describes the SS-ITAAC, and Section 14.3.2.3.3 identifies the Transmission Switchyard and Offsite Power System as meeting the ITAAC selection criteria. This section describes the SS-ITAAC.

- WLS SUP 14.3-2

The applicant provided supplemental information in WLS SUP 14.3-2 in WLS COL FSAR Section 14.3.2.3.3, "Other Site-Specific Systems," discussing the ITAAC screening summary for site-specific systems.

14.3.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the CDM are given in Section 14.3 of NUREG-0800.

The applicable regulatory requirements for SS-ITAAC are in 10 CFR 52.80(a) and 10 CFR 52.97, "Issuance of combined licenses."

The regulatory basis for STD COL 3.6-1 and STD COL 3.9-7 are provided in NUREG-0800.

14.3.4 Technical Evaluation

The NRC staff reviewed Section 14.3 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the CDMs. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the BLN Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from Section 14.3.4 of the VEGP SER:

<u>*AP1000 COL Information Items*</u>

- *STD COL 3.6-1 and STD COL 3.9-7*

The portion of STD COL 3.6-1 addressed in VEGP COL FSAR Section 14.3 is the discussion of the ITAAC established to provide reasonable assurance that the design portion of the pipe rupture hazard analysis will be conducted in conformity with the license and the Commission's rules and regulations. The portion of STD COL 3.9-7 addressed in VEGP COL FSAR Section 14.3 is the discussion of the ITAAC established to provide reasonable assurance that the piping design is completed appropriately for applicable systems.

In a letter dated March 18, 2010, as revised by letter dated April 23, 2010, in response to an open item in the NRC staff's SER for BLN (Open Item 3.6-1 in BLN SER Section 3.6.4), the VEGP applicant provided proposed revisions to the VEGP COL application related to the pipe rupture hazard analysis ITAAC. In addition, the applicant provided information related to the piping design ITAAC.

The VEGP applicant proposed to expand FSAR Section 14.3.3 to include, as part of STD COL 3.6-1 and STD COL 3.9-7, a description of the ITAAC established to provide reasonable assurance that the design portion of the pipe rupture hazard analysis and piping design will be conducted in conformity with the license and the Commission's rules and regulations. The applicant proposed revision of two license conditions in Part 10 of the COL application to address when the information would be available for staff review and expanding Appendix B of Part 10 to include the two ITAAC associated with review of the pipe rupture hazard analysis and the piping design. STD COL 3.6-1 and STD COL 3.9-7 are evaluated by the staff in Sections 3.6 and 3.12 respectively, of this SER, including the proposed pipe rupture hazard analysis ITAAC and piping design ITAAC.

- WLS COL 2.5-17

The NRC staff reviewed WLS COL 2.5-17 in WLS COL FSAR Section 14.3.3.1 related to waterproof membrane ITAAC. The staff's review of this ITAAC is documented in Section 3.8.5 of this SER.

<u>Supplemental Information</u>

- STD SUP 14.3-1, addressing SS-ITAAC

- WLS SUP 14.3-2, addressing ITAAC screening summary for additional site-specific systems

The following portion of this technical evaluation section is reproduced from Section 14.3.4 of the VEGP SER:

> *The following portion of this technical evaluation section is reproduced from Section 14.3 of the BLN SER. This portion of the BLN SER combined the evaluation of STD SUP 14.3-1 and BLN SUP 14.3-2. The NRC staff concludes that the evaluation of BLN SUP 14.3-2 applies to VEGP SUP 14.3-2, based on the similarities of these two plant-specific supplemental items.*

The NRC staff concludes that the evaluation of BLN SUP 14.3-2 and VEGP SUP 14.3-2 applies to WLS SUP 14.3-2, based on the similarities of these three plant-specific supplemental items.

The following portion of this technical evaluation section is reproduced from Section 14.3.4 of the VEGP SER:

> *As part of STD SUP 14.3-1 and BLN SUP 14.3-2, the applicant provided:*
>
> - *Site-specific ITAAC selection criteria*
> - *Site-specific ITAAC selection methodology*
> - *Site-specific ITAAC screening summary*
>
> *A table of ITAAC entries was provided for each site-specific system described in the BLN COL FSAR that meets the selection criteria, and that is not included in the certified design. The COL applicant adopted the same selection criteria and methodology as the AP1000 DCD for establishing the SS-ITAAC. The selection criteria and methodology contained in the AP1000 DCD was accepted by the NRC as described in NUREG-1793. Therefore, the staff finds the applicant's use of this criteria and methodology appropriate and acceptable. The ITAAC are provided in tables with information for the following three columns: design commitment; inspection, tests, analyses; and acceptance criteria.*
>
> *Emergency Planning-ITAAC (EP-ITAAC) are discussed in the application as required for inclusion in accordance with 10 CFR 52.80(a). The site-specific EP-ITAAC are based on the generic ITAAC provided in Appendix C.II.1-B of RG 1.206. The staff's review of the current set of EP-ITAAC and the information related to this ITAAC is contained in Chapter 13.6 [13.3] of the SER.*
>
> *Physical Security-ITAAC (PS-ITAAC) are discussed in the application as required for inclusion in accordance with 10 CFR 52.80(a). The site-specific PS-ITAAC are based on the generic ITAAC provided in Appendix C.II.1-C of RG 1.206. The NRC staff's review of the current set of PS-ITAAC and the information related to this ITAAC is contained in Chapter 13.4 [13.6] of the SER.*
>
> *The NRC staff reviewed the supplemental information relating to ITAACs included under Section 14.3.2 of the BLN COL. The applicant identified no additional site-specific systems meeting the ITAAC selection criteria. With the exception of the Transmission Switchyard and Offsite Power System, the staff*

agrees no additional site-specific ITAAC are required in accordance with 10 CFR 52.80(a).

In RAI-14.3-1, the staff asked the applicant to justify the omission of site-specific ITAAC for transmission switchyard and the offsite power system. Subsequently, in a letter dated May 11, 2009, the applicant agreed to include an ITAAC in the BLN COL FSAR for transmission switchyard and the offsite power system. The information related to this ITAAC is evaluated in Chapter 8 of the SER. This is Confirmatory Item 14.3-1, pending NRC review and approval of the revised BLN COL FSAR.

Resolution of Standard Content Confirmatory Item 14.3-1

Confirmatory Item 14.3-1 required the applicant to update its FSAR to include proposed ITAAC for the offsite power system. The NRC staff provides its evaluation of the proposed ITAAC for the offsite power system in Section 8.2.A of this SER. The NRC staff verified that the VEGP COL application was appropriately updated. As a result, Confirmatory Item 14.3-1 is resolved.

14.3.5 Post Combined License Activities

The SS-ITAAC in the previous section of this SER are considered post-COL activities and discussed in the individual SER sections as stated above.

14.3.6 Conclusion

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to the test program schedule, and there is no outstanding information to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the information presented in the WLS COL FSAR is acceptable because it meets the requirements of 10 CFR 52.80(a) and 10 CFR 52.97. The staff based its conclusions on the following:

- WLS DEP 3.2-1, related to design modifications to the condensate return portion of the Passive Core Cooling System, is reviewed and found acceptable by the staff in Section 21.1 of this SER.

- WLS DEP 6.4-1, related to design changes affecting habitability of the main control room and changes to the calculated doses to control room operators, is reviewed and found acceptable by the staff in Section 21.2 of this SER.

- WLS DEP 6.4-2, related to design changes affecting how the temperature and humidity in the main control room are maintained within the limits for reliable human performance, is reviewed and found acceptable by the staff in Section 21.3 of this SER.

- WLS DEP 7.3-1, related to required design changes for the PMS source range neutron flux doubling logic to comply with the requirements of IEEE Std. 603-1991, Clause 6.6, is reviewed and found acceptable by the staff in Section 21.5 of this SER.

- STD SUP 14.3-1, WLS SUP 14.3-2 and WLS COL 2.5-17 are acceptable because the ITAAC specified for the site-specific systems provide adequate assurance that these systems have been constructed and will be operated in conformity with the license and the Commission's rules and regulations.

15.0 ACCIDENT ANALYSIS

The evaluation of the safety of a nuclear power plant includes analyses of the plant's responses to postulated disturbances in process variables and postulated equipment failures or malfunctions. Such safety analyses provide a significant contribution to the selection of limiting conditions for operation, limiting safety system settings, and design specifications for components and systems from the standpoint of public health and safety. These analyses are a focal point of the combined license (COL) reviews. In Chapter 15 of the Final Safety Analysis Report (FSAR), the COL applicant discussed the applicable transient and accident analyses to justify its conformance to the applicable regulations.

The U.S. Nuclear Regulatory Commission (NRC) staff's review of William States Lee III Nuclear Station (WLS) COL FSAR Chapter 15 follows the format in WLS Chapter 15.

15.0 Accident Analysis (Related to Regulatory Guide (RG) 1.206, Section C.III.1, Chapter 15, C.I.15.1, "Transient and Accident Classification," C.I.15.2, "Frequency of Occurrence," C.I.15.3, "Plant Characteristics Considered in the Safety Evaluation," C.I.15.4, "Assumed Protection System Actions," and C.I.15.5, "Evaluation of Individual Initiating Events")

15.0.1 Introduction

Design basis transient and accident analyses are required as a part of an evaluation of the safety of a nuclear power plant to evaluate the plant's responses to postulated disturbances in process variables and postulated equipment failures or malfunctions. The safety analyses provide a significant contribution to the determination of limiting conditions for operation, limiting safety system settings, and design specifications for plant components and systems to protect public health and safety.

15.0.2 Summary of Application

Section 15.0 of the WLS COL FSAR, Revision 11, incorporates by reference Section 15.0 of the AP1000 Design Control Document (DCD), Revision 19.

Departures

- WLS DEP 3.2-1

The applicant provided additional information about WLS DEP 3.2-1 in Section 15.0.13 of the FSAR related to the performance of the condensate return portion of the Passive Core Cooling System. This information, as well as related WLS DEP 3.2-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.1 of this report.

- WLS DEP 6.4-1

The applicant provided additional information in Section 15.0.11 of the WLS COL FSAR about WLS DEP 6.4-1 related to design changes affecting habitability of the main control room and changes to the calculated doses to control room operators. This information, as well as related

WLS DEP 6.4-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.2 of this SER.

AP1000 COL Information Item

- STD COL 15.0-1

In letters dated November 4, 2010, and April 25, 2011, the applicant endorsed Vogtle Electric Generating Plant (VEGP) letters dated May 21, 2010, October 29, 2010, and February 8, 2011. In these letters, the applicant proposed Standard (STD) COL 15.0-1, adding new text to WLS COL FSAR Section 15.0. STD COL 15.0-1 was provided in a response to a request for additional information (RAI) related to the AP1000 design certification (DC) amendment review. Specifically, in its response dated May 6, 2009, to NRC RAI AP1000 DCD RAI-SRP15.0-SRSB-02, Westinghouse proposed COL Information Item 15.0-1 to provide documentation of the plant calorimetric uncertainty methodology. RAI-SRP15.0-SRSB-02 noted that the AP1000 DCD assumes a 2 percent power uncertainty for the initial condition for most transients and accidents. However, a 1 percent power uncertainty is assumed for the initial reactor power for the large-break loss-of-coolant accident (LOCA) in AP1000 DCD Section 15.6.5.4A, as well as the mass and energy release calculation in AP1000 DCD Sections 6.2.1.3 and 6.2.1.4. In response to this RAI, Westinghouse proposed a new COL information item to be included in a future revision to AP1000 DCD Section 15.0.15. COL Information Item 15.0-1 states:

> Following selection of the actual plant operating instrumentation and calculation of the instrumentation uncertainties of the operating plant parameters prior to fuel load, the Combined License holder will calculate the primary power calorimetric uncertainty. The calculations will be completed using an NRC acceptable method and confirm that the safety analysis primary power calorimetric uncertainty bounds the calculated values.

License Conditions

- License Condition 2, Item 15.0-1

In Part 10 of the COL application, the applicant provided License Condition 2 to address numerous COL items, including COL Information Item 15.0-1 related to documentation of plant calorimetric uncertainty methodology. The license condition for COL Information Item 15.0-1 is addressed by ITAAC Table 2.5.4-2, Item 4.

- License Condition 6, Items (j) and (k)

In Part 10 of the COL application, the applicant provided License Condition 6 to provide schedules to NRC to support NRC inspection of operational programs and other applicant activities, including activities related to power calorimetric uncertainty.

Inspections, Tests, Analyses and Acceptance Criteria

In Part 10 of the COL application, the applicant provided an ITAAC to address the instrumentation to measure feedwater flow and the calculation methodology for plant calorimetric uncertainty.

15.0.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," and its supplements.

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic licensing of production and utilization facilities," Appendix K, "ECCS [Emergency Core Cooling System] Evaluation Models," specifies that an assumed power level lower than 1.02 times the licensed power level (but not less than the licensed power level) may be used provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error. The review guidance in Section 15.0 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section I.3, "Plant Characteristics in the Safety Evaluation," states in part that "the reviewer also ensures that the application specifies the permitted fluctuations and uncertainties associated with reactor system parameters and assumes the appropriate conditions, within the operating band, as initial conditions for transient analysis."

15.0.4 Technical Evaluation

The NRC staff reviewed Section 15.0 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to accident analysis. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this safety evaluation report (SER) provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

[1] See Section 1.2.2 for a discussion of the staff's review related to verification of the scope of information to be included in a COL application that references a DC.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting.

The following portion of this technical evaluation section is reproduced from Section 15.0.4 of the VEGP SER:

> *AP1000 COL Information Item*
>
> - *STD COL 15.0-1*
>
> *In a letter dated May 21, 2010, as revised by a letter dated October 29, 2010, the VEGP applicant submitted information to address COL Information Item 15.0-1. In these letters, the applicant stated that the plant operating instrumentation for feedwater flow measurement would be the Caldon/Cameron LEFM CheckPlus™ system and referenced the NRC staff's final safety evaluation that approved the Caldon topical report, ER-157P, Revision 8, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or Checkplus™ System." The NRC staff has previously approved several plant applications of the Caldon/Cameron CheckPlus™ LEFM system to support a power measurement uncertainty lower than 1 percent. This AP1000 COL information item supports the 1 percent power uncertainty. The NRC staff's review herein focused on ensuring that the generically approved Caldon/Cameron topical reports are properly implemented for the VEGP COL application. The NRC staff verified compliance with the applicable conditions in the NRC staff's safety evaluations approving the topical reports. The NRC staff's review also confirmed that appropriate license conditions and ITAAC were established for those items that cannot be resolved prior to issuance of the COL.*
>
> *Compliance with Caldon/Cameron Topical Report ER-80P*
>
> *NRC staff approval of the Caldon/Cameron topical report ER-80P (safety evaluation (SE) dated March 8, 1999) established four criteria to be satisfied by each applicant or licensee. The VEGP applicant addressed each criterion as described below.*
>
>> *Criterion 1*
>>
>> *Discuss maintenance and calibration procedures that will be implemented with the incorporation of the LEFM, including processes and contingencies for inoperable LEFM instrumentation and the effect on thermal power measurements and plant operation.*

The VEGP applicant stated that calibration and maintenance programs would be developed in accordance with the Caldon/Cameron LEFM technical manuals and recommendations. Preventative Maintenance (PM) tasks would be periodically performed within the plant control system and support systems to provide continued reliability. Plant instrumentations that affect the power calorimetric, including the Caldon/Cameron LEFM CheckPlus™ inputs, would be monitored by plant system engineering personnel. These instruments would be included in the plant PM program for periodic calibration. The NRC staff finds these measures acceptable.

The VEGP applicant stated when the Caldon/Cameron LEFM CheckPlus™ flow meter becomes inoperable beyond the allowed outage time; the plant would be operated at de-rated conditions. De-rated operation is appropriate at power levels consistent with a 2 percent power uncertainty. With the plant operating at 100 percent load with 1 percent uncertainty, a de-rating to 99 percent maintains a 2 percent uncertainty. When the LEFM CheckPlus™ is inoperable, plant calorimetric power would be monitored with the use of feedwater venturi elements. An inoperable LEFM would not leave the plant in a condition where steady-state operation would be immediately compromised since it would not directly impact the calibration of the nuclear instrumentation utilized for power level related trips or safety system actuations. Thus, procedures require confirmation of the availability of alternate instrumentation (i.e., the feedwater venturi instrumentation) and initiation of the above described reduction in power within 48 hours. These measures are consistent with the operating plants. The NRC staff finds that operation with an inoperable Caldon/Cameron CheckPlus™ has been acceptably addressed.

Criterion 2

For plants that currently have LEFMs installed, provide an evaluation of the operational and maintenance history of the installed instrumentation and confirmation that the installed instrumentation is representative of the LEFM system and bounds the analyses and assumptions set forth in TR ER-80P.

The VEGP applicant stated that, since this application represents construction of a new plant with no previously installed LEFM equipment, this item is not applicable. The NRC staff finds the VEGP applicant's response acceptable.

Criterion 3

Confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current feedwater instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternative approach is used, the application should be justified and applied to both venturi and ultrasonic flow measurement instrumentation installations for comparison.

The VEGP applicant stated that the uncertainty of the LEFM would be calculated in accordance with the Westinghouse methodology as applied in the Beaver Valley Power Station Units 1 and 2 License Amendment Request Nos. 289 and 161, which was approved by the NRC staff in a letter dated September 24, 2001, titled, "Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2) – Issuance of Amendment Re: 1.4-Percent Power Uprate and Revised BVPS-2 Heatup and Cooldown Curves." The NRC staff reviewed this SE and found that the calculation methodology complies with the recommendations of American National Standards Institute/Independent Safety Assessment (ANSI/ISA) Standard 67.04-2000, "Setpoints for Nuclear Safety-Related Instrumentation," and Regulatory Guide (RG) 1.105, "Setpoints for Safety-Related Instrumentation," Revision 2. In these calculations, uncertainties for the parameters that are not statistically independent are arithmetically summed to produce groups that are independent of each other, which can be statistically combined. Then, all independent parameters/groups that contribute to the power measurement uncertainty are combined using a square root of sum of squares (SRSS) approach to determine the overall power measurement uncertainty. This methodology has been reviewed and approved by the NRC staff for Westinghouse pressurized-water reactors (PWRs) (e.g., Beaver Valley), and is also acceptable for AP1000, which is a Westinghouse-designed PWR. The staff finds the AP1000 design sufficiently similar to other Westinghouse PWR designs that have been approved such that the methodology applies to both designs. Therefore, the NRC staff finds that the VEGP applicant's response acceptable.

> *Criterion 4*
>
> *Licensees for plant installations where the ultrasonic meter (including LEFM) was not installed with flow elements calibrated to a site specific piping configuration (flow profiles and meter factors not representative of the plant specific installation), should provide additional justification for use. This justification should show that the meter installation is either independent of the plant specific flow profile for the stated accuracy, or that the installation can be shown to be equivalent to known calibrations and plant configurations for the specific installation including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed calibrated elements, the licensee should confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.*

The VEGP applicant stated that its application represents construction of a new plant with no previously installed flow metering equipment. The AP1000 main feedwater flow measurement instrumentation, consistent with the use of normalized flow meters, would be required to be calibrated at a certified test laboratory in hydraulic model geometry consistent with the AP1000 plant design. The LEFM commissioning process (i.e., installation acceptance testing) would confirm that the actual instrument performance is consistent with the *assumptions of the uncertainty calculation. The NRC staff finds this response acceptable.*

Compliance with Caldon/Cameron Topical Report ER-157P, Revision 8

The VEGP applicant addressed the five SE conditions found in the NRC SE for ER-157P, Revision 8, dated August 16, 2010, as described below.

> *Condition 1*
>
> *Continued operation at the pre-failure power level for a pre-determined time and the decrease in power that must occur following that time are plant-specific and must be acceptably justified.*

The VEGP applicant stated that a failure of the ultrasonic flow meter (UFM) will result in the use of the feedwater venturi as the input into the calorimetric calculation. Since the contingency is not based on continued reliance on the CheckPlus™ system, the NRC staff finds the VEGP applicant's response acceptable.

> *Condition 2*
>
> *A CheckPlus operating with a single failure is not identical to an LEFM Check. Although the effect on hydraulic behavior is expected to be negligible, this must be acceptably quantified if a licensee wishes to operate using the degraded CheckPlus at an increased uncertainty.*

The VEGP applicant stated that a degraded UFM resulting in an instrument uncertainty greater than the values assumed in the AP1000 calorimetric uncertainty calculation would be considered a failure and subject to compensatory actions as discussed above in response to Caldon/Cameron topical report (ER-80P) Criterion 1. Since the applicant does not intend to operate using a degraded CheckPlus™, *the NRC staff finds the VEGP applicant's response acceptable.*

> *Condition 3*
>
> *An applicant with a comparable geometry can reference the above Section 3.2.1 [of the SE for ER-157P] finding to support a conclusion that downstream geometry does not have a significant influence on CheckPlus calibration. However, CheckPlus test results do not apply to a Check and downstream effects with use of a CheckPlus with disabled components that make the CheckPlus comparable to a Check must be addressed. An acceptable method is to conduct applicable Alden Laboratory tests.*

The VEGP applicant stated that the AP1000 feedwater flow measurement instrumentation would be located in piping with downstream geometry more favorable than the arrangements referenced in Section 3.2.1 of the SE for ER-157P. Therefore, the effects of downstream piping geometry are not considered to have a significant influence on the accuracy of the UFM. Because the flow measurement instrumentation would be located in piping with favorable downstream geometry, the NRC staff finds the VEGP applicant's response acceptable.

> *Condition 4*
>
> *An applicant that requests a MUR [measurement uncertainty recapture] with the upstream flow straightener configuration discussed in Section 3.2.2 [of the SE for ER-157P] should provide justification for claimed CheckPlus uncertainty that extends the justification provided in Reference 17 [Letter from E. Hauser dated March 19, 2010]. Since the Reference 17 evaluation does not apply to the Check, a comparable evaluation must be accomplished if a Check is to be installed downstream of a tubular flow straightener.*

The VEGP applicant stated that the AP1000 UFM installation would not utilize an upstream flow straightener. Therefore, this condition is not applicable to the AP1000 design. The NRC staff finds the VEGP applicant's response acceptable.

> *Condition 5*
>
> *An applicant assuming large uncertainties in steam moisture content should have an engineering basis for the distribution of the uncertainties or, alternatively, should ensure that their calculations provide margin sufficient to cover the differences shown in Figure 1 of Reference 18 [Letter from E. Hauser dated March 18, 2010].*

The VEGP applicant stated that this AP1000 application of the CheckPlus™ LEFM is to support a 1 percent overall power uncertainty, as compared to lower than 0.5 percent typically justified for operating plants using CheckPlus™. The result of this application of the LEFM at a higher uncertainty (i.e., lower accuracy) is that the assumed steam separator/dryer performance becomes less of a relative contribution to the overall uncertainty. Furthermore, an engineering basis for the AP1000 moisture content assumption is in the calorimetric uncertainty calculation. Because the steam separator/dryer performance uncertainty is a relatively small contribution to the overall uncertainty of 1 percent, the NRC staff finds the VEGP applicant's response acceptable.

Based on its review of the VEGP applicant's responses, the NRC staff finds that the licensee has acceptably addressed all applicable conditions specified in the NRC staff's SEs for the Caldon/Cameron topical reports. Hence, the NRC staff finds that the Caldon/Cameron topical reports, ER-80P and ER-157P, are *acceptable for referencing in the VEGP COL application and that the applicant has adequately addressed COL Information Item 15.0-1.*

License Conditions

- *License Condition 2, Item 15.0-1*

In a letter dated May 21, 2010, the applicant proposed adding Item 15.0-1 to License Condition 2 that would confirm that the plant operating instrumentation installed for feedwater flow measurement is a Caldon/Cameron LEFM CheckPlus™ system. In its October 29, 2010, letter, the applicant revised Item 15.0-1 to state that the documentation of plant calorimetric uncertainty methodology would be addressed as a plant-specific ITAAC item in lieu of License Condition 2. The staff finds the use of ITAAC to confirm proper documentation of plant calorimetric uncertainty methodology to be acceptable. The plant-specific ITAAC item proposed by the applicant is evaluated below.

- *License Condition 6*

In a letter dated October 29, 2010, the applicant proposed adding new line items to proposed License Condition 6, associated with the power calorimetric uncertainty instrumentation. Specifically, the applicant proposed to add the following two items:

- *The availability of documented instrumentation uncertainties to calculate a power calorimetric uncertainty (prior to initial fuel load).*

- *The availability of administrative controls to implement maintenance and contingency activities related to the power calorimetric uncertainty instrumentation (prior to initial fuel load).*

The two items under License Condition 6 are needed because documentation for the actual instrument uncertainties would only be available after the equipment is procured and tested and administrative controls would not be available until after the equipment is procured, which would be after the COL license is issued. The staff finds the first item acceptable because, when combined with the methodology in the proposed ITAAC, it would allow the staff to confirm that the procured equipment results in a power uncertainty of no more than 1 percent prior to the start of plant operation. The staff finds the second item acceptable because it would allow the staff to confirm that the administrative controls are in place to meet ER-80P Criterion 1 prior to the start of plant operation. These items correspond to License Condition 15-1 in the following section.

Inspections, Tests, Analyses and Acceptance Criteria

In a letter dated October 29, 2010, the applicant proposed ITAAC associated with the plant calorimetric uncertainty methodology. The proposed ITAAC item is repeated in Table 15.0-1 of this SER. This ITAAC would confirm that: (1) the installed feedwater flow measurement device is the Caldon CheckPlus™ LEFM;

(2) the power calorimetric uncertainty calculation for that instrumentation is based on an acceptable Westinghouse methodology as described above in Criterion 3 for ER-80P and the uncertainty values in the calculation for that instrumentation are not lower than those for the actual installed instrumentation; and (3) the calculated calorimetric power uncertainty measurement values are bounded by the 1 percent uncertainty value assumed for the initial reactor power in the safety analysis. The proposed ITAAC would allow the NRC staff to confirm, prior to initial fuel load, that the necessary conditions for STD COL 15.0-1 (COL Information Item 15.0-1) have been satisfied. Therefore, the NRC staff found the proposed ITAAC acceptable.

The incorporation of the planned changes to the VEGP COL FSAR detailed in the applicant's letters dated May 21, 2010, and October 29, 2010 will be tracked as ***Confirmatory Item 15.0-1****.*

Resolution of Standard Content Confirmatory Item 15.0-1

Confirmatory Item 15.0-1 is an applicant commitment to revise its FSAR Section 15.0 to address COL Information Item STD COL 15.0-1. The staff verified that the VEGP COL FSAR was appropriately revised. As a result, Confirmatory Item 15.0-1 is now closed.

Evaluation of Additional Information Submitted by Applicant

In a letter dated February 6, 2011, submitted in response to a January 24, 2011, letter from the ACRS, the applicant provided additional information related to the flow meter instrumentation, including proposed changes to the FSAR. The applicant stated that, prior to installation, the LEFM CheckPlus^{TM} system will be calibrated at a certified facility with a test model representative of plant piping configurations. After installation in the plant, the LEFM CheckPlus^{TM} system will be tested in accordance with the LEFM CheckPlus^{TM} system commissioning procedure developed by Cameron to confirm that the actual instrument performance is consistent with the assumption of the uncertainty calculation. The staff found these changes acceptable because they clarified the applicant commitment regarding calibration and testing of the instrument. The staff verified that the VEGP COL FSAR was revised to include the proposed changes.

15.0.5 Post Combined License Activities

For the reasons discussed in the technical evaluation section above, the staff proposes to include the following ITAAC:

- The licensee shall perform and satisfy the plant calorimetric uncertainty and plant instrumentation performance analysis ITAAC defined in SER Table 15.0-1, “Power Calorimetric Uncertainty Methodology.”

For the reasons discussed in the technical evaluation section above, the staff finds the following license condition acceptable:

- License Condition (15-1) – No later than 12 months after issuance of the COL, the licensee shall submit to the Director of Office of New Reactors a schedule that supports planning for and conduct of NRC inspections of license calculations for power calorimetric uncertainty and administrative controls to implement maintenance and contingency activities related to the power calorimetric uncertainty instrumentation. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until the license condition has been fully implemented. This schedule shall address:
    - The availability of documented instrumentation uncertainties to calculate a power calorimetric uncertainty (prior to initial fuel load).
    - The availability of administrative controls to implement maintenance and contingency activities related to the power calorimetric uncertainty instrumentation (prior to initial fuel load).

15.0.6 Conclusion

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to accident analysis and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL application is acceptable and meets the NRC regulations. The staff based its conclusion on the following:

- WLS DEP 3.2-1, related to design modifications to the condensate return portion of the passive core cooling system, is reviewed and found acceptable by the staff in Section 21.1 of this SER.
- WLS DEP 6.4-1, related to design changes affecting habitability of the main control room and changes to the calculated doses to control room operators, is reviewed and found acceptable by the staff in Section 21.2 of this SER.
- STD COL 15.0-1 is acceptable because the applicant has demonstrated that the conditions identified by the NRC in its generic evaluation have been satisfied for the use of the Caldon/Cameron LEFM CheckPlus™ system for WLS Units 1 and 2. In addition, ITAAC and a license condition have been put in place to allow the staff to verify the plant calorimetric uncertainty methodology prior to initial fuel load.

15.1 Increase in Heat Removal from the Primary System (Related to RG 1.206, Section C.III.1, Chapter 15, C.I.15.6, "Event Evaluation")

Analyses focused on the increase in heat removal from the primary system address anticipated operational occurrences (AOOs) and accidents that increase the heat removal by the secondary system, which could result in a decrease in reactor coolant temperature. Increased heat removal can be caused by:

- Feedwater system malfunctions causing a reduction in feedwater temperature
- Feedwater system malfunctions causing an increase in feedwater flow
- Excessive increase in secondary steam flow
- Inadvertent opening of a steam generator relief or safety valve
- Steam system piping failure
- Inadvertent operation of the passive residual heat removal heat exchanger

Section 15.1 of the WLS COL FSAR, Revision 11, incorporates by reference, Section 15.1, “Increase in Heat Removal from the Primary System,” of Revision 19 of the AP1000 DCD. In addition, in the WLS COL FSAR, the applicant provided the following:

Departures

- WLS DEP 6.4-1

The applicant provided additional information in Section 15.1.5 of the WLS COL FSAR about WLS DEP 6.4-1 related to design changes affecting habitability of the main control room and changes to the design basis accident (DBA) radiological consequences analyses, including calculated doses to control room operators and offsite. This information, as well as related WLS DEP 6.4-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.2 of this SER.

The NRC staff reviewed Section 15.1 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this section. The NRC staff’s review confirmed that the applicant addressed the required information to satisfy the evaluation criteria. There is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

15.2 Decrease in Heat Removal By the Secondary System

Analyses focused on the decrease in heat removal by the secondary system address AOOs and accidents that could result in a reduction of the capacity of the secondary system to remove heat generated in the reactor coolant system (RCS). Decreased heat removal can be caused by:

- Steam pressure regulator malfunction or failure that results in decreasing steam flow
- Loss of external electrical load
- Turbine trip
- Inadvertent closure of main steam isolation valves
- Loss of condenser vacuum and other events resulting in turbine trip
- Loss of alternating current (ac) power to station auxiliaries
- Loss of normal feedwater flow
- Feedwater system pipe break

Section 15.2 of the WLS COL FSAR, Revision 11, incorporates by reference, Section 15.2, "Decrease in Heat Removal by the Secondary System," of Revision 19 of the AP1000 DCD. In addition, in the WLS COL FSAR, the applicant provided the following:

Departures

- WLS DEP 6.3-1

The applicant provided additional information about WLS DEP 6.3-1 in Section 15.2.6 of the FSAR related to quantifying the duration that the passive residual heat removal system heat exchanger can maintain safe shutdown conditions, changing the indefinite duration to greater than 14 days. This information, as well as related WLS DEP 6.3-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.1 of this report.

- WLS DEP 3.2-1

The applicant provided additional information about WLS DEP 3.2-1 in Section 15.2 of the FSAR related to the performance of the condensate return portion of the Passive Core Cooling System. This information, as well as related WLS DEP 3.2-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.1 of this report.

The NRC staff reviewed Section 15.2 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this section. The NRC staff's review confirmed that the applicant addressed the required information to satisfy the evaluation criteria. There is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

15.3 Decrease in Reactor Coolant System Flow Rate

Analyses focused on the decrease in RCS flow rate address AOOs and accidents that could result in a decrease in the RCS flow rate. Decreased flow rate can be caused by:

- Partial loss of forced reactor coolant flow
- Complete loss of forced reactor coolant flow
- Reactor coolant pump (RCP) shaft seizure (locked motor)
- RCP shaft break

Section 15.3 of the WLS COL FSAR, Revision 11, incorporates by reference, Section 15.3, "Decrease in Reactor Coolant System Flow Rate," of Revision 19 of the AP1000 DCD. In addition, in the WLS COL FSAR, the applicant provided the following:

Departures

- WLS DEP 6.4-1

The applicant provided additional information in Section 15.3.3 of the WLS COL FSAR about WLS DEP 6.4-1 related to design changes affecting habitability of the main control room and changes to the DBA radiological consequences analyses, including calculated doses to control room operators and offsite. This information, as well as related WLS DEP 6.4-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.2 of this SER.

The NRC staff reviewed Section 15.3 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this section. The NRC staff's review confirmed that the applicant addressed the required information to satisfy the evaluation criteria. There is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

15.4 <u>Reactivity and Power Distribution Anomalies</u>

15.4.1 Introduction

Analyses focused on reactivity and power distribution anomalies address AOOs and accidents that could result in anomalies in the reactivity or power distribution in the reactor core. Reactivity and power distribution anomalies can be caused by:

- Uncontrolled rod cluster control assembly (RCCA) bank withdrawal from a subcritical or low-power startup condition
- Uncontrolled RCCA bank withdrawal at power
- RCCA misalignment
- Startup of an inactive RCP at an incorrect temperature
- Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant
- Inadvertent loading and operation of a fuel assembly in an improper position
- Spectrum of RCCA ejection accidents

15.4.2 Summary of Application

Section 15.4 of the WLS COL FSAR, Revision 11, incorporates by reference Section 15.4 of the AP1000 DCD, Revision 19. In addition, in the WLS COL FSAR, the applicant provided the following:

<u>Departures</u>

- WLS DEP 6.4-1

The applicant provided additional information in Sections 15.4.8 and 15.4.10 of the WLS COL FSAR about WLS DEP 6.4-1 related to design changes affecting habitability of the main control room and changes to the DBA radiological consequences analyses, including calculated doses to control room operators and offsite. This information, as well as related WLS DEP 6.4-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.2 of this SER.

Generic Letter 85-05

In its letter dated November 4, 2010, the applicant endorsed a letter dated January 22, 2010, from the VEGP applicant that proposed to include Generic Letter (GL) 85-05, "Inadvertent Boron Dilution Events," in Table 1.9-204 of the FSAR as part of STD COL 1.9-2 to address Bulletins and GLs.

15.4.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

15.4.4 Technical Evaluation

The NRC staff reviewed Section 15.4 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to reactivity and power distribution anomalies. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) contains evaluation material from the SER for the Bellefonte Nuclear Plant (BLN), Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from Section 15.4.4 of the VEGP SER:

Generic Letter 85-05

GL 85-05, "Inadvertent Boron Dilution Events," informed each PWR licensee of the NRC staff position resulting from the evaluation of Generic Issue 22, "Inadvertent Boron Dilution Events," and urges each licensee to ensure that its plants have adequate protection against boron dilution events. GL 85-05 was evaluated as a part of the AP1000 DCD review, and the evaluation was documented in NUREG-1793, Chapter 20. GL 85-05 was resolved based on the analyses of inadvertent boron dilution events described in AP1000 DCD Section 15.4.6, which show that in all modes of operation the inadvertent boron dilution is prevented or responded to by automatic functions, or sufficient time is available for operator action to terminate the transient. The staff also stated that COL applicants should develop plant-specific emergency operating procedures (EOPs) that address the boron dilution events. The development of EOPs is identified as COL Information Item 13.5-1, Plant Procedures, which is addressed in BLN FSAR Section 13.5. Therefore, based on the above, the applicant needs to reinsert a reference to GL 85-05 in FSAR Table 1.9-204 and provide a cross reference to COL Information Item 13.5-1. This is ***Open Item 15.4-1.***

Resolution of Standard Content Open Item 15.4-1

To address Open Item 15.4-1 in the BLN SER with open items, the VEGP applicant stated in its letter dated January 22, 2010, that VEGP COL FSAR Table 1.9-204, "Generic Communications Assessment," would be revised to list GL 85-05 with a cross-reference to VEGP COL FSAR Section 13.5. Until this change is incorporated in a future version of the VEGP COL FSAR, this item is being tracked as ***Confirmatory Item 15.4-1.***

Resolution of Standard Content Confirmatory Item 15.4-1

Confirmatory Item 15.4-1 is an applicant commitment to revise its FSAR Table 1.9-204 to list GL 85-05 with a cross-reference to VEGP COL FSAR Section 13.5. The staff verified that the VEGP COL FSAR was appropriately revised. As a result, Confirmatory Item 15.4-1 is now closed.

15.4.5 Post Combined License Activities

There are no post-COL activities related to this section.

15.4.6 Conclusion

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to reactivity and power distribution anomalies, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR related to GL 85-05 is acceptable. Plant-specific EOPs, which will include responding to abnormal events such as the boron dilution events discussed in GL 85-05, are evaluated by the staff in Section 13.5 of this SER. WLS DEP 6.4-1, related to design changes affecting habitability of the main control room and changes to the calculated doses to control room operators, is reviewed and found acceptable by the staff in Section 21.2 of this SER.

15.5 Increase in Reactor Coolant Inventory

Analyses focused on the increase in reactor coolant inventory address AOOs that could result in an increase in RCS inventory. Increased inventory can be caused by:

- Inadvertent operation of the core makeup tanks during power operation
- Chemical and volume control system malfunctions that increases reactor coolant inventory

Section 15.5 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 15.5, “Increase in Reactor Coolant Inventory,” of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff’s review confirmed that there is no outstanding issue related to this section. The results of the NRC staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

15.6 Decrease in Reactor Coolant Inventory

Analyses focused on the decrease in reactor coolant inventory address AOOs and accidents that could result in a decrease in RCS inventory. Decreased inventory can be caused by the following:

- Inadvertent opening of a pressurizer safety valve or inadvertent operation of the automatic depressurization system

- Failure of small lines carrying primary coolant outside containment

- Steam generator tube failure

- LOCA resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (RCPB).

Section 15.6 of the WLS COL FSAR, Revision 11, incorporates by reference Section 15.6, “Decrease in Reactor Coolant Inventory,” of Revision 19 of the AP1000 DCD. In addition, in the WLS COL FSAR, the applicant provided the following:

Departures

- WLS DEP 6.4-1

The applicant provided additional information in Sections 15.6.2, 15.6.3, 15.6.5, and 15.6.6 of the WLS COL FSAR about WLS DEP 6.4-1 related to design changes affecting habitability of the main control room and changes to the DBA radiological consequences analyses, including calculated doses to control room operators and offsite. This information, as well as related WLS DEP 6.4-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.2 of this SER.

AP1000 COL Information Item

- WLS COL 2.3-4

The applicant provided additional information in WLS COL 2.3-4 related to site characteristic[2] atmospheric dispersion factor (χ/Q) values. The effect of WLS COL 2.3-4 on the design-basis accident (DBA) radiological consequences analyses is addressed in Section 15A of this SER.

The NRC staff reviewed Section 15.6 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this section. The NRC staff's review confirmed that the applicant addressed the required information to satisfy the evaluation criteria. There is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

15.7 Radioactive Release From a Subsystem or Component

15.7.1 Introduction

Analyses focused on radioactive release from a subsystem or component address AOOs and accidents that could result in a release of radioactive material to the environment. Radioactive releases can be caused by the following:

- Gas waste management system leak or failure
- Liquid waste management system leak or failure (atmospheric release)
- Release of radioactivity to the environment via liquid pathways
- Fuel handling accident
- Spent fuel cask drop accident

[2] In the WLS COL FSAR, the applicant uses the phrases "site-specific χ/Q values" and χ/Q "site characteristics" interchangeably. In this SER, the staff opts to use the term "site characteristics" because it is defined in 10 CFR Part 52. However, no distinction between the two terms is implied.

15.7.2 Summary of Application

Section 15.7 of the WLS COL FSAR, Revision 11, incorporates by reference Section 15.7 of the AP1000 DCD, Revision 19. In addition, in WLS COL FSAR Section 15.7, the applicant provided the following:

Departures

- WLS DEP 6.4-1

The applicant provided additional information in Section 15.7.4 of the WLS COL FSAR about WLS DEP 6.4-1 related to design changes affecting habitability of the main control room and changes to the DBA radiological consequences analyses, including calculated doses to control room operators and offsite. This information, as well as related WLS DEP 6.4-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.2 of this SER.

AP1000 COL Information Item

- WLS COL 15.7-1

The applicant provided additional information in WLS COL 15.7-1 to address COL Information Item 15.7-1, "Consequences of Tank Failures." This COL item is addressed by the applicant in WLS COL FSAR Section 2.4.13.

15.7.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the radioactive release from a subsystem or component are given in Section 11.2 of NUREG-0800, Branch Technical Position (BTP) 11-6, and Section 2.4.13 of NUREG-0800, Acceptance Criterion Number 5.

The regulatory basis for acceptance of the supplementary information on consequences of a tank failure is established in:

- 10 CFR Part 20, "Standards for protection against radiation," Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage"

- 10 CFR 20.1301, "Dose limits for individual members of the public"

- 10 CFR 20.1406, "Minimization of contamination"

- 10 CFR Part 50, "Domestic licensing of production and utilization facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criteria (GDC) 60, "Control of Releases of Radioactive Materials to the Environment," and GDC 61, "Fuel Storage and Handling and Radioactivity Control"

- 10 CFR 50.34a, "Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors"
- 10 CFR 50.36a, "Technical specifications on effluents from nuclear power reactors"
- 10 CFR 52.80(a), "Contents of applications; additional technical information"
- RG 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning"
- RG 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1
- RG 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," Revision 1
- RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," Revision 2, Regulatory Position C.1.1

15.7.4 Technical Evaluation

The NRC staff reviewed Section 15.7 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the radioactive release from a subsystem or component. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the WLS COL FSAR:

AP1000 COL Information Item

- WLS COL 15.7-1

COL Information Item 15.7-1 states:

> Combined License applicant referencing the AP1000 certified design will perform an analysis of the consequences of potential release of radioactivity to the environment due to a liquid tank failure as outlined in subsection 15.7.3.

The applicant addresses the consequence of a liquid waste tank failure in WLS COL FSAR Section 2.4.13. The staff's evaluation of liquid waste tank failure is described in Section 11.2, "Liquid Waste Management Systems," of this SER.

15.7.5 Post Combined License Activities

There are no post-COL activities related to this section.

15.7.6 Conclusion

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to radioactive release from a subsystem or component, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the regulatory guidance in Sections 2.4.13 and 11.2 of NUREG-0800. The staff based its conclusion on the following:

- WLS DEP 6.4-1, related to design changes affecting habitability of the main control room and changes to the calculated doses to control room operators, is reviewed and found acceptable by the staff in Section 21.2 of this SER.

- WLS COL 15.7-1 is acceptable based on the evaluations in Sections 2.4.13 and 11.2 of this SER.

15.8 <u>Anticipated Transients Without Scram</u>

Analyses focused on anticipated transients without scram (ATWS) address an AOO during which an automatic reactor scram is required but fails to occur due to a common mode fault in the reactor protection system.

Section 15.8 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 15.8, "Anticipated Transients Without Scram," of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Appendix 15A Evaluation Models and Parameters for Analysis of Radiological Consequences of Accidents

15A.1 Introduction

This appendix includes the parameters and models that form the basis of the radiological consequences analyses for the various postulated accidents.

15A.2 Summary of Application

In the WLS COL FSAR, Revision 11, Chapter 15, "Accident Analyses," the applicant incorporated by reference Appendix 15A to Chapter 15, "Accident Analyses," of the AP1000 DCD, Revision 19.

In addition, the applicant provided the following:

Departures

- WLS DEP 6.4-1

The applicant provided additional information in Appendix 15A of the WLS COL FSAR about WLS DEP 6.4-1 related to design changes affecting habitability of the main control room and changes to the DBA radiological consequences analyses, including calculated doses to control room operators and offsite. WLS DEP 6.4-1 revises the analysis of the radiological consequences described in this section of the SER. This information, as well as related WLS DEP 6.4-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.2 of this SER.

AP1000 COL Information Item

- WLS COL 2.3-4

In WLS COL FSAR Sections 15.6 and 15A, the applicant provided additional information in WLS COL 2.3-4 on site characteristic χ/Q values to partially resolve COL Information Item 2.3-4. The applicant provided additional information in WLS COL FSAR Section 2.3.4 to resolve the remaining portion of COL Information Item 2.3-4, and the staff's review of this portion is in Section 2.3.4 of this SER.

15A.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the accident analyses are given in Section 15.0.3 of NUREG-0800.

Requirements for the technical information in the FSAR are given in 10 CFR 52.79. In particular, 10 CFR 52.79(a)(1)(vi) requires a description and safety assessment of the site on which the facility is to be located, including an evaluation of the offsite radiological consequences of postulated accidents to show that the site characteristics comply with the following offsite radiological consequence evaluation factors:

(A) An individual located at any point on the exclusion area boundary (EAB) for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sievert (Sv) (25 roentgen equivalent man (rem)) total effective dose equivalent (TEDE), and

(B) An individual located at any point on the outer boundary of the low population zone (LPZ), who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 0.25 Sv (25 rem) TEDE.

Applications for DCs must include similar evaluations to show compliance with 10 CFR 52.47(a)(2), which includes the same offsite radiological consequence evaluation factors as given in 10 CFR 52.79(a)(1). In other words, both the AP1000 DCD and the COL FSAR must have DBA radiological consequences analyses that estimate a dose at or below 0.25 Sv (25 rem) TEDE at the EAB and LPZ receptors.

Compliance with the control room habitability dose requirements of 10 CFR Part 50, Appendix A, GDC 19, "Control Room," requires that the applicant show that, for a plant located at the WLS site, the control room provides adequate radiation protection to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) TEDE to permit access and occupancy of the control room under accident conditions for the duration of the accident.

15A.4 Technical Evaluation

The NRC staff reviewed Appendix 15A to Chapter 15 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to radiological consequences of accidents. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the WLS COL FSAR:

Departures

- WLS DEP 6.4-1

The applicant provided additional information in Appendix 15A of the WLS COL FSAR about WLS DEP 6.4-1 related to design changes affecting habitability of the main control room and changes to the DBA radiological consequences analyses, including calculated doses to control room operators and offsite. This information revises the analysis of the radiological consequences described in this section of the SER and is reviewed in Section 21.2 of this SER.

WLS DEP 6.4-1 is based on revised DBA radiological consequence analyses that make changes to specific parameters and methodologies that were used in the DBA radiological consequence analyses discussed in AP1000 DCD Chapter 15. The remainder of the analysis assumptions, inputs, and methodologies are the same as given in AP1000 DCD that the staff previously evaluated and found acceptable in NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," Initial Report, Section 15.3.

AP1000 COL Information Item

- WLS COL 2.3-4

In WLS COL FSAR Sections 15.6 and 15A, the applicant stated that it provided additional information in WLS COL 2.3-4 to partially resolve COL Information Item 2.3-4, which states:

> Combined License applicants referencing the AP1000 certified design will address the site-specific χ/Q values specified in [DCD] subsection 2.3.4. For a site selected that exceeds the bounding χ/Q values, the Combined License applicant will address how the radiological consequences associated with the controlling design basis accident continue to meet the dose reference values given in 10 CFR Part 50.34 and control room operator dose limits given in General Design Criteria 19 using site-specific χ/Q values. The Combined License applicant should consider topographical characteristics in the vicinity of the site for restrictions of horizontal and/or vertical plume spread, channeling or other changes in airflow trajectories, and other unusual conditions affecting atmospheric transport and diffusion between the source and receptors. No further action is required for sites within the bounds of the site parameters for atmospheric dispersion.
>
> With regard to assessment of the postulated impact of an accident on the environment, the COL applicant will provide χ/Q values for each cumulative frequency distribution which exceeds the median value (50 percent of the time).

The commitment was also captured as COL Action Items 2.3.4-1, 2.3.4-2, and 2.3.4-3 in Appendix F of NUREG-1793, which states:

> The COL applicant will determine the site specific χ/Q values. If the site-specific values exceed the bounding χ/Q values, the COL applicant will address how the radiological consequences associated with the controlling DBA continue to meet the radiological dose consequence criteria given in Title 10, Section 50.34(a)(1)(ii)(D)*(1) and (2)*, of the *Code of Federal Regulations* (10 CFR 50.34), using site-specific χ/Q values.
>
> The COL applicant will determine the site specific χ/Q values. If the site-specific values exceed the bounding χ/Q values, the COL applicant will address how the radiological consequences associated with the controlling DBA continue to meet the control room operator dose limits given in General Design Criteria 19, using site-specific χ/Q values.
>
> The COL applicant will provide χ/Q values for each cumulative frequency distribution that exceeds the median value (50 percent of the time).

WLS COL 2.3-4 added text to the end of Section 15.6.5.3.7.3 and Section 15A.3.3 of the AP1000 DCD to state that the site-specific atmospheric dispersion (χ/Q) values provided in WLS COL FSAR Section 2.3 are bounded by the values given in AP1000 DCD Table 15A-5, "Offsite Atmospheric Dispersion Factors (χ/Q) For Accident Dose Analysis," (offsite receptors) and Table 15A-6, "Control Room Atmospheric Dispersion Factors (χ/Q) For Accident Dose Analysis" (control room receptors).

The NRC staff reviewed the impact of the site characteristic χ/Q values given in response to WLS COL 2.3-4 on the radiological consequences of DBAs. The applicant did not provide site-specific doses at the EAB, LPZ, or control room for the DBAs referenced in AP1000 DCD, Chapter 15, but instead incorporated by reference the analysis of the radiological consequences in AP1000 DCD, Chapter 15.

AP1000 DCD, Chapter 15, over several sections, describes and provides results of the radiological consequences analyses for the DBAs applicable to the AP1000 design. A list of the DBAs analyzed for radiological consequences and the corresponding sections where the radiological consequences analyses for those DBAs are discussed in the AP1000 DCD is given below.

DCD Section	Design Basis Accident
15.1.5.4	Main Steam Line Break
15.3.3.3	Reactor Coolant Pump Shaft Seizure (Locked Rotor)
15.4.8.3	Control Rod Ejection
15.6.2	Small Line Break
15.6.3.3	Steam Generator Tube Rupture
15.6.5.3	Loss of Coolant Accident (LOCA)
15.7.4.3	Fuel Handling Accident

The DBA radiological consequences analyses in the AP1000 DCD were based, in part, on site parameter atmospheric dispersion values (i.e., χ/Q values). These site parameter χ/Q values are the only postulated environmental feature of an assumed site that are used in DBA radiological consequence analyses. The AP1000 site parameter χ/Q values used in the DBA radiological consequence analyses were selected to bound 70 to 80 percent of U.S. sites. As a result, 70 to 80 percent of U.S. sites would be expected to have site characteristic χ/Q values that are less than the AP1000 site parameter χ/Q values, and would, therefore, have corresponding DBA radiological consequences that are lower than those described in the AP1000 DCD. To resolve WLS COL 2.3-4, the applicant discussed the WLS site characteristic short-term (accident) χ/Q values in WLS COL FSAR Section 2.3.4. The WLS site characteristic EAB and LPZ χ/Q values for DBAs are given in WLS COL FSAR Table 2.0-201, and the control room χ/Q values for DBAs are given in WLS COL FSAR Table 2.0-202. In Section 2.3.4 of this SER, the NRC staff discusses its review of the WLS site characteristic χ/Q values and resolution to WLS COL 2.3-4.

As described in Section 2.3.4 of this SER, the WLS site characteristic χ/Q values for each time averaging period are less than the corresponding AP1000 site parameter χ/Q values. This causes the postulated radiological consequences of DBAs at the WLS site to be lower than those reported in the AP1000 DCD. Therefore, since the offsite radiological consequence requirements of 10 CFR 50.34(a)(1)(ii)(D)(1) and (2)[3] and the DBA control room radiological consequence criteria in GDC 19 are met for the AP1000, then these same requirements are also met by the applicant for the WLS site.

The effect of the site-specific χ/Q values on the Technical Support Center radiological habitability is evaluated by the NRC staff in SER Section 13.3 as part of its evaluation of WLS DEP 18.8-1.

Although WLS DEP 6.4-1 is a site-specific departure from the AP1000 DCD, the revised DBA dose analyses provided by the applicant are generic analyses in that they use the same short-term (accident) atmospheric dispersion factor (χ/Q) values given as site parameters in AP1000 DCD, Section 2.3.4. For WLS DEP 6.4-1, no changes were made to the WLS site characteristic short-term χ/Qs given in FSAR 2.3.4; therefore, in accordance with the discussion of WLS COL 2.3-4 above, the WLS site-specific short-term χ/Q values are less than those used in the revised generic analysis supporting WLS DEP 6.4-1. By the same logic above, the WLS site-specific estimated total dose at the EAB, LPZ, and the MCR for each DBA is, therefore, less than the generic revised estimated total dose at the same receptor location for each DBA, as provided in the additional FSAR information for WLS DEP 6.4-1.

15A.5 Post Combined License Activities

There are no post-COL activities related to this section.

15A.6 Conclusion

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to the evaluation models and parameters for analysis of radiological consequences of accidents, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the requirements of 10 CFR 52.79(a)(1) and 10 CFR Part 50, Appendix A, GDC 19. The staff based its conclusion on the following:

- WLS DEP 6.4-1 provides additional information related to design changes affecting habitability of the main control room and changes to the DBA radiological consequences analyses, including calculated doses to control room operators and offsite. This information revises the analysis of the radiological consequences described in this section of the SER and is reviewed and found acceptable by the staff in Section 21.2 of this SER.

- WLS COL 2.3-4 is acceptable because the DBA offsite radiological consequences meet the requirements of 10 CFR 52.79(a)(1) and the DBA control room radiological consequences meet the requirements of GDC 19.

[3] The radiological consequence criteria in 50.34(a)(1)(ii)(D)(1) and (2) are the same criteria for DC applications in 10 CFR 52.47(a)(2)(iv) and for COL applicants in 10 CFR 52.79(a)(1)(vi)

Appendix 15B **<u>Removal of Airborne Activity from the Containment Atmosphere Following a LOCA</u>**

This appendix includes information related to the AP1000 design, which does not depend on active systems to remove airborne particulates or elemental iodine from the containment atmosphere following a postulated LOCA with core melt. The AP1000 applicant stated that naturally occurring passive removal processes provide significant removal capability such that airborne elemental iodine is reduced to very low levels within a few hours and the airborne particulates are reduced to extremely low levels within 12 hours.

Appendix 15B of the WLS COL FSAR, Revision 11, incorporates by reference, Appendix 15B, “Removal of Airborne Activity from the Containment Atmosphere Following a LOCA,” of Revision 19 of the AP1000 DCD. In addition, in the WLS COL FSAR, the applicant provided the following:

<u>Departures</u>

- WLS DEP 6.4-1

The applicant provided additional information in Appendix 15B of the WLS COL FSAR about WLS DEP 6.4-1 related to design changes affecting habitability of the main control room and changes to the calculated doses to control room operators. This information, as well as related WLS DEP 6.4-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.2 of this SER.

The NRC staff reviewed Appendix 15B of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this section. The NRC staff’s review confirmed that the applicant addressed the required information to satisfy the evaluation criteria. There is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Table 15.0-1. Power Calorimetric Uncertainty Methodology

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. The plant calorimetric uncertainty and plant instrumentation performance is bounded by the 1 percent calorimetric uncertainty value assumed for the initial reactor power in the safety analysis.	Inspection will be performed of the plant operating instrumentation installed for feedwater flow measurement, its associated power calorimetric uncertainty calculation, and the calculated calorimetric values.	a) the as-built system takes input for feedwater flow measurement from a Caldon [Cameron] LEFM CheckPlus™ System; b) the power calorimetric uncertainty calculation documented for that instrumentation is based on an NRC-accepted Westinghouse methodology and the uncertainty values for that instrumentation are not lower than those for the actual installed instrumentation; and c) the calculated calorimetric power uncertainty measure values are bounded by the 1 percent uncertainty value assumed for the initial reactor power in the safety analysis.

16.0 TECHNICAL SPECIFICATIONS

This chapter discusses the plant-specific technical specifications (PTS), as well as the design reliability assurance program (D-RAP) and the controls for systems, structures, and components (SSCs) required for defense-in-depth in accordance with the program for regulatory treatment of nonsafety systems (RTNSS).

16.1 Technical Specifications

16.1.1 Introduction

Section 16.1, "Technical Specifications," of the William States Lee III Nuclear Station (WLS) combined license (COL) Final Safety Analysis Report (FSAR), and the WLS COL Part 4, "Technical Specifications," provide the PTS for WLS Units 1 and 2, in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36, "Technical specifications," and 10 CFR 52.79(a)(30). Technical Specifications (TS) impose limits, operating conditions, and other requirements upon reactor facility operation for the public health and safety. The TS are derived from the analyses and evaluations in the safety analysis report. In general, TS must include: (1) safety limits and limiting safety system settings; (2) limiting conditions for operation (LCO); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. The PTS are derived from the analyses and evaluations in the AP1000 Design Control Document (DCD) and the WLS COL FSAR.

As part of the regulatory standardization effort, the U.S. Nuclear Regulatory Commission (NRC) staff has prepared standard technical specifications (STS) for each of the light-water reactor nuclear steam supply systems and associated balance-of-plant equipment systems. In 1992, the NRC issued the STS to clarify the content and format of requirements necessary to ensure safe operation of nuclear power plants. The STS for Westinghouse pressurized water reactors are included in NUREG-1431, "Standard Technical Specifications - Westinghouse Plants." Volume 1 addresses the STS, and Volume 2 addresses the associated STS Bases. The STS include bases for safety limits, limiting safety system settings, LCO, and associated action and surveillance requirements. Major revisions to the STS were published in 1995 (Revision 1), 2001 (Revision 2), and 2004 (Revision 3).

The format and content of the PTS and Bases for a COL referencing a certified design should be based on the generic TS (GTS) and Bases for that design. For a COL application that references a certified design, the proposed PTS and Bases may include appropriate plant-specific departures from the referenced GTS and Bases when warranted. These departures, if included with the COL application, need to be justified to demonstrate that the requirements of 10 CFR 50.36 are met.

16.1.2 Summary of Application

Section 16.1 of the WLS COL FSAR, Revision 11, incorporates by reference Sections 16.1.1 and 16.1.2 of the AP1000 DCD, Revision 19. Part 4 of the WLS COL incorporates by reference the AP1000 GTS and Bases in Section 16.1 of the DCD. In accordance with Section IV(A)(2)(c) of Appendix D, "Design Certification Rule for the AP1000 Design" to 10 CFR Part 52, "Licenses, certifications, and approvals for nuclear power plants," the applicant's PTS consist of the

AP1000 GTS and site-specific information. The applicant took departures from the AP1000 GTS.

The AP1000 GTS includes items that a COL applicant must satisfy in order to complete a particular GTS provision. Detailed design information, equipment selection, instrumentation settings, and other information not available at the time of design certification (DC) are needed to establish the values or information to be included in the PTS. Locations for the addition of this information are signified in the GTS by square brackets [ ] or reviewer's notes to indicate that the COL applicant must provide plant-specific values or alternate text.

In WLS COL application Part 4, the applicant provided the following:

Departures

- WLS DEP 3.2-1

The applicant provided additional information about WLS DEP 3.2-1 in WLS COL Part 4, including changes to TS SR 3.5.4.7 and corresponding Bases, Bases B3.3.3 (LCO Section), and Bases B3.5.4 (Background Section), related to design modifications to the condensate return portion of the Passive Core Cooling System. This information, as well as related WLS DEP 3.2-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.1 of this SER.

- WLS DEP 6.4-1

The applicant provided additional information about WLS DEP 6.4-1 in WLS COL Part 4, including changes to TS LCO 3.7.4, TS SR 3.7.4.1, and Bases 3.4.10, 3.7.4, and 3.7.6 related to design changes affecting habitability of the main control room and changes to the calculated doses to control room operators. This information, as well as related WLS DEP 6.4-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.2 of this SER.

- WLS DEP 6.4-2

The applicant provided additional information about WLS DEP 6.4-2 in WLS COL Part 4, including changes to TS 3.3.2 and corresponding Bases and TS 3.7.6 and corresponding Bases related to design changes affecting how the temperature and humidity in the main control room are maintained within the limits for reliable human performance. This information, as well as related WLS DEP 6.4-2 information appearing in other chapters of the FSAR, is reviewed in Section 21.3 of this SER.

- WLS DEP 7.3-1

The applicant provided additional information about WLS DEP 7.3-1 in WLS COL Part 4, including changes to TS Table 3.3.2-1 and associated Bases, related to required design changes for the PMS source range neutron flux doubling logic to comply with the requirements of IEEE Std. 603-1991, Clause 6.6. This information, as well as related WLS DEP 7.3-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.5 of this SER.

<u>AP1000 COL Information Item</u>

- WLS COL 16.1-1

The applicant provided additional information in WLS COL 16.1-1 to resolve COL Information Item 16.1-1 (COL Action Item 16.2-1). The applicant provided additional information to address each of the remaining brackets [ ] and reviewer's notes in the AP1000 GTS.

The following sections of the WLS PTS and Bases include information that the applicant addressed as part of COL Information Item 16.1-1:

- PTS 3.3.1, 3.3.2, and 3.6.4
- PTS 4.1, 4.1.1, and 4.1.2
- PTS 5.1.1, 5.1.2, 5.2.1.a, 5.2.1.b, 5.2.2, 5.3, 5.3.1, 5.6.1, and 5.6.2

16.1.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793, "Final Safety Evaluation Report [FSER] Related to Certification of the AP1000 Standard Design," and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for TS and Bases reviews are given in Section 16 of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition." Areas of review that interface with other sections of the SRP can also be found in Section 16 of NUREG-0800.

The applicable regulatory requirements for the information being reviewed in this section are:

- 10 CFR 50.36, "Technical Specifications."
- 10 CFR 50.36a, "Technical specifications on effluents from nuclear power reactors."
- 10 CFR 52.79(a)(30), "Contents of applications."

16.1.4 Technical Evaluation

The NRC staff reviewed Section 16.1 of the WLS COL FSAR and Part 4 of the WLS COL application, and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic[1]. The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the TS. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

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[1] See Section 1.2.2 for a discussion of the staff's review related to verification of the scope of information to be included in a COL application that references a DC.

Section 1.2.3 of this safety evaluation report (SER) provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (Vogtle Electric Generating Plant [VEGP], Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from requests for additional information (RAIs).

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the Bellefonte Nuclear Plant (BLN), Units 3 and 4 COL application.

Many VEGP SER section numbers were changed from those used in the BLN SER to more closely follow the PTS numbering. Therefore, the corresponding BLN SER section numbers are frequently identified when quoting standard content material from the SER for the reference COL application (VEGP).

The staff reviewed the information in the WLS COL FSAR and the WLS COL application, Part 4:

<u>*AP1000 COL Information Item*</u>

- WLS COL 16.1-1

The following portion of this technical evaluation section is reproduced from Section 16.1.4 of the VEGP SER:

> *The following portion of this technical evaluation section is reproduced from Section 16.1.4 of the BLN SER:*
>
> *In Section 16.1.1 of the BLN COL FSAR, the applicant provided additional information in BLN COL 16.1-1 to resolve COL Information Item 16.1-1 (COL Action Item 16.2-1) listed under the Section 16.1.1 header, "Combined License Information," of the AP1000 DCD, Revision 17, which states:*
>
> > *This set of technical specifications is intended to be used as a guide in the development of the plant-specific technical*

> *specifications. The preliminary information originally provided in brackets [ ] has been revised with the updated information APP-GW-GLR-064 and APP-GW-GLN-075. Combined License applicants referencing the AP1000 will be required to provide the final information for the remaining brackets [ ] with final plant-specific information.*

In Section 16.1 of the BLN COL FSAR, the applicant noted that the GTS and Bases provided with Chapter 16 of the AP1000 DCD are incorporated by reference into the PTS provided in Part 4 of the BLN COL application.

The staff evaluated the applicant's disposition of each of the remaining bracketed information items in the respective TS sections listed below.

The staff did not review portions of the BLN PTS and Bases that were identical to the AP1000 GTS and Bases. The technical evaluation for those portions that are identical to the AP1000 GTS and Bases can be found in the NRC staff's FSER for the AP1000 DCD.

16.1.4.1 <u>Use and Application</u>

Section 1.0 of the BLN PTS includes definitions of terms used in the context of plant TS, and examples to illustrate the applications of logical connectors, completion times for required actions, and frequencies for surveillance requirements (SRs). Section 1.0 of the BLN PTS is identical to the AP1000 GTS. There is no site-specific information that the applicant needed to provide to complete this section.

16.1.4.2 <u>Safety Limits</u>

Section 2.0 of the BLN PTS and Bases include[s] requirements for safety limits to ensure that the fuel design limits are not exceeded during steady state conditions, normal operational transients, and anticipated operational occurrence. Section 2.0 of the BLN PTS and Bases are [is] identical to the AP1000 GTS and Bases. There is no site-specific information that the applicant needed to provide to complete this section.

16.1.4.3.0 Limiting Condition for Operation and Surveillance Requirement Applicability

The following portion of this technical evaluation section is reproduced from Section 16.1.4.3 of the BLN SER:

Section 3.0 of the BLN PTS and Bases include[s] general provisions regarding determination of equipment operability and performance of SRs in specific TS sections (i.e., TS 3.1 through TS 3.9). Section 3.0 of the BLN PTS and Bases are [is] identical to the AP1000 GTS and Bases. There is no site-specific information that the applicant needed to provide to complete this section.

16.1.4.3.1 Reactivity Control Systems

The following portion of this technical evaluation section is reproduced from Section 16.1.4.4 of the BLN SER:

Section 3.1 of the BLN PTS and Bases include[s] requirements for the reactivity control systems which are designed to reliably control reactivity changes, and under postulated accident conditions, ensure that the capability to cool the core is maintained. Section 3.1 of the BLN PTS and Bases are [is] identical to the AP1000 GTS and Bases. There is no site-specific information that the applicant needed to provide to complete this section.

16.1.4.3.2 Power Distribution Limits

The following portion of this technical evaluation section is reproduced from Section 16.1.4.5 of the BLN SER:

Section 3.2 of the BLN PTS and Bases include[s] requirements for the reactor core power distribution limits which are designed to reliably control core thermal limits and core power distribution consistent with the design safety analysis. Section 3.2 of the BLN PTS and Bases are [is] identical to the AP1000 GTS and Bases. There is no site-specific information that the applicant needed to provide to complete this section.

16.1.4.3.3 Instrumentation

The following portion of this technical evaluation section is reproduced from Section 16.1.4.6 of the BLN SER:

Section 3.3 of the BLN PTS and Bases include[s] requirements for the instrumentation systems that display information required to protect against violating core fuel design limits and Reactor Coolant System (RCS) integrity, and to mitigate accidents.

The BLN instrumentation will be selected after COL issuance, and therefore, in accordance with COL/DC-ISG-8, "Necessary Content of Plant-Specific Technical Specifications When a Combined License is Issued," all trip setpoints and allowable values must be established through a staff-approved administrative control TS that specifies use of an NRC-approved methodology for determining the trip setpoints and allowable values, and a document controlled by 10 CFR 50.59 for recording this information. The trip setpoints and allowable *values, referred to in Tables 3.3.1-1 and 3.3.2-1, will be determined after selection of specific instrumentation.*

Request for additional information (RAI) 16-1 was issued in accordance with COL/DC-ISG-8, and requested that the applicant identify the method of determining the trip setpoints and allowable values, as well as establish an associated document in which to record the site-specific values and other restrictions necessary to satisfy 10 CFR 50.36. The applicant should clarify that after selection of specific instrumentation, the trip setpoints and allowable values, referred to in Tables 3.3.1-1 and 3.3.2-1, will be calculated using the setpoint control program that specifies the approved methodology (i.e., WCAP-16361, APP-PMS-JEP-001, Revision 0, May 2006, "Westinghouse Setpoint Methodology for Protection Systems – AP1000"). In addition, the applicant should propose a setpoint control program to be added in the Administrative Control section of the TS, as stated in COL/DC-ISG-8. ***This is identified as Open Item 16.1-1****.*

<u>Resolution of Standard Content Open Item 16.1-1</u>

Resolution to this issue was brought forward at a public meeting on September 3, 2009, attended by the staff, Westinghouse, and the AP1000 COL applicants. Westinghouse committed to provide an acceptable setpoint control program in the AP1000 DC amendment application, which would then be adoptable by any COL applicants. This program was submitted to the staff in a letter dated February 19, 2010, and revised on May 6, 2010. The review of this program is documented in a supplement to NUREG-1793.

The applicant, in its May 21, 2010, supplemental response to this open item, committed to calculate trip setpoints and allowable values using the approved methodology cited above and to incorporate the AP1000 DCD setpoint control program in the Administrative Controls section of its PTS. The staff finds this response acceptable, since it ensures the applicant will use approved methodologies and a comprehensive administrative program to calculate setpoint values. The incorporation of this program into the VEGP TS in a later revision is ***Confirmatory Item 16.1-1.***

<u>Resolution of Standard Content Confirmatory Item 16.1-1</u>

Confirmatory Item 16.1-1 is an applicant commitment to revise its PTS to incorporate the AP1000 DCD setpoint control program in the Administrative Controls section of its PTS. The staff verified that the PTS was appropriately revised. As a result, Confirmatory Item 16.1-1 is now closed. [The Administrative controls section of the WLS PTS cites Revision 1 of WCAP-16361, consistent with the GTS in the AP1000 certified design.]

16.1.4.3.4 Reactor Coolant System

The following portion of this technical evaluation section is reproduced from Section 16.1.4.7 of the BLN SER:

Section 3.4 of the BLN PTS and Bases include[s] requirements for various RCS parameters (i.e., pressure, temperature, flow, etc.) and subsystems (i.e., RCS loops, pressurizer, low-temperature overpressure protection, etc.) to ensure the fuel integrity and the RCPB [reactor coolant pressure boundary] integrity are preserved during all modes of plant operation. Section 3.4 of the BLN PTS and Bases are [is] identical to the AP1000 GTS and Bases. There is no site-specific information that the applicant needed to provide to complete this section.

16.1.4.3.5 Emergency Core Cooling Systems

The following portion of this technical evaluation section is reproduced from Section 16.1.4.8 of the BLN SER:

Section 3.5 of the BLN PTS and Bases include[s] requirements for the safety-related passive core cooling system, which is designed to perform emergency core decay heat removal, RCS emergency makeup and boration, and safety injection. Section 3.5 of the BLN PTS and Bases are [is] identical to the AP1000 GTS and Bases. There is no site-specific information that the applicant needed to provide to complete this section.

16.1.4.3.6 Containment Systems

The following portion of this technical evaluation section is reproduced from Section 16.1.4.9 of the BLN SER:

Section 3.6 of the BLN PTS and Bases include[s] requirements for the containment systems, which are designed to shield [contain] fission products that may be in the containment atmosphere following accident conditions. Section 3.6 of the BLN PTS and Bases are [is] identical to the AP1000 GTS and Bases, except for the deletion of a reviewer's note. For TS 3.6.4, the reviewer's note is not applicable to the PTS, and the applicant has appropriately removed the information. This is acceptable to the staff. There is no site-specific information that the applicant needed to provide to complete this section.

16.1.4.3.7 Plant Systems

The following portion of this technical evaluation section is reproduced from Section 16.1.4.10 of the BLN SER:

Section 3.7 of the BLN PTS and Bases include[s] requirements for various systems in the secondary side of the steam generators (i.e., the main steam safety valves, the main steam isolation valves, the main feedwater isolation valves, etc.), the spent fuel pool water level and makeup systems, and the main *control room habitability system. Section 3.7 of the BLN PTS and Bases are [is] identical to the AP1000 GTS and Bases. There is no site-specific information that the applicant needed to provide to complete this section.*

16.1.4.3.8 Electrical Power Systems

The following portion of this technical evaluation section is reproduced from Section 16.1.4.11 of the BLN SER:

Section 3.8 of the BLN PTS and Bases include[s] requirements for the plant electrical systems that provide redundant, diverse and dependable power sources for all plant operating conditions. In the event of a total loss of off-site power, batteries and back-up on-site diesel generators are provided to supply electrical power equipment necessary for the safe shutdown of the plant. Section 3.8 of the BLN PTS and Bases are [is] identical to the AP1000 GTS and Bases. There is no site-specific information that the applicant needed to provide to complete this section.

16.1.4.3.9 Refueling Operations

The following portion of this technical evaluation section is reproduced from Section 16.1.4.12 of the BLN SER:

Section 3.9 of the BLN PTS and Bases include[s] requirements for boron concentration, unborated water sources, nuclear instrumentation, containment penetrations, and water inventory in the refueling pool during Mode 6. Section 3.9 of the BLN PTS and Bases are [is] identical to the AP1000 GTS and Bases. There is no site-specific information that the applicant needed to provide to complete this section.

16.1.4.4 *Design Features*

Section 4.0 of the WLS PTS includes other design features not covered elsewhere in the PTS such as the site location, the site maps, and other information related to core design and fuel storage design. Section 4.0 of the WLS PTS is identical to the AP1000 GTS except for site-specific information provided by the applicant. In Section 4.1, the applicant provided the WLS site location information to replace the bracketed information in the GTS. The staff found the added information acceptable since it is consistent with related information found in FSAR Section 2.1.1, and in accordance with guidance provided in the GTS. In Section 4.1.1, the applicant provided Figure 4.1-2, which describes its site boundary and exclusion area boundaries. The staff found the added information acceptable since it is consistent with related information found in WLS COL FSAR Sections 2.1.1.2 and 2.1.1.3, and in accordance with the guidance provided in the GTS. In Section 4.1.2, the applicant also provided the site location in Figure 4.1-1 and a description of the radius, which establishes its low population zone. The staff found the added information acceptable since it is consistent with related information found in WLS COL FSAR Section 2.1.3, and is in accordance with the guidance provided in the GTS.

The following portion of this technical evaluation section is reproduced from Section 16.1.4.5 of the VEGP SER:

16.1.4.5 Administrative Controls

The following portion of this technical evaluation section is reproduced from Section 16.1.4.14 of the BLN SER:

This section of the BLN PTS includes provisions, which address various administrative controls related to plant key personnel responsibilities, plant procedures, special programs and reports, etc., to ensure the plant is safely operated. As discussed in Section 16.1.4.6 above, [WLS SER Section 16.1.4.3.3,] *the BLN instrumentation will be selected after COL issuance, and therefore, in accordance with COL/DC-ISG-8, all trip setpoints and allowable values must be established through a staff-approved administrative control TS that specifies use of an NRC-approved methodology for determining the trip setpoints and allowable values, and a document controlled by 10 CFR 50.59 for recording this information. The trip setpoints and allowable values, referred to in Tables 3.3.1-1 and 3.3.2-1, will be determined after selection of specific instrumentation.*

The staff issued RAI 16-1 and requested that the applicant identify the method of determining the trip setpoints and allowable values, as well as establish an associated document in which to record the site-specific values and other restrictions necessary to satisfy 10 CFR 50.36. The applicant should clarify that after selection of specific instrumentation, the trip setpoints and allowable values, referred to in Tables 3.3.1-1 and 3.3.2-1, will be calculated using the setpoint control program that specifies the approved methodology (i.e., WCAP-16361, APP-PMS-JEP-001, Revision 0, May 2006, "Westinghouse Setpoint Methodology for Protection Systems – AP1000"). In addition, the applicant should propose a setpoint control program to be added in the Administrative Control section of the TS, as stipulated in COL/DC-ISG-8. ***This is identified as Open Item 16.1-1****.*

Resolution of Standard Content Open Item 16.1-1

The resolution of this issue is discussed in the evaluation of Section 16.1.4.3.3, "Instrumentation," above. The applicant committed to adopting the setpoint control program approved in the AP1000 DC, which will be verified in a future revision of the VEGP TS. This is ***Confirmatory Item 16.1-1****.*

Resolution of Confirmatory Item 16.1-1

Confirmatory Item 16.1-1 is an applicant commitment to revise its PTS to incorporate the AP1000 DCD setpoint control program in the Administrative Controls section of its PTS. The staff verified that the PTS was appropriately revised. As a result, Confirmatory Item 16.1-1 is now closed. [The

Administrative controls section of the WLS PTS cites Revision 1 of WCAP-16361, consistent with the GTS in the AP1000 certified design.]

The following portion of this technical evaluation section is reproduced from Section 16.1.4.14 of the BLN SER:

In Section 5.3.1 of the BLN PTS, the applicant replaced the GTS bracketed information, clarifying that each member of the unit staff shall meet or exceed minimum qualifications of RG [Regulatory Guide] *1.8, Revision 3 except for during cold license operator training where portions of RG 1.8, Revision 2 will apply. The staff finds this acceptable because RG 1.8, Revision 3 does not address cold license operator training. In other respects, Sections 5.0, 5.1.1, 5.1.2, 5.2.1a, 5.2.1b, 5.2.2, 5.3, 5.6.1, and 5.6.2 of the BLN PTS are identical to the AP1000 GTS, except for site-specific information provided by the applicant to replace the bracketed information in the GTS. The site-specific information provided was administrative in nature and the staff found it acceptable.*

In Section 5.2.2 of the VEGP PTS, the applicant proposed to remove the brackets around the COL item related to unit staff organization, as well as removing work hour restrictions in TS 5.2.2.d. The applicant refers to 73 Federal Register (FR) 79923 which provides the NRC's model application for adopting Technical Specification Task Force (TSTF)-511, Revision 0, "Eliminate Working Hour Restrictions from TS 5.2.2 to Support Compliance with 10 CFR Part 26 ["Fitness for Duty Programs"]*." The staff finds this deletion acceptable since it conforms to the guidance provided in the TSTF and working hour restrictions in 10 CFR Part 26, and therefore, is no longer required to be in the TS. This appropriately meets the intent of completing this bracketed information.*

16.1.5 Post Combined License Activities

There are no post-COL activities related to this section.

16.1.6 Conclusion

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to the WLS PTS and Bases, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements. The staff based its conclusion on the following:

- WLS DEP 3.2-1, related to design modifications to the condensate return portion of the Passive Core Cooling System, is reviewed and found acceptable by the staff in Section 21.1 of this SER.

- WLS DEP 6.4-1, related to design changes affecting habitability of the main control room and changes to the calculated doses to control room operators, is reviewed and found acceptable by the staff in Section 21.2 of this SER.

- WLS DEP 6.4-2, related to design changes affecting how the temperature and humidity in the main control room are maintained within the limits for reliable human performance, is reviewed and found acceptable by the staff in Section 21.3 of this SER.

- WLS DEP 7.3-1, related to required design changes for the PMS source range neutron flux doubling logic to comply with the requirements of IEEE Std. 603-1991, Clause 6.6, is reviewed and found acceptable by the staff in Section 21.5 of this SER.

- WLS COL 16.1-1, related to PTS and their Bases, is acceptable because the site-specific information is either identical to the GTS or will be completed using NRC-approved methodologies.

For the reasons set forth above, the staff finds that Section 16.1 of the WLS COL FSAR and Part 4 of the WLS COL application are acceptable and satisfy the requirements of 10 CFR 50.36; 10 CFR 50.36a, "Technical specifications on effluents from nuclear power reactors"; and 10 CFR 52.79(a)(30).

16.2 Design Reliability Assurance Program (Related to RG 1.206, Section C.III.1, Chapter 17, C.I.17.4, "Reliability Assurance Program Guidance")

The D-RAP comprises the reliability assurance activities that assure that the plant is consistent with the certified design when fuel is loaded for the first time.

Section 16.2 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 16.2, "Design Reliability Assurance Program," of Revision 19 of the AP1000 DCD, which in turn refers to Section 17.4 for a description of the program. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The NRC staff's review of the applicant's D-RAP is documented in Section 17.4 of this SER.

16.3 Investment Protection

16.3.1 Introduction

The AP1000 design includes active systems that provide defense-in-depth capabilities (identified as "investment protection" by the applicant) for RCS makeup and decay heat removal. These active systems are the first line of defense in reducing challenges to the passive systems in the event of transients or plant upsets. Most active systems in the AP1000 design are designated as nonsafety-related. Because some active systems reduce challenges to safety-related systems to a significant degree, short-term availability controls are necessary to provide reasonable assurance that these SSCs are operable during anticipated events.

A detailed evaluation of the regulatory treatment of non-safety systems for the AP1000 design, and the concept of investment protection, is addressed in Chapter 22 of NUREG-1793.

16.3.2 Summary of Application

Section 16.3 of the WLS COL FSAR, Revision 11, incorporates by reference Section 16.3 of the AP1000 DCD, Revision 19.

In addition, in WLS COL FSAR Section 16.3, the applicant provided the following:

AP1000 COL Information Item

- STD COL 16.3-1

The applicant provided additional information in Standard (STD) COL 16.3-1 to address COL Information Item 16.3-1. This item is related to the development of a procedure to control the operability of investment protection SSCs.

16.3.3 Regulatory Basis

The regulatory basis of the information incorporated by reference, and the additional information presented in this application, is addressed in NUREG-1793 and its supplements.

16.3.4 Technical Evaluation

The NRC staff reviewed Section 16.3 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to SSCs required for defense-in-depth. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is

identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the BLN Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from Section 16.3.4 of the VEGP SER:

> *AP1000 COL Information Item*
>
> - *STD COL 16.3-1*
>
> *The applicant provided supplemental information by adding the following statement to DCD Section 16.3-1:*
>
> > *Station procedures govern and control the operability of investment protection systems, structures, and components in accordance with Table 16.3-2 of the DCD, and provide the operating staff with instruction for implementing required actions when operability requirements are not met. Procedure development is addressed in FSAR Section 13.5.*
>
> *Section 22.5.9 of the NRC staff's FSER related to the DCD (NUREG-1793) evaluated the short-term availability controls proposed by Westinghouse for important non-safety-related SSCs. The NRC staff concluded that the administrative controls for the SSCs required for defense in depth, listed in Table 16.3-2 of the AP1000 DCD, were acceptable. COL applicants referencing the AP1000 are responsible for developing a procedure to control the operability of these SSCs in accordance with DCD Table 16.3-2 (COL Information Item 16.3.2-1* [16.3-1]*).*
>
> *The applicant's response to STD COL 16.3-1 is acceptable because there were no exceptions taken to the list of SSCs required for defense in depth nor to the administrative procedures included in AP1000 DCD Table 16.3-2. The applicant also committed to place this information in station procedures. The information in DCD Table 16.3-2 also provides the operating staff with instruction for implementing required actions when operability requirements are not met.*

16.3.5 Post Combined License Activities

There are no post-COL activities related to this section.

16.3.6 Conclusion

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information related to defense-in-depth using nonsafety-related SSCs, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the

NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable based on the regulatory basis addressed in NUREG-1793. The staff based its conclusion on the following:

- STD COL 16.3-1, as related to SSCs required for defense-in-depth, is acceptable because it states that station procedures will govern and control the operability of these SSCs, in accordance with Table 16.3-2 of the AP1000 DCD, without exceptions. The information in DCD Table 16.3-2 also provides the operating staff with guidance for taking required actions when operability requirements are not met.

17.0 QUALITY ASSURANCE (RELATED TO RG 1.206, SECTION C.III.1, CHAPTER 17, C.I.17, "QUALITY ASSURANCE AND RELIABILITY ASSURANCE")

The quality assurance (QA) program for design, fabrication, construction, testing, and operation, the design reliability program, and the maintenance rule program are discussed in this chapter.

17.1 Quality Assurance During the Design and Construction Phases

17.1.1 Introduction

The QA program related to design and construction activities is discussed in this section. It addresses the QA program implemented during combined license (COL) application development, including site characterization activities, design and construction phases.

17.1.2 Summary of Application

Section 17.1, "Quality Assurance During the Design and Construction Phases," of the William States Lee III Nuclear Station, Units 1 and 2 (WLS), COL Final Safety Analysis Report (FSAR), Revision 11, incorporates by reference Section 17.1 of the AP1000 Design Control Document (DCD), Revision 19.

In addition, in WLS COL FSAR Section 17.1, the applicant provided the following:

AP1000 COL Information Item

- WLS COL 17.5-1

The applicant provided additional information in WLS COL 17.5-1 to address COL Information Item 17.5-1. In WLS COL 17.5-1, the applicant addresses the quality assurance program under which the COL application was developed for the design and construction phases, which is applicable until COL issuance. Section 17.5, "Quality Assurance Program Description – New License Applicants," of the WLS COL FSAR addresses the QA program for the remaining portion of the design and construction phases following COL issuance, which is described in the Duke Nuclear Plant Development (NPD) Quality Assurance Program Description (QAPD).

17.1.3 Regulatory Basis

The regulatory basis of the information incorporated by reference into WLS COL FSAR Section 17.1 is addressed in NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," and its supplements.

In addition, the relevant requirements of the Commission regulations for the resolution of WLS COL 17.5-1 are established in Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, "Domestic licensing of production and utilization facilities," Appendix B, "Quality assurance criteria for nuclear power plants and fuel reprocessing plants," as required by 10 CFR 52.79(a)(25).

17.1.4 Technical Evaluation

The staff of the Nuclear Regulatory Commission (NRC) reviewed Section 17.1 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to QA during design and construction phases. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this safety evaluation report (SER) provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the design certification (DC) and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (Vogtle Electric Generating Plant, Units 3 and 4 [VEGP]) were equally applicable to the WLS Units 1 and 2 COL application, the NRC staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from requests for additional information (RAIs).

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the Bellefonte Nuclear Plant (BLN), Units 3 and 4 COL application. Any confirmatory items in the standard content material retain the numbers assigned in the VEGP SER.

The staff reviewed the following information in the WLS COL FSAR:

AP1000 COL Information Item

- WLS COL 17.5-1

The NRC staff reviewed the partial resolution of WLS COL 17.5-1 related to QA during the design and construction phases until COL issuance, which is included under Section 17.1 of the WLS COL FSAR. The remaining information for WLS COL 17.5-1 is included in Section 17.5 of the WLS COL FSAR. The staff's review of WLS COL 17.5-1 is a combination of plant-specific evaluation and standard content evaluation.

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[1] See SER Section 1.2.2 for a discussion of the NRC staff's review related to verification of the scope of information to be included in a COL application that references a design certification (DC).

Duke supplemented the information in AP1000 DCD Section 17.1 with new text to address the QA program requirements for design and construction activities implemented from COL application development through operations. Upon review of the additional text provided by the applicant, the NRC staff identified areas where additional information was needed.

In RAI 17.5-8, dated October 6, 2008, the NRC staff noted that WLS COL FSAR Section 17.1 states that the Duke Energy QA program and the Westinghouse Electric Company Quality Management System establish the QA requirements for design activities until the Duke NPD QAPD becomes effective. The staff requested clarification on the expected Duke Energy and Westinghouse Electric Company scope of work related to the applicant's COL application design activities from the time of docketing until the time the COL might be issued.

In its response letter, dated December 11, 2008, the applicant referenced a letter dated February 6, 2008, which provided a clarification to Section 17.1 of the WLS COL FSAR. Enclosure 1 of that letter revised the WLS COL FSAR to remove the reference to the Westinghouse Electric Company Quality Management System and clarified that the Duke QA program is applicable to design, procurement, and construction activities associated with WLS Units 1 and 2 that may occur before as well as after the COL is issued.

By letter dated December 17, 2010, Duke Energy submitted Revision 3 of the WLS COL FSAR. The staff reviewed Section 17.1 of the WLS COL FSAR and confirmed that the applicant had (1) adequately identified which QA programs applied to the design, procurement, and construction activities described in section 17.1 of the WLS COL FSAR, and (2) adequately described the expected scope of work related to the COL activities. Therefore, RAI 17.5-8 is closed.

In RAI 17.5-9, dated October 6, 2008, the NRC staff requested that the applicant confirm when the Duke NPD QAPD will become effective, as well as clarify the difference between the statements that "the Duke NPD QAPD will become effective at COL issuance," located in the WLS COL FSAR Section 17.1, and that "the QA Program - Operation will be implemented 30 days prior to initial fuel loading," located in WLS COL FSAR Table 13.4-201.

In its response letter, dated December 11, 2008, the applicant stated that the Duke NPD QAPD becomes effective at COL issuance, and establishes the QA program requirements for the remaining portion of the design and construction phases. However, full implementation of operations-related requirements is not expected until 30 days prior to fuel load, as indicated in FSAR Table 13.4-201. Accordingly, as part of its response to RAI 17.5-9, the applicant proposed to revise the last paragraph of Section 17.1 of the WLS COL FSAR to state:

> Implementation of the applicable portions of the "Quality Assurance Program Description" (QAPD) discussed in Section 17.5 begins at COL issuance. The program establishes the QA program requirements for the remaining portion of the design and construction phases and for operations; full implementation of the operations related requirements will be no later than as indicated in Table 13.4-201.

By letter dated December 17, 2010, Duke Energy submitted Revision 3 of the WLS COL FSAR. The staff reviewed Section 17.1 of the WLS COL FSAR and confirmed that the applicant had provided (1) an acceptable implementation schedule for the Duke NPD QAPD, and (2) an adequate description regarding implementation of the Duke NPD QAPD at COL issuance, and establishment of the appropriate QA program requirements for the remaining portion of the design and construction phases. Therefore, RAI 17.5-9 is closed.

The following portion of this section is reproduced from Section 17.1.4 of the VEGP SER:

> *In addition, the applicant proposed revisions to Appendix 1AA in its letter, dated August 19, 2008, in response to the NRC staff's RAI 1-5. In its response, the applicant proposed to change the exception statements to address the version of NQA-1 instead of addressing the QAPD included in Part 11 of the BLN COL application. The NRC staff has verified that the proposed revision was incorporated into Revision 1 of the BLN COL FSAR for those RGs with QA requirements. RAI 1-5 is closed for all RGs that contain exception statement referencing NQA-1 (i.e., RG 1.28, 1.30, 1.38, 1.39, 1.94, and 1.116) except for RG 1.33.*
>
> *In RAI 1-11, dated December 16, 2008, the NRC staff requested that the applicant document the mechanism for incorporation of the requirements of RG 1.33 since these requirements are not covered by NQA-1. In its letter, dated January 27, 2009, the applicant stated that conformance with RG 1.33 will be supplemented in a future amendment to include a reference to Nuclear Energy Institute (NEI) 06-14A. The NRC staff has addressed this issue with NEI since NEI 06-14A does not commit to RG 1.33. This issue will remain open until closure is reached with NEI 06-14A or the applicant. This is identified as* ***Open Item 17.1-1****.*
>
> *<u>Resolution of Standard Content Open Item 17.1-1</u>*
>
> *In its letter, dated December 31, 2009, the applicant proposed to revise VEGP COL FSAR Section 1.9, Table 1.9-201, "Regulatory Guide/FSAR Section Cross-References," to document that RG 1.33, "Quality Assurance Program Requirements (Operation)," Revision 2, is addressed in Section IV of the QAPD. Additionally, the applicant proposed to revise Appendix 1AA of the VEGP COL FSAR to document conformance to RG 1.33. Therefore, Open Item 17.1-1 is resolved for VEGP and the proposed revisions are identified as* ***Confirmatory Item 17.1-1****, pending formal revision of the VEGP COL FSAR.*
>
> *<u>Resolution of Standard Content Confirmatory Item 17.1-1</u>*
>
> *Confirmatory Item 17.1-1 is an applicant commitment to revise its FSAR Table 1.9-201 and Appendix 1AA to document conformance to RG 1.33. The staff verified that the VEGP COL FSAR was appropriately updated. As a result, Confirmatory Item 17.1-1 is now closed.*

<u>WLS Resolution of Standard Content Open and Confirmatory Item 17.1-1</u>

In a letter dated November 4, 2010, the applicant endorsed the standard content material provided by VEGP. By letter dated December 17, 2010, the applicant provided Revision 3 of the WLS COL FSAR and Revision 3 of the Duke NPD QAPD. In Revision 3 of the Duke NPD QAPD, the applicant addressed the information related to Standard Content Open Item 17.1-1. The NRC staff has confirmed through review of Revision 3 of the Duke NPD QAPD that (1) the applicant has incorporated the appropriate changes to Appendix 1AA of the WLS COL FSAR and Part IV, “Regulatory Commitments,” of the Duke NPD QAPD, and (2) the applicant has adequately identified and specified conformance to RG 1.33, “Quality Assurance Program Requirements (Operation),” Revision 2, consistent with the NRC-approved NEI 06-14A, Revision 7, guidance. This adequately addresses the issue outlined by Confirmatory Item 17.1-1; therefore, Standard Content Open Item 17.1-1 is resolved for the WLS COL application.

In January 2011 the NRC staff conducted a limited scope inspection at the Duke Energy facility in Charlotte, North Carolina, as documented in inspection report numbers 05200018/2011-201 and 05200198/2011-201 dated March 16, 2011. The purpose of the NRC inspection was to verify that the QA processes and procedures were effectively implemented with regard to the WLS COL application. During this inspection, the NRC inspectors identified one violation and one non-cited violation of NRC requirements related to the WLS QA program. Duke Energy responded to the Notice of Violation in a letter dated April 15, 2011. Duke Energy identified its actions to correct and prevent recurrence of the violation and noted that full compliance was achieved. Based on this response, the staff does not intend to conduct a follow-up inspection as a part of the ongoing licensing process.

17.1.5 Post Combined License Activities

There are no post COL activities related to this section.

17.1.6 Conclusion

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff’s review confirmed that the applicant addressed the required information relating to QA during the design and construction phase, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Based on the information provided by the applicant, the staff concludes that WLS COL 17.5-1 meets the applicable Appendix B to 10 CFR Part 50 and 10 CFR 52.79(a)(25) requirements.

17.2 Quality Assurance During the Operations Phase

Section 17.2, “Quality Assurance During the Operations Phase,” of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 17.2 of Revision 19 to the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issues relating to this section remained for review.[1] The review confirmed that there are no outstanding issues related to this section. The results of the NRC staff’s technical evaluation of the information incorporated by reference into the WLS COL application are documented in NUREG-1793 and its supplements.

17.3 Quality Assurance During Design, Procurement, Fabrication, Inspection, and/or Testing of Nuclear Power Plant Items (Related to RG 1.206, Section C.III.1, Chapter 17, C.I.17.3, "Quality Assurance Program Description")

Section 17.3, "Quality Assurance During Design, Procurement, Fabrication, Inspection, and/or Testing of Nuclear Power Plant Items," of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 17.3 of Revision 19 to the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issues relating to this section remained for review.[1] The review confirmed that there are no outstanding issues related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference into the WLS COL application are documented in NUREG-1793 and its supplements.

17.4 Design Reliability Assurance Program (Related to RG 1.206, Section C.III.1, Chapter 17, C.I.17.4, "Reliability Assurance Program Guidance")

17.4.1 Introduction

This reliability assurance program (RAP) provides reasonable assurance that a plant is designed, constructed, and operated in a manner that is consistent with the assumptions and risk insights related to structures, systems, and components (SSCs) that are identified as being significant contributors to plant safety as determined by using probabilistic, deterministic, or other methods of analysis. The information is obtained from sources such as the plant- and site-specific `probabilistic risk assessment (PRA), industry operating experience, relevant component failure databases, and expert panels.

The RAP is implemented in two stages. The first stage, the design reliability assurance program (D-RAP), comprises the reliability assurance activities necessary to provide confidence that the plant is consistent with the certified design when fuel is loaded for the first time. The second stage comprises the operational phase reliability assurance activities (OPRAAs) that are to be integrated into other programs.

17.4.2 Summary of Application

Section 17.4, "Design Reliability Assurance Program," of the WLS COL FSAR, Revision 11, incorporates by reference Section 17.4 of the AP1000 DCD, Revision 19.

In addition, in WLS COL FSAR Section 17.4, the applicant provided the following:

Supplemental Information

- STD SUP 17.4-1

The applicant provided supplemental (SUP) information in standard (STD) SUP 17.4-1 regarding the QA requirements for nonsafety-related SSCs within the scope of the D-RAP.

17.4.3 Regulatory Basis

The regulatory basis of the information incorporated by reference into WLS COL FSAR Section 17.4 is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the D-RAP are given in Section 17.4 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants." SECY-95-132, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," states the following:

> An application for advanced reactor DC or a COL must include: (1) the description of the RAP used during the design that includes, scope, purpose, and objectives; (2) the process used to evaluate and prioritize the SSCs in the design, based on their degree of risk significance; (3) a list of the SSCs designated as risk significant; and (4) for those SSCs designated as risk significant: (i) a process to determine dominant failure modes that considered industry experience, analytical models, and applicable requirements; and (ii) key assumptions and risk insights from probabilistic, deterministic, or other methods that considered operations, maintenance, and monitoring activities.
>
> Each licensee that references the advanced reactor design must implement the design reliability assurance program approved by the NRC.

The Commission approved this position in the associated staff requirements memorandum (SRM) dated June 28, 1995.

Regulatory Guide (RG) 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," describes an acceptable way to satisfy these requirements.

17.4.4 Technical Evaluation

The NRC staff reviewed Section 17.4 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the D-RAP. The results of the staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the NRC staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.
- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.
- The staff verified that the site-specific differences were not relevant.

The NRC staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the BLN Units 3 and 4 COL application. Any confirmatory items in the standard content material retain the numbers assigned in the VEGP SER.

The following portion of this technical evaluation section is reproduced from Section 17.4.4 of the VEGP SER:

> *<u>Supplemental Information</u>*
>
> - *STD SUP 17.4-1*
>
> *The applicant provided supplemental information in STD SUP 17.4-1 to describe the QA requirements for nonsafety-related SSCs within the scope of D-RAP.*
>
> *The following portion of this technical evaluation section is reproduced from Section 17.4.4 of the BLN SER:*
>
> *No site specific structures, systems, and components (SSCs) have been added to the D-RAP. The applicant asserts that the AP1000 DCD and PRA bound all site specific hazards and associated risks. The staff's evaluation of the probabilistic methods used to reach this conclusion is documented in Chapter 19 of this safety evaluation. The staff concludes that the list of SSCs incorporated by reference to the DCD is an acceptable list for the BLN COL.*
>
> *The staff noted that risk metrics may change with modifications to the plant design or other new information and requested additional information on how the applicant would address risk significant SSCs that are identified after the COL is issued (RAI 17.4-1). In its response dated September 17, 2008, the applicant stated that such changes would be captured and included in the appropriate OPRAAs in accordance with procedures developed under the QA program. In addition, the response states that the [Maintenance Rule] MR program is to be consistent with NEI 07-02A, "Generic FSAR Template Guidance for Maintenance Rule Program Description for Plants Licensed under 10 CFR Part 52," which has been endorsed by the staff in a letter to NEI, dated January 24, 2008.*
>
> *The Maintenance Rule program description calls for establishment of an expert panel prior to fuel load. As additional information is developed, such a panel alters the scope of OPRAAs as appropriate. Because this provides assurance that changes will receive appropriate review, the staff finds it acceptable; therefore, RAI 17.4-1 is closed.*
>
> *However, the staff requested that the applicant supplement the BLN COL FSAR to describe the organizational and process aspects of the RAP that will be performed by the COL holder (RAI 17.4-2). In its response dated April 9, 2009, the applicant proposed to revise the BLN COL FSAR Section 17.4 to include a*

standard supplement identifying the quality assurance requirements for nonsafety-related SSCs within the scope of D-RAP. This is consistent with RG 1.206 and is therefore an acceptable method for meeting the Commission's policy for RAP. The staff identifies the need for a revision to the BLN COL FSAR as ***Confirmatory Item 17.4-1****.*

Resolution of Standard Content Confirmatory Item 17.4-1

Confirmatory Item 17.4-1 required the applicant to update its FSAR to include a standard supplement identifying the QA requirements for nonsafety-related SSCs within the scope of D-RAP. The NRC staff verified that the VEGP COL FSAR was appropriately updated with STD SUP 17.4-1. As a result, Confirmatory Item 17.4-1 is resolved.

The NRC staff verified that the WLS COL FSAR was appropriately updated with STD SUP 17.4-1. As a result, Standard Content Confirmatory Item 17.4-1 is resolved.

17.4.5 Post Combined License Activities

There are no post COL activities related to this section.

17.4.6 Conclusion

The NRC staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to the D-RAP, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The NRC staff concludes that the relevant information presented in Section 17.4 of the WLS COL FSAR is consistent with the guidance provided in SECY-95-132, as well as the requirements of 10 CFR 52.47(b)(1), and 10 CFR 52.80(a). Therefore, the WLS D-RAP described in Section 17.4 of the WLS COL FSAR is acceptable.

17.5 Quality Assurance Program Description – New License Applicants (Related to RG 1.206, Section C.III.1, Chapter 17, C.I.17.5, "Quality Assurance Program Guidance")

17.5.1 Introduction

The QA program during the design, fabrication, construction, testing, and operation phases of a nuclear power plant is discussed in this section. Implementation of the applicable portions of the QAPD referenced in Section 17.5, "Quality Assurance Program Description – New License Applicants," of the WLS COL FSAR begins at COL issuance with full implementation of the operations-related requirements consistent with the outline provided in WLS COL FSAR Table 13.4-201, "Operational Programs Required by NRC Regulations."

17.5.2 Summary of Application

In Part 11 of the WLS COL application, the applicant provided a QAPD to be in place during the design, construction, and operations phases of WLS Units 1 and 2. This QAPD will be incorporated by reference in Section 17.5 of the WLS COL FSAR upon resolution of WLS Confirmatory Item 17.5-1, as discussed in the technical section below.

In addition, in WLS COL FSAR Section 17.5, the applicant provided the following:

AP1000 COL Information Items

- WLS COL 17.5-1

The applicant provided additional information in WLS COL 17.5-1 to address COL Information Item 17.5-1. In WLS COL 17.5-1, the applicant addresses the quality assurance program under which the COL application was developed for the design and construction phases, which is applicable until COL issuance. Section 17.5 of the WLS COL FSAR addresses the QA program for the remaining portion of the design and construction phases following COL issuance, which is described in the Duke NPD QAPD.

- STD COL 17.5-2

The applicant provided additional information in STD COL 17.5-2 to address COL Information Item 17.5-2. STD COL 17.5-2 addresses QA programs for procurement, fabrication, installation, construction, and testing of SSCs in the plant.

- STD COL 17.5-4

The applicant provided additional information in STD COL 17.5-4 to address COL Information Item 17.5-4. STD COL 17.5-4 addresses the QA program for operations, and uses FSAR Table 13.4-201 to provide milestones for operational quality assurance program implementation.

- STD COL 17.5-8

The applicant provided additional information in STD COL 17.5-8 to address COL Information Item 17.5-8. STD COL 17.5-8 addresses operational RAP integration with the QA program.

17.5.3 Regulatory Basis

The acceptance criteria associated with the relevant requirements of the Commission regulations for the QAPD are given in Section 17.5 of NUREG-0800. The applicable regulatory requirements for the QAPD are as follows:

Appendix B to 10 CFR Part 50 requires that the application include a description of the QA program to be applied to the design, fabrication, construction, and testing of the SSCs of the facility and establishes QA requirements for the design, construction, and operation of those SSCs. The pertinent requirements of Appendix B apply to all activities affecting the safety-related functions of the SSCs, including designing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying.

Section 10 CFR 52.79(a)(17) requires that the application include information with respect to compliance with technically relevant positions of the Three Mile Island requirements of 10 CFR 50.34(f).

Section 10 CFR 52.79(a)(25) requires that the description of the QA program include a discussion of how the applicable requirements of Appendix B have been and will be satisfied, and also include a discussion of how the QA program will be implemented.

Further, 10 CFR 52.79(a)(27) requires that the application include information on the managerial and administrative controls to be used for a nuclear power plant and include a discussion of how the applicable requirements of Appendix B will be satisfied.

17.5.4 Technical Evaluation

The NRC staff reviewed Section 17.5 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the QAPD. The results of the staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the NRC staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The NRC staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the BLN Units 3 and 4 COL application. Any confirmatory items in the standard content material retain the numbers assigned in the VEGP SER.

Although the NRC staff concluded that the evaluation performed for the standard content is directly applicable to the WLS COL application, there were differences between the information provided by the WLS applicant and that provided by the VEGP applicant regarding details in the

WLS COL FSAR and the Duke NPD QAPD. The resolutions of these differences for WLS are evaluated by the staff following the standard content material to which they apply.

The following portion of this technical evaluation section is reproduced from Section 17.5.4 of the VEGP SER:

> *The NRC staff reviewed Section 17.5 of the BLN COL FSAR and the QAPD provided in Part 11 of the BLN COL application. In RAI 17.5-9, dated May 12, 2008, the NRC staff requested that the applicant explain why the QAPD provided in Part 11 of the BLN COL application is not referenced or incorporated by reference in the BLN COL FSAR Section 17.5. In its letters, dated June 26, 2008, and October 16, 2008, the applicant proposed to revise Section 17.5 of the BLN COL FSAR to state that the QAPD is incorporated by reference. In addition, the applicant proposed to revise Section 17.5 of the BLN COL FSAR to provide the title of the QAPD that is incorporated by reference. The NRC staff has reviewed the proposed revisions to Section 17.5 and concluded that the proposed changes are responsive to RAI 17.5-9. The NRC staff has verified that the proposed revision was incorporated into Revision 1 of the BLN COL FSAR. RAI 17.5-9 is closed.*
>
> *The NRC staff has verified that the proposed revision to incorporate the QAPD by reference was incorporated into the VEGP COL FSAR. In its letter dated January 29, 2010, the applicant proposed to revise Section 17.5 of the VEGP COL FSAR to provide the title of the QAPD that is incorporated by reference. This item is identified as* ***Confirmatory Item 17.5-1****, pending formal revision of the VEGP COL FSAR.*
>
> *<u>Resolution of Standard Content Confirmatory Item 17.5-1</u>*
>
> *Confirmatory Item 17.5-1 is an applicant commitment to revise its FSAR Section 17.5 to specify the title of the QAPD. The staff verified that the VEGP COL FSAR was appropriately updated. As a result, Confirmatory Item 17.5-1 is now closed.*

<u>WLS Resolution of Standard Content Confirmatory Item 17.5-1</u>

In RAI 17.5-11, dated October 6, 2008, the NRC staff noted that WLS COL FSAR Section 17.5 does not either include or incorporate the Duke NPD QAPD by reference. In its response letter, dated December 11, 2008, the applicant proposed to revise Section 17.5 of the WLS COL FSAR to identify that the QAPD is included as Part 11 of the Lee COL application and is incorporated by reference.

By letter dated December 17, 2010, Duke Energy submitted Revision 3 of the WLS COL FSAR. The staff reviewed Section 17.5 of the WLS COL FSAR and confirmed that the applicant had (1) adequately identified which QA program is in place during the design, construction, and operations phases, as described by the Duke NPD QAPD, and (2) adequately identified that the QAPD is included as Part 11 of the Lee COL application and is incorporated by reference. Therefore, RAI 17.5-11 is closed.

In addition, by letter dated November 4, 2010, the applicant endorsed the standard content material provided by VEGP in its letters dated January 29, 2010, and April 2, 2010, in reference to the BLN response to RAI 17.5-9 as standard, and proposed to incorporate the standard content in a future revision of the WLS COL FSAR. The applicant provided its commitment to incorporate the standard content material that consists of revising Section 17.5 of the WLS COL FSAR to incorporate the Duke NPD QAPD by reference and to provide the title of the QAPD that is incorporated by reference. By letter dated December 17, 2010, the applicant provided Revision 3 of the WLS COL FSAR. The NRC staff confirmed that Revision 3 included reference to the Duke NPD QAPD by title in Section 17.1 and 17.5 of the WLS COL FSAR; therefore, Standard Content Confirmatory Item 17.5-1 is resolved for the WLS COL application.

In RAI 17.5-1, dated October 6, 2008, the NRC staff noted that the Duke NPD QAPD, Part I, Section 1.1, “Scope / Applicability,” states that the QAPD applies to COL / construction / pre-operation and/or operation activities, which is not consistent with the scope of the QAPD as stated in QAPD Part I, Section 1 or WLS COL FSAR Table 13.4-201, which lists the QA program as a required operational program. Accordingly, the staff requested that the applicant clarify the scope of the Duke NPD QAPD since it applies to all the stated activities.

In its response letter, dated December 11, 2008, the applicant stated that the Duke NPD QAPD will be revised to clarify that the scope of the Duke NPD QAPD applies to all stated activities. By letter dated December 17, 2010, Duke Energy submitted Revision 3 of the WLS COL FSAR as well as Revision 3 of the Duke NPD QAPD. The staff reviewed Part I, Section 1.1 of the QAPD and confirmed that the applicant had provided an acceptable scope for the activities addressed by the Duke NPD QAPD. Therefore, RAI 17.5-1 is closed.

In RAI 17.5-10, dated October 6, 2008, the NRC staff requested that the applicant provide an evaluation of the existing Duke Energy QA program against the applicable acceptance criteria in Section 17.5 of NUREG-0800, pursuant to the requirements of 10 CFR 52.79(a)(41), which requires that COL applicants must provide an evaluation of the facility against the SRP revision in effect six months before the docket date of the application.

In its response letter, dated December 11, 2008, the applicant stated that the Duke Energy Carolinas Topical Report, Quality Assurance Program, was reviewed and evaluated by the NRC and determined to meet the requirements of Appendix B to 10 CFR Part 50, utilizing the applicable acceptance criteria in NUREG-0800, Section 17.5. The applicant also stated that the QAPD described in Section 17.5 of the WLS COL FSAR, which will apply after COL issuance, has been evaluated for conformance to NUREG-0800, Section 17.5 and discussed in Table 1.9-202 of the WLS COL FSAR and found acceptable. The NRC staff has reviewed the response and determined that the applicant’s response is acceptable. Therefore, RAI 17.5-10 is closed.

In RAI 17.5-2, dated October 6, 2008, the NRC staff requested that the applicant provide clarification of how WLS siting activities, as described in Duke NPD QAPD Part I, Section 1.1, would be subject to the provisions of the QAPD, since siting activities for WLS would be complete at the time of COL issuance.

In its response letter, dated December 11, 2008, the applicant stated that siting under the Duke NPD QAPD would not be applicable to WLS. However, the siting activity was included in the listing of activities to which the Duke NPD QAPD applies based on the development of the

QAPD to serve as topical report for all potential future Duke Energy new nuclear plant development activities.

On the basis of the Duke Energy response, which clarified how siting activities discussed in the WLS COL FSAR would be subject to the Duke NPD QAPD described in Section 17.5 of the WLS COL FSAR, the NRC staff determined that the issue has been adequately resolved for the WLS COL application. Therefore, RAI 17.5-2 is closed.

The NRC staff reviewed the resolution of COL information items STD COL 17.5-2, STD COL 17.5-4, STD COL 17.5-8, and WLS COL 17.5-1, which are addressed in the Duke NPD QAPD. The Duke NPD QAPD is based on NEI 06-14A, "Quality Assurance Program Description," Revision 7, which was approved by the NRC staff using Section 17.5 of NUREG-0800. The staff's review of these four COL information items is a combination of plant-specific evaluation and standard content evaluation.

AP1000 COL Information Items

- STD COL 17.5-2, STD COL 17.5-4, STD COL 17.5-8 and WLS COL 17.5-1

The following portion of this section is reproduced from Section 17.5.4 of the VEGP SER:

> *The NEI 06-14A template provided generic information and format for QAPDs with bracketed areas for applicants to provide plant-specific information. The generic information in NEI 06-14A provides the information required for STD COL 17.5-2, 17.5-4, and 17.5-8. In its review of TVA QAPD, the NRC staff used Section 17.5 of NUREG-0800 and RG 1.206 as guidance. The NRC staff developed Section 17.5 of NUREG-0800 using American Society of Mechanical Engineers (ASME) standard ASME NQA-1-1994, "Quality Assurance Requirements for Nuclear Facility Applications," as supplemented by additional regulatory and industry guidance for nuclear operating facilities.*

Further NRC staff evaluation of the COL information items and the associated sections of the Duke NPD QAPD is provided in the following sections.

17.5.4.1 *Organization*

The following portion of this technical evaluation section is reproduced from Section 17.5.4.1 of the VEGP SER:

> *TVA's QAPD follows the guidance of Section 17.5 of NUREG-0800, paragraph II.A. The QAPD describes and defines the responsibility and authority for planning, establishing, and implementing an effective overall QA program. The QAPD provides a description of an organizational structure, functional responsibilities, levels of authority, and interfaces for establishing, executing, and verifying QAPD implementation. The QAPD establishes independence between the organization responsible for checking a function and the organization that performs the function. In addition, the QAPD allows TVA management to size the QA organization commensurate with the duties and responsibilities assigned.*

> *In the QAPD, TVA commits to comply with the quality standards described in NQA-1-1994, Basic Requirement 1, and Supplement 1S-1.*

<u>WLS RAI 17.5-3 and RAI 17.5-4</u>

During its review of the Duke NPD QAPD, the NRC staff identified several issues regarding QAPD Part II, Section 1, "Organization," that required further clarification.

In RAI 17.5-3, dated October 6, 2008, the NRC staff requested that the applicant restructure the Organization section of the Duke NPD QAPD as follows:

> Clearly delineate (1) how the QA program is implemented during the period of construction and testing, and (2) how the QA program is implemented during the operations phase. The transition process during which the operational programs become effective should be described. Position descriptions, including roles, responsibilities, and lines of authority, should be included for applicable corporate and line positions that implement and verify elements of the QA program and the associated administrative controls.

In RAI 17.5-4, dated October 6, 2008, the NRC staff requested that the applicant provide (1) clarification regarding the inclusion of organizational charts in the Duke NPD QAPD, and (2) additional clarifications and expanded level of detail regarding organizational descriptions provided in Part II, Section 1, of the Duke NPD QAPD.

In its response letter, dated December 11, 2008, the applicant stated, in part, that Duke Energy will revise the Duke NPD QAPD to define the appropriate organizational structure, roles and responsibilities, and reporting relationships for the Duke Energy organizations that will implement the requirements of the QAPD for the development, construction, and operation of new nuclear generating plants. The organizational descriptions and organization charts contained within the Duke NPD QAPD, as revised, will define the corporate and Nuclear Generation Group organizations that implement the quality assurance requirements of the Duke NPD QAPD in support of the development, construction, and operation of the units.

Specifically, the response to RAI 17.5-4 provides descriptions of how the QA program is implemented during the construction and testing phases and during the operations phase, as well as the transition process. The response to RAI 17.5-4 also provides additional information on the QA program responsibilities, implementation, and administrative controls. In addition, the applicant noted that the basic implementation of the QA program is the same during construction and testing and operations; only the activities being implemented differ. However, the manager responsible for Quality Assurance and Oversight performs the independent oversight functions throughout all phases of QA program implementation.

<u>Resolution of WLS RAI 17.5-3 and RAI 17.5-4</u>

By letter dated December 17, 2010, Duke Energy submitted Revision 3 of the WLS COL FSAR as well as Revision 3 of the Duke NPD QAPD. The staff confirmed that Revision 3 of the Duke NPD QAPD incorporated a description of the WLS organization, including organizational charts, consistent with the applicant's RAI responses. Therefore, RAI 17.5-3 and RAI 17.5-4 are closed.

17.5.4.2 *Quality Assurance Program*

The following portion of this technical evaluation section is reproduced from Section 17.5.4.2 of the VEGP SER:

> *TVA's QAPD follows the guidance of Section 17.5 of NUREG-0800, paragraph II.B. The QAPD establishes measures to implement a QA program to ensure that the design, construction, and operation of a nuclear power plant are in accordance with governing regulations and license requirements. The QA program comprises those planned and systematic actions necessary to provide confidence that SSCs will perform their intended safety function, including certain nonsafety-related SSCs and activities that are significant contributors to plant safety, as described in the applicant's FSAR. The QA program requires that a list or system identifying SSCs and activities to which the QAPD applies be maintained.*
>
> *The QAPD provides measures to assess the adequacy of the QAPD and to ensure its effective implementation at least once each year or at least once during the life of the activity, whichever is shorter. The program allows the period for assessing the QAPD during the operations phase to be extended to once every 2 years. In addition, consistent with Section 17.5 of NUREG-0800, paragraph II.B.8, the QAPD applies a grace period of 90 days to activities that must be performed on a periodic basis. The next due date for the performance of an activity that invokes the 90-day grace period remains unchanged. The next due date for an activity performed before the scheduled due date is moved backwards so that the interval prescribed for the performance of the activity is not exceeded.*
>
> *The QAPD also follows the guidance of Section 17.5 of NUREG-0800, paragraphs II.S and II.T. The QAPD describes measures to establish and maintain formal indoctrination and training programs for personnel performing, verifying, or maintaining activities within the scope of the QAPD to ensure that they achieve and maintain suitable proficiency. The plant's technical specifications delineate the minimum qualifications for plant and support staff. Personnel are required to complete the training for positions identified in 10 CFR 50.120, "Training and Qualification of Nuclear Power Plant Personnel," according to programs accredited by the National Nuclear Accrediting Board of the National Academy for Nuclear Training. The QAPD also provides the minimum training requirements for managers responsible for QAPD implementation, in addition to the minimum training requirements for the individuals responsible for planning, implementing, and maintaining the QAPD.*
>
> *The QAPD also follows Section 17.5 of NUREG-0800, paragraph II.W. The QAPD provides measures for establishing an independent review program for activities occurring during the operational phase. In the QAPD, TVA commits to comply with the quality standards described in NQA-1-1994, Basic Requirement 2, and Supplements 2S-1, 2S-2, 2S-3, and 2S-4, with the following alternatives:*
>
> - *NQA-1-1994, Supplement 2S-1, includes NQA-1-1994, Appendix 2A-1. The QAPD proposes the following alternatives to the implementation of Supplement 2S-1 and Appendix 2A-1:*

- *NQA-1-1994, Supplement 2S-1, states that the organization designate those activities that require qualified inspectors and test personnel and establish written procedures for the qualification of these personnel. As an alternative to this requirement, the QAPD proposes that a qualified engineer may plan inspections, evaluate the capabilities of an inspector, or evaluate the training program for inspectors. For the purposes of these functions, a qualified engineer is one who has a baccalaureate degree in engineering in a discipline related to the inspection or test activity (i.e., electrical, mechanical, or civil engineering) and has at least 5 years of engineering work experience, with at least 2 years of this experience regarding nuclear facilities. The NRC staff evaluated this proposed alternative and determined that the designation of a qualified engineer to plan inspections, evaluate inspectors, or evaluate the inspector qualification programs is consistent with the training and qualification criteria of 10 CFR Part 50, Appendix B, Criterion II, "Quality Assurance Program," and NQA-1-1994, Supplement 2S-1. Therefore, the NRC staff concluded that this alternative is acceptable.*

- *NQA-1-1994, Appendix 2A-1 provides guidance for qualifying inspection and test personnel as Level I, II, or III. As an alternative to this guidance, the QAPD proposes that personnel performing independent quality verification inspections, examinations, measurements, or tests will be required to possess qualifications equal to or better than those required for performing the task being verified. In addition, the verification performed must be within the skills of these personnel and addressed by procedures. These personnel will not be responsible for planning quality verification inspections or tests (i.e., establishing hold points and acceptance criteria in procedures, and determining responsibility for performing the inspection), evaluating inspection training programs, or certifying inspection personnel. The NRC staff evaluated this proposed alternative and determined that it is consistent with inspection and test personnel initial qualification requirements specified in Section 17.5 of NUREG-0800, paragraph II.T.5. Therefore, the NRC staff concluded that this alternative is acceptable.*

- *NQA-1-1994, Supplement 2S-2, states that nondestructive examination personnel must be qualified. As an alternative to this requirement, the QAPD proposes to follow the applicable standard cited in Sections III and XI of the ASME Boiler and Pressure Vessel Code. 10 CFR 50.55a, "Codes and Standards," also requires the use of the latest Edition and Addenda of Sections III and XI of the ASME Code. The NRC staff evaluated this proposed alternative and determined that it is consistent with the regulation in 10 CFR 50, Appendix B, Criterion II, "Quality Assurance Program." Therefore, the NRC staff concluded that this alternative is acceptable.*

- *NQA-1-1994, Supplement 2S-3, states that the prospective lead auditors must have participated in a minimum of five audits in the previous*

> *3 years. As an alternative to this requirement, the QAPD proposes to follow the guidance provided in Section 17.5 of NUREG-0800, paragraph II.S.4.c, which states that prospective lead auditors shall demonstrate their ability to properly conduct the audit process, as implemented by the company, to effectively lead an audit team, and to effectively organize and report results, including participation in at least one nuclear audit within the year preceding the date of qualification. The NRC staff evaluated this proposed alternative and determined that it is consistent with the regulation in 10 CFR Part 50, Appendix B, Criterion II. Therefore, the NRC staff concluded that this alternative is acceptable.*

In RAI 17.5-5, dated May 12, 2008, the NRC staff requested that the applicant revise the TVA QAPD Part II, Section 2.5 to cite the correct regulation of 10 CFR 52.79(a)(27) versus 10 CFR 50.34(b)(6)(ii). In its response dated June 26, 2008, the applicant proposed to revise the TVA QAPD Part II, Section 2.5 consistent with the proposed wording in NEI Technical Report 06-14A, "Quality Assurance Program Description," Revision 5, dated May 2008. Revision 5 of NEI 06-14A has not been approved by the NRC staff; therefore, this issue will remain open until Revision 5 of NEI 06-14A is approved and TVA has incorporated the approved changes into the TVA QAPD. This is identified as ***Open Item 17.5-1***.

Resolution of Standard Content Open Item 17.5-1

Revision 7 of NEI 06-14A was approved by the NRC staff in a letter dated November 3, 2009, and adequately addressed RAI 17-5-5. In a letter dated December 31, 2009, the VEGP applicant provided a markup of Revision 9 of the SNC QAPD. The NRC staff has reviewed the markup of SNC QAPD, Revision 9, and determined that conforming changes have been proposed to Section 2.5 consistent with NEI 06-14A, Revision 7. On this basis, Open Item 17.5-1 is ***Confirmatory Item 17.5-7*** *for the VEGP COL application.*

Resolution of Standard Content Confirmatory Item 17.5-7

Confirmatory Item 17.5-7 is an applicant commitment to revise its QAPD. The staff verified that the VEGP COL application was appropriately updated. As a result, Confirmatory Item 17.5-7 is now closed.

WLS Resolution of Standard Content Open Item 17.5-1 and Confirmatory Item 17.5-7

In a letter dated November 4, 2010, the applicant endorsed the standard content material provided by VEGP. By letter dated December 17, 2010, the applicant provided Revision 3 of the WLS COL FSAR and Revision 3 of the Duke NPD QAPD. In Revision 3 of the Duke NPD QAPD, the applicant addressed the information related to Standard Content Open Item 17.5-1. The NRC staff has confirmed through review of Revision 3 of the Duke NPD QAPD that the applicant has incorporated the appropriate changes to Part II, Section 2.5, of the Duke NPD QAPD, which is consistent with the guidance contained in NRC-approved NEI 06-14A, Revision 7. This adequately addresses the issue outlined by Confirmatory Item 17.5-7; therefore, Standard Content Open Item 17.5-1 is resolved for the WLS COL application.

The following portion of this technical evaluation section is reproduced from Section 17.5.4.2 of the VEGP SER:

In RAI 17.5-6, the NRC staff requested that the applicant explain how the discussion of the Independent Review Committee responsibilities in Part II, Section 2.7 of the TVA QAPD is consistent with the requirements of American National Standards Institute (ANSI) N18.7. In its response dated June 26, 2008, the applicant proposed to revise the TVA QAPD Part II, Section 2.7 consistent with the proposed wording in NEI 06-14A, Revision 5. This issue will remain open until Revision 5 of NEI 06-14A is approved and TVA has incorporated the approved changes into the TVA QAPD. This is identified as ***Open Item 17.5-2****.*

Resolution of Standard Content Open Item 17.5-2

NEI 06-14A, Revision 7, adequately addressed RAI 17.5-6. In a letter dated December 31, 2009, the applicant provided a markup of Revision 9 of the SNC QAPD. The NRC staff has reviewed the markup of SNC QAPD, Revision 9, and determined that conforming changes have been proposed to Section 2.7 consistent with NEI 06-14A, Revision 7. On this basis, Open Item 17.5-2 is ***Confirmatory Item 17.5-8*** *for the VEGP COL application.*

Resolution of Standard Content Confirmatory Item 17.5-8

Confirmatory Item 17.5-8 is an applicant commitment to revise its QAPD. The staff verified that the VEGP COL application was appropriately updated. As a result, Confirmatory Item 17.5-8 is now closed.

WLS Resolution of Standard Content Open Item 17.5-2 and Confirmatory Item 17.5-8

In a letter dated November 4, 2010, the applicant endorsed the standard content material provided by VEGP. By letter dated December 17, 2010, the applicant provided Revision 3 of the WLS COL FSAR and Revision 3 of the Duke NPD QAPD. In Revision 3 of the Duke NPD QAPD, the applicant addressed the information related to Standard Content Open Item 17.5-2. The NRC staff has confirmed through review of Revision 3 of the Duke NPD QAPD that the applicant has incorporated the appropriate changes to Part II, Section 2.7, of the Duke NPD QAPD, which is consistent with the guidance contained in NRC-approved NEI 06-14A, Revision 7. This adequately addresses the issue outlined by Confirmatory Item 17.5-8; therefore, Standard Content Open Item 17.5-2 is resolved for the WLS COL application.

WLS RAI 17.5-5

During its review of the Duke NPD QAPD, the NRC staff identified several issues regarding QAPD Part II, Section 2, “Quality Assurance Program,” that required further clarification.

In RAI 17.5-5, dated October 6, 2008, the NRC staff requested that the applicant revise the language in Part II, Section 2, of the Duke NPD QAPD, which states that the QAPD applies to those quality-related activities that involve the functions of safety-related activities of structures, systems, and components (SSCs), as described in the WLS COL FSAR. The revised language should include a description of the QA program applied to the design, and to be applied to the fabrication, construction, and testing, of the SSCs of the facility, as well as to the managerial and administrative controls to be used to assure safe operation. The NRC staff also requested that the applicant identify the corresponding WLS COL FSAR section(s) that describe safety-related SSCs or clarify the purpose of this statement in the Duke NPD QAPD.

In its response letter, dated December 11, 2008, the applicant stated that Duke Energy developed and prepared the Duke NPD QAPD consistent with NRC-approved template NEI 06-14A for the format and content of standard and site specific sections. Duke Energy committed to review and implement the appropriate standard and site specific text changes to Part II, Section 2 of the Duke NPD QAPD, in order to describe these programmatic controls within the QAPD, following approval of NEI 06-14, Revision 5, by the NRC. Since that time, the NRC staff has reviewed and approved NEI 06-14A, Revision 7, which has been adopted by the applicant as the foundation for the Duke NPD QAPD.

The applicant also noted that safety-related SSCs are appropriately described in the DCD and associated FSAR content consistent with RG 1.206; however, SSC requirements were included in Duke NPD QAPD Part II, Section 2, in order to allow the QAPD to be used for potential future new plant development activities. The NRC staff has reviewed the applicant's response to this portion of RAI 17.5-5 and determined that the applicant's response is acceptable.

By letter dated December 17, 2010, Duke Energy submitted Revision 3 of the WLS COL FSAR as well as Revision 3 of the Duke NPD QAPD. The staff confirmed that Revision 3 of the Duke NPD QAPD incorporated a description of the quality-related activities that involve the functions of safety-related activities of SSCs, consistent with the NRC-approved NEI 06-14A, Revision 7, description. Therefore, RAI 17.5-5 is closed.

<u>WLS RAI 17.5-6</u>

In RAI 17.5-6, dated October 6, 2008, the NRC staff requested that the applicant provide clarification of how the Duke NPD QAPD applies to Early Site Permit (ESP) applications / activities, which are referenced in QAPD Part II, Section 2, as well as elsewhere in the document, or remove these references from the QAPD.

In its response letter, dated December 11, 2008, the applicant stated that since Duke Energy did not execute an ESP for WLS, ESP requirements in the Duke NPD QAPD do not apply to the WLS site. WLS COL FSAR Section 17.1 clarifies the timing for and applicability of the QAPD to activities for WLS. However, ESP requirements were included in the Duke NPD QAPD in order to allow the QAPD to be used for potential future new plant development activities.

On the basis of the Duke Energy response, which clarified how ESP activities discussed in the WLS COL FSAR would be subject to the Duke NPD QAPD described in Section 17.5 of the WLS COL FSAR, the NRC staff determined that the issue has been adequately resolved for the WLS COL application. Therefore, RAI 17.5-6 is closed.

<u>WLS RAI 17.5-7</u>

In RAI 17.5-7, dated October 6, 2008, the NRC staff requested that the applicant identify the site-specific design basis activities in Part II, Section 2.3, of the Duke NPD QAPD, consistent with the guidance of NEI 06-14A, Section 2.3, which states that the QAPD in the COLA will be annotated to identify these activities; or justify their omission. Section 2.3 of the Duke NPD QAPD states that this information will be maintained in a project planning document.

In its response letter, dated December 11, 2008, the applicant stated that the Duke NPD QAPD Part II, Section 2.3, addresses identification of QA controls for ESP and COL application development. The wording of QAPD Part II, Section 2.3, is intended to address the activities that would apply to any future ESP or COL application development, and is not intended to be a commitment for the applicant to develop or submit annotated outlines for the COL application. Since Duke Energy did not execute an ESP for WLS, and the COL application preparation was not performed under the Duke NPD QAPD, this requirement would not be applicable to WLS. The applicant committed to revise Duke NPD QAPD Part II, Section 2.3 for clarity.

By letter dated December 17, 2010, Duke Energy submitted Revision 3 of the WLS COL FSAR as well as Revision 3 of the Duke NPD QAPD. The staff confirmed that Revision 3 of the Duke NPD QAPD incorporated a description of the site specific safety-related design basis activities, consistent with the applicant's RAI response, as well as the applicable NRC-approved NEI 06-14A, Revision 7, description. Therefore, RAI 17.5-7 is closed.

The following portion of this technical evaluation section is reproduced from Section 17.5.4.3 of the VEGP SER:

> *17.5.4.3 Design Control*
>
> *The following portion of this technical evaluation section is reproduced from Section 17.5.4.3 of the BLN SER:*
>
> *TVA's QAPD follows the guidance of Section 17.5 of NUREG-0800, paragraph II.C. The QAPD establishes the necessary measures to control the design, design changes, and temporary modifications (e.g., temporary bypass lines, electrical jumpers and lifted wires, and temporary setpoints) of items that are subject to the provisions of the QAPD. The QAPD design process includes provisions to control design inputs, outputs, changes, interfaces, records, and organizational interfaces with the applicant and its suppliers. These provisions ensure that the design inputs (i.e., design bases and the performance, regulatory, quality, and quality verification requirements) are correctly translated into design outputs (i.e., analyses, specifications, drawings, procedures, and instructions). In addition, the QAPD provides for individuals knowledgeable in QA principles to review design documents to ensure that they contain the necessary QA requirements.*
>
> *In the QAPD, TVA commits to comply with the quality standards described in NQA-1-1994, Basic Requirement 3 and Supplement 3S-1, to establish the program for design control and verification, Subpart 2.20 for the subsurface investigation requirements, and Subpart 2.7 for the standards for computer software QA controls.*

The following portion of this technical evaluation section is reproduced from Section 17.5.4.4 of the VEGP SER:

17.5.4.4 Procurement Document Control

The following portion of this technical evaluation section is reproduced from Section 17.5.4.4 of the BLN SER:

TVA's QAPD follows the guidance of Section 17.5 of NUREG-0800, paragraph II.D. The QAPD establishes the necessary administrative controls and processes to ensure that procurement documents include or reference applicable regulatory, technical, and QA program requirements. As noted in Section 17.5 of NUREG-0800, paragraph II.D.1, applicable technical, regulatory, administrative, quality, and reporting requirements (such as specifications, codes, standards, tests, inspections, special processes, and the regulation in 10 CFR Part 21, "Reporting of Defects and Noncompliance") are invoked for procurement of items and services.

In the QAPD, TVA commits to comply with the quality standards described in NQA-1-1994, Basic Requirement 4, and Supplement 4S-1, with the following alternatives and commitment:

- *NQA-1-1994, Supplement 4S-1, Section 2.3, states that procurement documents must require suppliers to have a documented QA program that implements NQA-1-1994, Part I.*

  - *As an alternative to this requirement, the QAPD proposes that suppliers have a documented QA program that meets Appendix B to 10 CFR Part 50, as applicable to the circumstances of the procurement. The NRC staff evaluated this proposed alternative and determined that it is consistent with Appendix B, Criterion IV, "Procurement Document Control." Therefore, the NRC staff concluded that this alternative is acceptable.*

  - *As an alternative to this requirement, the QAPD proposes that procurement documents allow suppliers to work under TVA's QAPD, including implementing procedures, if suppliers do not have their own QA program. The NRC staff evaluated this proposed alternative and determined that TVA's QAPD follows the guidance in Section 17.5 of NUREG-0800, paragraph II.G, regarding "Control of Purchased Material, Equipment, and Services." Specifically, the QAPD provides measures to evaluate prospective suppliers so that only qualified suppliers are selected, acceptance actions are performed for procured products and services, and suppliers are periodically audited and evaluated to ensure that qualified suppliers continue to provide acceptable products and services. Therefore, the NRC staff concluded that this alternative is acceptable.*

- *NQA-1-1994, Supplement 4S-1, Section 3, states that procurement documents are to be reviewed before award of the contract. As an* *alternative to this requirement, the QAPD proposes to conduct the QA review of procurement documents through review of the applicable procurement specification, including the technical and quality procurement requirements, before contract award. In addition, procurement document changes (e.g., scope, technical, or quality requirements) will also receive QA review. The NRC staff evaluated this proposed alternative and determined that it provides adequate QA review of procurement documents before awarding the contract and after any change. Therefore, the NRC staff concluded that this alternative is acceptable.*

- *In the QAPD, TVA commits that procurement documents prepared for commercial-grade items, procured as safety-related items, shall contain technical and quality requirements such that the procured item can be appropriately dedicated. The NRC staff evaluated this proposed commitment and determined that it is consistent with NRC staff guidance in Generic Letter (GL) 89-02, "Actions to Improve the Detection of Counterfeit and Fraudulently Marked Products," dated March 21, 1989, and GL 91-05, "Licensee Commercial-Grade Procurement and Dedication Programs," dated April 9, 1991, as delineated in Section 17.5 of NUREG-0800, paragraphs II.U.1.d and II.U.1.e.Therefore, the NRC staff concluded that this commitment is acceptable.*

*In RAI 17.5-7, dated May 12, 2008, the NRC staff requested that the applicant revise TVA QAPD Part II, Section 4 to substitute "TVA's" for "licensee's" to make it clear that a supplier may work under TVA's approved QA program. In its response dated June 26, 2008, the applicant stated that current use of "licensee's" is consistent with the wording in NEI 06-14A, Revision 4, which has been approved by the NRC staff. In a letter, dated September 17, 2008, the NRC staff requested NEI to address this question as part of a future revision to NEI 06-14A. This issue will remain open until Revision 5 of NEI 06-14A is approved and TVA has incorporated the approved changes into the TVA QAPD. This is identified as **Open Item 17.5-3**.*

Resolution of Standard Content Open Item 17.5-3

*NEI 06-14A, Revision 7, adequately addressed RAI 17.5-7. In a letter dated December 31, 2009, the applicant provided a markup of Revision 9 of the SNC QAPD. The NRC staff has reviewed the markup of SNC QAPD, Revision 9, and determined that conforming changes have been proposed to Section 4 consistent with NEI 06-14A, Revision 7. On this basis, Open Item 17.5-3 is **Confirmatory Item 17.5-9** for the VEGP COL application.*

Resolution of Standard Content Confirmatory Item 17.5-9

Confirmatory Item 17.5-9 is an applicant commitment to revise its QAPD. The staff verified that the VEGP COL application was appropriately updated. As a result, Confirmatory Item 17.5-9 is now closed.

WLS Resolution of Standard Content Open Item 17.5-3 and Confirmatory Item 17.5-9

In a letter dated November 4, 2010, the applicant endorsed the standard content material provided by VEGP. By letter dated December 17, 2010, the applicant provided Revision 3 of the WLS COL FSAR and Revision 3 of the Duke NPD QAPD. In Revision 3 of the Duke NPD QAPD, the applicant addressed the information related to Standard Content Open Item 17.5-3. The NRC staff has confirmed through review of Revision 3 of the Duke NPD QAPD that the applicant has incorporated the appropriate changes to Part II, Section 4, of the Duke NPD QAPD, which is consistent with the guidance contained in NRC-approved NEI 06-14A, Revision 7. This adequately addresses the issue outlined by Confirmatory Item 17.5-9; therefore, Standard Content Open Item 17.5-3 is resolved for the WLS COL application.

The following portion of this technical evaluation section is reproduced from Section 17.5.4.5 of the VEGP SER:

17.5.4.5 Instructions, Procedures, and Drawings

The following portion of this technical evaluation section is reproduced from Section 17.5.4.5 of the BLN SER:

TVA's QAPD follows the guidance of Section 17.5 of NUREG-0800, paragraph II.E. The QAPD establishes the necessary measures and governing procedures to ensure that activities affecting quality are prescribed by and performed in accordance with documented instructions, procedures, and drawings.

In the QAPD, TVA commits to comply with the quality standards described in NQA-1-1994, Basic Requirement 5, to establish procedural controls.

The following portion of this technical evaluation section is reproduced from Section 17.5.4.6 of the VEGP SER:

17.5.4.6 Document Control

The following portion of this technical evaluation section is reproduced from Section 17.5.4.6 of the BLN SER:

TVA's QAPD follows the guidance of Section 17.5 of NUREG-0800, paragraph II.F. The QAPD establishes the necessary measures and governing procedures to control the preparation, review, approval, issuance, and changes of documents that specify quality requirements or prescribe measures for controlling activities affecting quality, including organizational interfaces. The QAPD provides measures to ensure that the same organization that performed the original review and approval also review and approve revisions or changes to documents, unless other organizations are specifically designated.

A listing of all controlled documents identifying the current approved revision or date is maintained so personnel can readily determine the appropriate document for use. To ensure effective and accurate procedures during the operational phase, applicable procedures are reviewed and updated as necessary, *consistent with NRC staff guidance provided in Section 17.5 of NUREG-0800, paragraph II.F.8.*

In the QAPD, TVA commits to comply with the quality standards described in NQA-1-1994, Basic Requirement 6 and Supplement 6S-1, to establish provisions for document control.

The following portion of this technical evaluation section is reproduced from Section 17.5.4.7 of the VEGP SER:

17.5.4.7 *<u>Control of Purchased Material, Equipment, and Services</u>*

The following portion of this technical evaluation section is reproduced from Section 17.5.4.7 of the BLN SER:

TVA's QAPD follows the guidance of Section 17.5 of NUREG-0800, paragraph II.G. The QAPD establishes the necessary measures and governing procedures to control the procurement of items and services to ensure conformance with specified requirements. The program provides measures to evaluate prospective suppliers so that only qualified suppliers are selected. In addition, the program requires that suppliers be periodically audited and evaluated to ensure that qualified suppliers continue to provide acceptable products and services.

The program provides for acceptance actions, such as source verification, receipt inspection, pre- and post-installation tests, and review of documentation, such as certificates of conformance, to ensure that procurement, inspection, and test requirements have been satisfied before relying on the item to perform its intended safety function. Purchased items (such as components, spares, and replacement parts necessary for plant operation, refueling, maintenance, and modifications) and services are subject to quality and technical requirements at least equivalent to those specified for original equipment or by properly reviewed and approved revisions to ensure that the items are suitable for the intended service and are of acceptable quality, consistent with their effect on safety.

In the QAPD, TVA commits to comply with the quality standards described in NQA-1-1994, Basic Requirement 7 and Supplement 7S-1, to establish procurement verification control, with the following exceptions and alternatives:

- *NQA-1-1994, Basic Requirement 7 and Supplement 7S-1, state that procurement sources and suppliers' performance are to be evaluated. As an exception to these requirements, the QAPD proposes that other 10 CFR Part 50 licensees (other than TVA), authorized nuclear inspection agencies, the National Institute of Standards and Technology (NIST), and other State and Federal agencies that may provide items or services to TVA are not required to be evaluated or audited.*

  The NRC staff acknowledges that 10 CFR Part 50 licensees, authorized nuclear inspection agencies, the National Voluntary Laboratory Accreditation Program (NVLAP) administered by NIST, and other state *and federal agencies perform work under quality programs acceptable to the NRC, and that no additional audits or evaluations are required. However, TVA remains responsible for ensuring that procured items or services conform to its Appendix B program, applicable ASME Boiler and Pressure Vessel Code requirements, and other regulatory requirements and commitments. TVA also remains responsible for ensuring that the items or services are suitable for the intended application and for documenting the evaluation that supports this conclusion. The proposed exception provides an appropriate level of quality and safety. The NRC staff determined that this exception is acceptable as documented in a previous SE.*

- *Section 17.5 of NUREG-0800, paragraph II.L.8, establishes provisions for the procurement of commercial-grade calibration services for safety-related applications. As an exception to these provisions, the QAPD proposes that procurement source evaluation and selection measures not be required, provided all of the following conditions are met:*

  - *Purchase documents impose additional technical and administrative requirements to satisfy any licensee-specific QAPD and technical requirements.*

  - *Purchase documents require reporting as-found calibration data when calibrated items are found to be out of tolerance.*

  - *A documented review of the supplier's accreditation will be performed and will include a verification of the following:*

    - *The calibration laboratory holds a domestic accreditation by any one of the following accrediting bodies, which are recognized by the International Laboratory Accreditation Cooperation (ILAC) Mutual Recognition Arrangement (MRA):*

      - *National Voluntary Laboratory Accreditation Program (NVLAP), administered by the National Institute of Standards & Technology,*

      - *American Association for Laboratory Accreditation (A2LA).*

    - *The accreditation encompasses ANS/ISO/IEC 17025, "General Requirements for the Competence of Testing and Calibration Laboratories."*

    - *The published scope of accreditation for the calibration laboratory covers the necessary measurement parameters, range, and uncertainties.*

The NRC staff evaluated and found to be acceptable the NVLAP and A2LA accreditation programs. In RAI 17.5-13, dated May 12, 2008, the NRC staff *requested that the applicant justify the wording discrepancy between TVA QAPD Part II, Section 7.2 and Section 17.5 of NUREG-0800, Section II.L.8.c, regarding the NRC approved alternative for commercial grade calibration services. In its response dated June 24, 2008, the applicant stated that wording is consistent with the wording in NEI 06-14A, Revision 4, which has been approved by the NRC staff. In a letter, dated September 17, 2008, the NRC staff requested NEI to address this question as part of Revision 5 to NEI 06-14A. This issue will remain open until Revision 5 of NEI 06-14A is approved and TVA has incorporated the approved changes into the TVA QAPD. This is identified as* ***Open Item 17.5-4****.*

Resolution of Standard Content Open Item 17.5-4

NEI 06-14A, Revision 7, adequately addressed RAI 17.5-13. In a letter dated December 31, 2009, the VEGP applicant provided a markup of Revision 9 of the SNC QAPD. The NRC staff has reviewed the markup of SNC QAPD, Revision 9, and determined that conforming changes have been proposed to Section 7.2 consistent with NEI 06-14A, Revision 7. On this basis, Open Item 17.5-4 is ***Confirmatory Item 17.5-10*** *for the VEGP COL application.*

Resolution of Standard Content Confirmatory Item 17.5-10

Confirmatory Item 17.5-10 is an applicant commitment to revise its QAPD. The staff verified that the VEGP COL application was appropriately updated. As a result, Confirmatory Item 17.5-10 is now closed.

WLS Resolution of Standard Content Open Item 17.5-4 and Confirmatory Item 17.5-10

In a letter dated November 4, 2010, the applicant endorsed the standard content material provided by VEGP. By letter dated December 17, 2010, the applicant provided Revision 3 of the WLS COL FSAR and Revision 3 of the Duke NPD QAPD. In Revision 3 of the Duke NPD QAPD, the applicant addressed the information related to Standard Content Open Item 17.5-4. The NRC staff has confirmed through review of Revision 3 of the Duke NPD QAPD that the applicant has incorporated the appropriate changes to Part II, Section 7.2, of the Duke NPD QAPD, which is consistent with the guidance contained in NRC-approved NEI 06-14A, Revision 7. This adequately addresses the issue outlined by Confirmatory Item 17.5-10; therefore, Standard Content Open Item 17.5-4 is resolved for the WLS COL application.

The following portion of this technical evaluation section is reproduced from Section 17.5.4.7 of the VEGP SER:

The following portion of this technical evaluation section is reproduced from Section 17.5.4.7 of the BLN SER:

- *NQA-1-1994, Supplement 7S-1, Section 8.1, states that documentary evidence that items conform to procurement documents shall be available at the nuclear facility site prior to installation or use. As an alternative to the requirement for procurement documentary evidence to be available at the nuclear facility site during construction. The QAPD proposes that documentary evidence may be stored in physical form or in electronic media,* *under the control of TVA or its supplier(s), at a location(s) other than the nuclear facility site, as long as the documents can be accessed at the nuclear facility site during construction. After completion of construction, TVA will have sufficient documentary evidence to support operations. The NRC staff determined that implementation of this alternative would allow access to and review of the necessary procurement documentary evidence at the nuclear facility site, both before installation and use. Therefore, the NRC staff concluded that this alternative is acceptable.*

- *As an alternative to the requirements for the control of commercial-grade items and services in NQA-1-1994, Supplement 7S-1, Section 10, TVA commits in the QAPD to follow NRC guidance discussed in GL 89-02 and GL 91-05. In addition, TVA commits to establish and describe special quality verification requirements in applicable documents to assure that the commercially procured items will perform satisfactorily in service. In addition, the documents should provide for determining critical characteristics, technical evaluation, receipt requirements, and quality evaluation of the items to ensure that the items are suitable for their intended use. The NRC staff determined that this alternative will improve detection of counterfeit and fraudulently marked products and will improve the commercial-grade dedication programs. This alternative is consistent with the guidance of Section 17.5 of NUREG-0800, paragraphs II.U.1.d and II.U.1.e. Therefore, the NRC staff concluded that this alternative is acceptable.*

- *As an alternative to the requirements for the control of commercial-grade items and services in NQA-1-1994, Supplement 7S-1, Section 10, TVA commits to use other appropriate approved regulatory means and controls to support TVA commercial grade dedication activities. One example of this is NRC Regulatory Issue Summary (RIS) 2002-22, "Use of EPRI/NEI Joint Task Force Report, 'Guideline on Licensing Digital Upgrades: EPRI TR-102348, Revision 1, NEI 01-01: A Revision of EPRI TR-102348 to Reflect Changes to the 10 CFR 50.59 Rule.'" TVA will assume 10 CFR Part 21 reporting responsibility for all items that TVA dedicates as safety-related.*

In RAI 17.5-14, the NRC staff requested that the applicant provide an explanation as to how RIS 2002-22 represents an example of other approved regulatory means for commercial grade dedication activities. In its response dated June 24, 2008, the applicant stated that wording is consistent with the wording in NEI 06-14A, Revision 4, which has been approved by the NRC staff. In a letter, dated September 17, 2008, the NRC staff requested NEI to address this question as part of Revision 5 to NEI 06-14A. This issue will remain open until Revision 5 of NEI 06-14A is approved and TVA has incorporated the approved changes into the TVA QAPD. This is identified as ***Open Item 17.5-5****.*

Resolution of Standard Content Open Item 17.5-5

NEI 06-14A, Revision 7, adequately addressed RAI 17.5-14. In a letter dated December 31, 2009, the VEGP applicant provided a markup of Revision 9 of the SNC QAPD. The NRC staff has reviewed the markup of SNC QAPD, Revision 9, and determined that conforming changes have been proposed to Section 7.2 *consistent with NEI 06-14A, Revision 7. On this basis, Open Item 17.5-5 is* ***Confirmatory Item 17.5-11*** *for the VEGP COL application.*

Resolution of Standard Content Confirmatory Item 17.5-11

Confirmatory Item 17.5-11 is an applicant commitment to revise its QAPD. The staff verified that the VEGP COL application was appropriately updated. As a result, Confirmatory Item 17.5-11 is now closed.

WLS Resolution of Standard Content Open Item 17.5-5 and Confirmatory Item 17.5-11

In a letter dated November 4, 2010, the applicant endorsed the standard content material provided by VEGP. By letter dated December 17, 2010, the applicant provided Revision 3 of the WLS COL FSAR and Revision 3 of the Duke NPD QAPD. In Revision 3 of the Duke NPD QAPD, the applicant addressed the information related to Standard Content Open Item 17.5-5. The NRC staff has confirmed through review of Revision 3 of the Duke NPD QAPD that the applicant has incorporated the appropriate changes to Part II, Section 7.2, of the Duke NPD QAPD, which is consistent with the guidance contained in NRC-approved NEI 06-14A, Revision 7. This adequately addresses the issue outlined by Confirmatory Item 17.5-11; therefore, Standard Content Open Item 17.5-5 is resolved for the WLS COL application.

The following portion of this technical evaluation section is reproduced from Section 17.5.4.8 of the VEGP SER:

17.5.4.8 Identification and Control of Materials, Parts, and Components

The following portion of this technical evaluation section is reproduced from Section 17.5.4.8 of the BLN SER:

TVA's QAPD follows the guidance of Section 17.5 of NUREG-0800, paragraph II.H. The QAPD establishes the necessary measures for the identification and control of items such as materials, including consumables and items with limited shelf life, parts, components, and partially fabricated subassemblies. The identification of items is maintained throughout fabrication, erection, installation, and use so that the item can be traced to its documentation, consistent with the item's effect on safety.

In the QAPD, TVA commits to comply with the quality standards described in NQA-1-1994, Basic Requirement 8 and Supplement 8S-1, to establish provisions for identification and control of items.

The following portion of this technical evaluation section is reproduced from Section 17.5.4.9 of the VEGP SER:

17.5.4.9 Control of Special Processes

The following portion of this technical evaluation section is reproduced from Section 17.5.4.9 of the BLN SER:

TVA's QAPD follows the guidance of Section 17.5 of NUREG-0800, paragraph II.I. The QAPD establishes programs, procedures, and processes to ensure that special processes requiring interim process controls to ensure quality, such as welding, heat treating, chemical cleaning, and nondestructive examinations are implemented and controlled in accordance with applicable codes, specifications, and standards.

In the QAPD, TVA commits to comply with the quality standards described in NQA-1-1994, Basic Requirement 9 and Supplement 9S-1, to establish measures for the control of special processes.

The following portion of this technical evaluation section is reproduced from Section 17.5.4.10 of the VEGP SER:

17.5.4.10 Inspection

The following portion of this technical evaluation section is reproduced from Section 17.5.4.10 of the BLN SER:

TVA's QAPD follows the guidance of Section 17.5 of NUREG-0800, paragraph II.J. The QAPD establishes the necessary measures to implement inspections that ensure items, services, and activities affecting safety meet established requirements and conform to applicable documented specifications, instructions, procedures, and design documents. The inspection program establishes requirements for planning inspections, determining applicable acceptance criteria, setting the frequency of inspection, and identifying special tools needed to perform the inspection. Properly qualified personnel independent of those who performed or directly supervised the work are required to perform the inspections.

In the QAPD, TVA commits to comply with NQA-1-1994, Basic Requirement 10, Supplement 10S-1, and Subparts 2.4, 2.5, and 2.8, to establish inspection requirements, with the following commitment and alternative:

- *NQA-1-1994, Subpart 2.4, requires the use of the Institute of Electrical and Electronic Engineers (IEEE) Standard 336-1985, "IEEE Standard Installation, Inspection, and Testing Requirements for Power, Instrumentation, and Control Equipment at Nuclear Facilities." IEEE Standard 336-1985 refers to IEEE 498-1985, "IEEE Standard Requirements for the Calibration and Control of Measuring and Test Equipment Used in Nuclear Facilities." Each of these standards uses the definition of safety systems equipment from IEEE Standard 603-1980, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations." IEEE Standard 603-1980 defines "safety system" as:*

  > *Those systems (the reactor trip system, an engineered safety feature, or both, including all their auxiliary supporting features and other auxiliary feature) which provide a safety function. A safety system is comprised of*

more than one safety group of which any one safety group can provide the safety function.

The QAPD must commit to the definition of safety systems equipment from IEEE Standard 603-1980 to appropriately implement NQA-1-1994, Subpart 2.4. In the QAPD, TVA commits to the definition of safety systems equipment from IEEE Standard 603-1980, but does not commit to the balance of IEEE Standard 603-1980. This definition applies only to equipment in the context of Subpart 2.4. The NRC staff determined that the use of the definition of safety systems equipment is acceptable because it is consistent with the requirements of NQA-1-1994, Subpart 2.4.

- *NQA-1-1994, Supplement 10S-1, Section 3.1, states that inspection personnel shall not report to the immediate supervisor who is responsible for performing the work being inspected. As an alternative to this requirement, the QAPD proposes that QA inspectors will report to quality control management while performing such inspections. The NRC staff determined that the use of this alternative is consistent with guidance provided in Section 17.5 of NUREG-0800, paragraph II.J.1. Therefore, the NRC staff concluded that this alternative is acceptable.*

In a letter dated December 31, 2009, the VEGP applicant provided a markup of Revision 9 of the SNC QAPD that includes the alternative to NQA-1-1994, Supplement 10S-1, Section 3.1, discussed above. The NRC staff has reviewed the markup of SNC QAPD, Revision 9, and determined that the proposed changes are consistent with the alternative evaluated in the BLN SER. These items are identified as ***Confirmatory Item 17.5-12****, pending NRC review of the revised QAPD as referenced in Section 17.5 of the VEGP COL FSAR.*

Resolution of Standard Content Confirmatory Item 17.5-12

Confirmatory Item 17.5-12 is an applicant commitment to revise its QAPD. The staff verified that the VEGP COL application was appropriately updated. As a result, Confirmatory Item 17.5-12 is now closed.

WLS Resolution of Standard Content Confirmatory Item 17.5-12

By letter dated July 29, 2011, the applicant provided Revision 4 of the WLS COL FSAR and Revision 4 of the Duke NPD QAPD. In Revision 4 of the Duke NPD QAPD, the applicant addressed the information related to Standard Content Open Item 17.5-12. The NRC staff has confirmed through review of Revision 4 of the Duke NPD QAPD that the applicant has incorporated the appropriate changes to Part II, Section 10.3, of the Duke NPD QAPD, which is consistent with the guidance contained in NRC-approved NEI 06-14A, Revision 7. This adequately addresses the issue outlined by Confirmatory Item 17.5-12; therefore, Standard Content Confirmatory Item 17.5-12 is resolved for the WLS COL application.

The following portion of this technical evaluation section is reproduced from Section 17.5.4.11 of the VEGP SER:

17.5.4.11 Test Control

The following portion of this technical evaluation section is reproduced from Section 17.5.4.11 of the BLN SER:

TVA's QAPD follows the guidance of Section 17.5 of NUREG-0800, paragraph II.K. The QAPD establishes the necessary measures and governing provisions to demonstrate that items subject to the provisions of the QAPD will perform satisfactorily in service, that the plant can be operated safely as designed, and that the operation of the plant, as a whole, is satisfactory.

In the QAPD, TVA commits to comply with the quality standards described in NQA-1-1994, Basic Requirement 11 and Supplement 11S-1, to establish provisions for testing.

In the QAPD, TVA commits to comply with the quality standards described in NQA-1-1994, Supplement 11S-2 and Subpart 2.7, to establish provisions to ensure that computer software used in applications affecting safety be prepared, documented, verified, tested, and used such that the expected outputs are obtained and configuration control maintained.

The following portion of this technical evaluation section is reproduced from Section 17.5.4.12 of the VEGP SER:

17.5.4.12 Control of Measuring and Test Equipment

The following portion of this technical evaluation section is reproduced from Section 17.5.4.12 of the BLN SER:

TVA's QAPD follows the guidance of Section 17.5 of NUREG-0800, paragraph II.L. The QAPD establishes the necessary measures to control the calibration, maintenance, and use of measuring and test equipment that provide information important to safe plant operation.

In the QAPD, TVA commits to comply with the quality standards described in NQA-1-1994, Basic Requirement 12 and Supplement 12S-1, to establish provisions for control of measuring and test equipment, with the following clarification and exception:

- *The QAPD clarifies that the out-of-calibration conditions, described in paragraph 3.2 of Supplement 12S-1 of NQA-1-1994, refer to cases where the measuring and test equipment are found to be out of the required accuracy limits (i.e., out of tolerance) during calibration. The NRC staff determined that the clarification for the out-of-calibration conditions is consistent with Supplement 12S-1. Therefore, the NRC staff concluded that this clarification is acceptable.*

- *As an alternative to the NQA-1-1994, Subpart 2.4, Section 7.2.1, calibration labeling requirements, the QAPD proposes that, when it is impossible or impractical to mark equipment with required calibration information because of equipment size or configuration, the required calibration information will be documented and traceable to the equipment. The NRC staff determined that this alternative is consistent with NRC staff guidance provided in Section 17.5 of NUREG-0800, paragraph II.L.3. Therefore, the NRC staff concluded that this alternative is acceptable.*

The following portion of this technical evaluation section is reproduced from Section 17.5.4.13 of the VEGP SER:

17.5.4.13 <u>Handling, Storage, and Shipping</u>

The following portion of this technical evaluation section is reproduced from Section 17.5.4.13 of the BLN SER:

TVA's QAPD follows the guidance of Section 17.5 of NUREG-0800, paragraph II.M. The QAPD establishes the necessary measures to control the handling, storage, packaging, shipping, cleaning, and preservation of items to prevent inadvertent damage or loss and to minimize deterioration.

In the QAPD, TVA commits to comply with NQA-1-1994, Basic Requirement 13 and Supplement 13S-1, and to establish provisions for handling, storage, and shipping. In the QAPD, TVA also commits to comply with NQA-1-1994, Subparts 2.1 and 2.2 during the construction and pre-operations phase of the plant, as applicable, with the following alternative:

- *NQA-1-1994, Subpart 2.2, Section 6.6, states that the preparation of records must include information on personnel access to QA records. The QAPD establishes the necessary measures to document personnel authorized to access storage areas and recording personnel access. However, the QAPD proposes to not consider these documents as quality records. As an alternative, SNC will retain these documents in accordance with plant administrative controls. The NRC staff determined that these records do not meet the classification of a quality record as defined in NQA-1-1994, Supplement 17S-1, Section 2.7. Therefore, the NRC staff concluded that this alternative is acceptable.*

- *NQA-1-1994, Subpart 2.2, Section 7.1, refers to Subpart 2.15 for requirements related to handling of items. The QAPD clarifies that the scope of Subpart 2.15 includes hoisting, rigging and transporting of items for nuclear power plants during construction. The NRC staff has determined that this clarification is acceptable because it distinguishes between the requirements for construction and operation.*

NQA-1-1994, Subpart 3.2, Appendix 2.1, Section 3, provides cleaning recommendations and precautions. In a letter dated December 31, 2009, the VEGP applicant proposed a revision to the SNC QAPD to clarify that only the *precautions in Section 3 are committed to in accordance with RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," Revision 1. The NRC staff has determined that this clarification is acceptable because commitment to Subpart 3.2, Appendix 2.1, Section 3 is consistent with Regulatory Position 3 of RG 1.37. These items are identified as* ***Confirmatory Item 17.5-13****, pending NRC review of the revised QAPD as referenced in Section 17.5 of the VEGP COL FSAR.*

Resolution of Standard Content Confirmatory Item 17.5-13

Confirmatory Item 17.5-13 is an applicant commitment to revise its QAPD. The staff verified that the VEGP COL application was appropriately updated. As a result, Confirmatory Item 17.5-13 is now closed.

WLS Resolution of Standard Content Confirmatory Item 17.5-13

By letter dated July 29, 2011, the applicant provided Revision 4 of the WLS COL FSAR and Revision 4 of the Duke NPD QAPD. In Revision 4 of the Duke NPD QAPD, the applicant addressed the information related to Standard Content Open Item 17.5-13. The NRC staff has confirmed through review of Revision 4 of the Duke NPD QAPD that the applicant has incorporated the appropriate changes to Part II, Section 13.2, of the Duke NPD QAPD, which is consistent with the guidance contained in NRC-approved NEI 06-14A, Revision 7. This adequately addresses the issue outlined by Confirmatory Item 17.5-13; therefore, Standard Content Confirmatory Item 17.5-13 is resolved for the WLS COL application.

The following portion of this technical evaluation section is reproduced from Section 17.5.4.14 of the VEGP SER:

17.5.4.14 Inspection, Test, and Operating Status

The following portion of this technical evaluation section is reproduced from Section 17.5.4.14 of the BLN SER:

TVA's QAPD follows the guidance of Section 17.5 of NUREG-0800, paragraph II.N. The QAPD establishes the necessary measures to identify the inspection, test, and operating status of items and components subject to the provisions of the QAPD to maintain personnel and reactor safety and avoid inadvertent operation of equipment.

In the QAPD, TVA commits to comply with the quality standards described in NQA-1-1994, Basic Requirement 14, for identifying inspection, test, and operating status.

The following portion of this technical evaluation section is reproduced from Section 17.5.4.15 of the VEGP SER:

17.5.4.15 Nonconforming Materials, Parts, or Components

The following portion of this technical evaluation section is reproduced from Section 17.5.4.15 of the BLN SER:

TVA's QAPD follows the guidance of Section 17.5 of NUREG-0800, paragraph II.O. The QAPD establishes the necessary measures to control items, including services that do not conform to specified requirements to prevent inadvertent installation or use. Nonconformances are evaluated for their impact on operability of quality SSCs to ensure that the final condition does not adversely affect safety, operation, or maintenance of the item or service. The results of evaluations of conditions adverse to quality are analyzed to identify quality trends, documented, and reported to upper management in accordance with applicable procedures.

In addition, the QAPD provides for establishing the necessary measures to implement the requirements of Subparts A and C of 10 CFR Part 52, 10 CFR 50.55(e), and 10 CFR Part 21, as applicable.

In the QAPD, TVA commits to comply with the quality standards described in NQA-1-1994, Basic Requirement 15 and Supplement 15S-1, to establish measures for nonconforming material.

The following portion of this technical evaluation section is reproduced from Section 17.5.4.16 of the VEGP SER:

17.5.4.16 Corrective Action

The following portion of this technical evaluation section is reproduced from Section 17.5.4.16 of the BLN SER:

TVA's QAPD follows the guidance of Section 17.5 of NUREG-0800, paragraph II.P. The QAPD establishes the necessary measures to promptly identify, control, document, classify, and correct conditions adverse to quality. The QAPD requires personnel to identify known conditions adverse to quality. Reports of conditions adverse to quality are analyzed to identify trends. Significant conditions adverse to quality are documented and reported to responsible management. In the case of suppliers working on safety-related activities or similar situations, TVA may delegate specific responsibility for the corrective action program, but TVA maintains responsibility for the program's effectiveness.

In addition, the QAPD provides for establishing the necessary measures to implement a reporting program in accordance with the requirements of 10 CFR Part 21.

In the QAPD, TVA commits to comply with the quality standards described in NQA-1-1994, Basic Requirement 16, to establish a corrective action program.

The following portion of this technical evaluation section is reproduced from Section 17.5.4.17 of the VEGP SER:

17.5.4.17 Quality Assurance Records

The following portion of this technical evaluation section is reproduced from Section 17.5.4.17 of the BLN SER:

TVA's QAPD follows the guidance of Section 17.5 of NUREG-0800, paragraph II.Q. The QAPD establishes the necessary measures to ensure that sufficient records of items and activities affecting quality are generated, identified, retained, maintained, and retrievable.

Concerning the use of electronic records storage and retrieval systems, the QAPD complies with the NRC guidance given in RIS 2000-18, "Guidance on Managing Quality Assurance Records in Electronic Media," dated October 23, 2000, and associated Nuclear Information and Records Management Association (NIRMA) guidelines TG 11-1998, TG 15-1998, TG 16-1998 and TG 21-1998.

In the QAPD, TVA commits to comply with the quality standards described in NQA-1-1994, Basic Requirement 17 and Supplement 17S-1, to establish provisions for records, with the following alternative:

- *NQA-1-1994, Supplement 17S-1, Section 4.2(b) states that records must be firmly attached in binders or placed in folders or envelopes for storage in steel file cabinets or on shelving in containers. As an alternative to this requirement, the QAPD proposes that hard-copy records be stored in steel cabinets or on shelving in containers, except that methods other than binders, folders, or envelopes may be used to organize records for storage. The NRC staff determined that this alternative is acceptable as documented in an SER dated September 1, 2005 for Nuclear Management Company.*

The following portion of this technical evaluation section is reproduced from Section 17.5.4.18 of the VEGP SER:

17.5.4.18 Quality Assurance Audits

The following portion of this technical evaluation section is reproduced from Section 17.5.4.18 of the BLN SER:

TVA's QAPD follows the guidance of Section 17.5 of NUREG-0800, paragraph II.R. The QAPD establishes the necessary measures to implement audits to verify that activities covered by the QAPD are performed in conformance with documented requirements. The audit program is reviewed for effectiveness as part of the overall audit process.

The QAPD provides for the applicant or holder to conduct periodic internal and external audits. Internal audits are conducted to determine that the program and *procedures being audited comply with the QAPD. Internal audits, conducted after placing the facility in operation, are performed with a frequency commensurate with safety significance and in such a manner as to ensure that an audit of all applicable QA program elements is completed for each functional area within a period of 2 years. External audits determine the adequacy of a supplier's or contractor's QA program.*

TVA ensures that audits are documented and reviews audit results. TVA responds to all audit findings and initiates appropriate corrective actions. In addition, where corrective actions are indicated, TVA documents follow-up of applicable areas through inspections, review, re-audits, or other appropriate means to verify implementation of assigned corrective actions.

In the QAPD, TVA commits to comply with the quality standards described in NQA-1-1994, Basic Requirement 18 and Supplement 18S-1, to establish the independent audit program.

The following portion of this technical evaluation section is reproduced from Section 17.5.4.19 of the VEGP SER:

17.5.4.19 *Nonsafety-Related SSCs Quality Assurance Control*

17.5.4.19.1 Nonsafety-Related SSCs - Significant Contributors to Plant Safety

The following portion of this technical evaluation section is reproduced from Section 17.5.4.19.1 of the BLN SER:

TVA's QAPD follows the guidance of Section 17.5 of NUREG-0800, paragraph II.V.1. The QAPD establishes program controls applied to nonsafety-related SSCs that are significant contributors to plant safety and to which Appendix B does not apply. The QAPD applies specific controls to these items in a selected manner, targeting the characteristics or critical attributes that render the SSC a significant contributor to plant safety consistent with applicable sections of the QAPD.

The Duke NPD QAPD incorporates the guidance of Section 17.5 of NUREG-0800, Paragraph II.V.1 that establishes specific, targeted program controls for nonsafety related SSCs, and is therefore acceptable to the staff.

17.5.4.19.2 Nonsafety-Related SSCs Credited for Regulatory Events

The Duke NPD QAPD follows the guidance of Section 17.5 of NUREG-0800, Paragraph II.V.2, to establish the quality requirements for nonsafety-related SSCs credited for regulatory events. In the QAPD, Duke Energy commits to comply with the following regulatory guidance:

- Duke Energy implements quality requirements for the fire protection system in accordance with Regulatory Position 1.7, "Quality Assurance," in RG 1.189, "Fire Protection for Operating Nuclear Power Plants," issued April 2001.

- Duke Energy implements the quality requirements for anticipated transient without scram (ATWS) equipment in accordance with Part III, Section 1, of the Duke NPD QAPD.

- Duke Energy implements quality requirements for station blackout (SBO) equipment in accordance with Part III, Section 1, of the Duke NPD QAPD. Regulatory Guide 1.155, "Station Blackout," dated August 1988, is not applicable for the AP1000 design in accordance with the certified design as shown in the DCD, Appendix 1A. Regulatory Guide 1.155 relates to the availability of safety related functions supported by AC power. Since AC power is not required to support the availability of safety-related functions, the guidance is not applicable to the WLS COL application.

17.5.4.20 *Regulatory Commitments*

The Duke NPD QAPD follows the guidance of Section 17.5 of NUREG-0800, Paragraph II.U. The QAPD establishes QA program commitments. By letter dated December 17, 2010, Duke Energy submitted Revision 3 of the WLS COL FSAR as well as Revision 3 of the Duke NPD QAPD. In Part IV of the Duke NPD QAPD, Duke Energy commits to comply with the following NRC regulatory guides and other QA standards to supplement and support the QAPD:

- RG 1.8, “Qualification and Training of Personnel for Nuclear Power Plants,” Revision 3.

- RG 1.26, “Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” Revision 4.

- RG 1.28, “Quality Assurance Program Requirements (Design and Construction),” Revision 3.

- RG 1.29, “Seismic Design Classification,” Revision 4.

- RG 1.33, “Quality Assurance Program Requirements (Operations),” Revision 2.

- RG 1.37, “Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components for Water-Cooled Nuclear Power Plants,” Revision 1.

- ASME NQA-1-1994 Edition, Parts I, II, and III.

- Nuclear Information and Records Management Association, Inc. (NIRMA) technical guides, as described in Part II, Section 17 of the Duke NPD QAPD.

The following portion of this technical evaluation section is reproduced from Section 17.5.4.20 of the VEGP SER:

> *In RAI 17.5-15 dated May 12, 2008, the NRC staff requested that the applicant revise the TVA QAPD Part IV to commit to RG 1.37 Revision 1, “Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants,” issued March 2007. In its response dated June 24, 2008, the applicant stated that Part IV of the TVA QAPD is consistent with Revision 4 of NEI 06-14A. In a letter, dated September 17, 2008, the NRC staff requested NEI to address this question as*

part of Revision 5 to NEI 06-14A. However, the applicant committed to RG 1.37, Revision 1, in Revision 1 of the BLN QAPD. RAI 17.5-15 is closed.

In a letter dated December 31, 2009, the VEGP applicant provided a markup of Revision 9 of the SNC QAPD. The NRC staff has reviewed the markup of SNC QAPD, Revision 9, and determined that conforming changes have been proposed to Part IV consistent with NEI 06-14A, Revision 7. On this basis, the updating of the SNC QAPD for closure of standard content RAI 17.5-15 is ***Confirmatory Item 17.5-16*** *for the VEGP COL application.*

Resolution of Standard Content Confirmatory Item 17.5-16

Confirmatory Item 17.5-16 is an applicant commitment to revise its QAPD. The staff verified that the VEGP COL application was appropriately updated. As a result, Confirmatory Item 17.5-16 is now closed.

WLS Resolution of Standard Content Confirmatory Item 17.5-16

In a letter dated November 4, 2010, the applicant endorsed the standard content material provided by VEGP. By letter dated December 17, 2010, the applicant provided Revision 3 of the WLS COL FSAR and Revision 3 of the Duke NPD QAPD. In Revision 3 of the Duke NPD QAPD, the applicant addressed the information related to Standard Content Open Item 17.5-16. The NRC staff has confirmed through review of Revision 3 of the Duke NPD QAPD that the applicant has incorporated the appropriate changes to Part IV of the Duke NPD QAPD, which is consistent with the guidance contained in NRC-approved NEI 06-14A, Revision 7. This adequately addresses the issue outlined by Confirmatory Item 17.5-16; therefore, Standard Content Confirmatory Item 17.5-16 is resolved for the WLS COL application.

The following portion of this technical evaluation section is reproduced from Section 17.5.4.20 of the VEGP SER:

The NRC staff also reviewed Appendix 1AA of the BLN COL FSAR, which lists BLN's conformance with NRC RGs and provides any exceptions to conformance with those RGs. In RAI 17.5-17, the NRC staff requested that the applicant explain how the QAPD provides an acceptable exception to the RGs described in Appendix 1AA. In its response (ML081780171), the applicant stated that Part IV of the TVA QAPD is consistent with Revision 4 of NEI 06-14A. Additionally, the applicant provided further information addressing these RGs in response to RAIs 17.5-15 and 17.5-17. The response to RAI 17.5-15 proposed revisions to Appendix 1AA and Parts II and *IV of the QAPD, whereas the response to RAI 17.5-17 provided further justification. The applicant provided a response to RAI 1-5 in a letter dated August 19, 2008, to address the discrepancies between the revisions of the RGs addressed in Appendix 1AA and those addressed in Westinghouse DCD Appendix 1A. The information in this letter appears to have superseded the changes that were proposed and acceptable to the NRC staff in the applicant's June 24, 2008 letter, thereby reopening the issue identified in RAI 17.5-17. This is identified as* ***Open Item 17.5-6****.*

Resolution of Standard Content Open Item 17.5-6

In a letter dated July 29, 2009, the VEGP applicant stated that the revisions to the COL application identified in the referenced TVA August 19, 2008, letter do supersede the changes identified in the referenced TVA June 24, 2008, letter, as shown in Revision 1 of the BLN COL application. In a letter dated December 31, 2009, the VEGP applicant proposed additional changes to FSAR Chapter 1, Appendix 1AA to address conformance to RG 1.33, Revision 2. The NRC staff has reviewed the proposed changes to VEGP COL FSAR Chapter 1, Appendix 1AA, and determined that the changes are responsive to RAI 17.5-17. On this basis, Open Item 17.5-6 is ***Confirmatory Item 17.5-17*** *for the VEGP COL application*.

Resolution of Standard Content Confirmatory Item 17.5-17

Confirmatory Item 17.5-17 is an applicant commitment to revise its FSAR Appendix 1AA. The staff verified that the VEGP COL FSAR was appropriately updated. As a result, Confirmatory Item 17.5-17 is now closed.

WLS Resolution of Standard Content Open Item 17.5-6 and Confirmatory Item 17.5-17

In a letter dated November 4, 2010, the applicant endorsed the standard content material provided by VEGP. By letter dated December 17, 2010, the applicant provided Revision 3 of the WLS COL FSAR and Revision 3 of the Duke NPD QAPD. In Revision 3 of the Duke NPD QAPD, the applicant addressed the information related to Standard Content Open Item 17.5-6, in regard to applicability of the RGs identified in Part IV of the Duke NPD QAPD and Appendix 1AA of the WLS COL FSAR.

The NRC staff has confirmed through review of Revision 3 of the Duke NPD QAPD and Appendix 1AA of the WLS COL FSAR that the applicant has identified and conforms to Regulatory Guides 1.8, 1.26, 1.29, 1.33, and 1.37. With respect to RG 1.28, the applicant identifies conformance with RG 1.28 for the DCD scope of work, and commits to ASME NQA-1-1994, Parts I, II, and III, in lieu of a commitment to RG 1.28 for the remaining scope of work. This is consistent with the guidance contained in NRC-approved NEI 06-14A, Revision 7. The NRC staff determined that the revisions to the Duke NPD QAPD and Appendix 1AA of the WLS COL FSAR adequately address the issue outlined by Confirmatory Item 17.5-17; therefore, Standard Content Open Item 17.5-6 and Confirmatory Item 17.5-17 are resolved for the WLS COL application.

17.5.5 Post Combined License Activities

There are no post COL activities related to this section.

17.5.6 Conclusion

The NRC staff used the requirements of Appendix B to10 CFR Part 50 and the guidance of Section 17.5 of NUREG-0800 as the basis for evaluating the acceptability of the Duke NPD QAPD and concludes that:

- The QAPD provides adequate guidance for Duke Energy to describe the authority and responsibility of management and supervisory personnel, performance / verification personnel, and self-assessment personnel.

- The QAPD provides adequate guidance for Duke Energy to provide for organizations and persons to perform verification and self-assessment functions with the authority and independence to conduct their activities without undue influence from those directly responsible for costs and schedules.

- The QAPD provides adequate guidance for Duke Energy to apply a QAPD to activities and items that are important to safety.

- The QAPD provides adequate guidance for Duke Energy to establish controls that, when properly implemented, comply with the requirements of 10 CFR Part 52; Appendix B to 10 CFR Part 50; 10 CFR Part 21; and 10 CFR 50.55(e), with the acceptance criteria associated with Section 17.5 of NUREG-0800, and with the commitments to applicable regulatory guidance.

The Duke NPD QAPD addresses WLS COL 17.5-1, STD COL 17.5-2, STD COL 17.5-4, and STD COL 17.5-8.

Based on the information provided by the applicant and the evaluation above, the staff concludes that Section 17.5 of the WLS COL FSAR and the Duke NPD QAPD meet the requirements of Appendix B to 10 CFR Part 50; 10 CFR 52.79(a)(17); 10 CFR 52.79(a)(25); and 10 CFR 52.79(a)(27).

17.6 Maintenance Rule Program (Related to RG 1.206, Section C.III.1, Chapter 17, C.I.17.6, "Description of the Applicant's Program for Implementation of 10 CFR 50.65, The Maintenance Rule")

17.6.1 Introduction

This section addresses the program for maintenance rule (MR) implementation. It is based on the requirements of 10 CFR Part 52 and the guidance provided to the industry by the Nuclear Management and Resources Council (NUMARC) and its successor, the NEI. NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," is endorsed by the NRC in RG 1.160, "Design Response Spectra for Seismic Design of Nuclear Power Plants." Section 11.0 of NUMARC 93-01 was later revised; the revision, as modified by RG 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants," is also endorsed by the NRC. NEI 07-02A, "Generic FSAR Template Guidance for Maintenance Rule Program Description for Plants Licensed Under 10 CFR Part 52," provides a template for presenting this information that has also been endorsed by the NRC staff in a letter addressed to NEI dated January 24, 2008.

17.6.2 Summary of Application

In Section 17.6, "Maintenance Rule Program," of the WLS COL FSAR, the applicant provided the following:

AP1000 COL Information Items

- STD COL 3.8-5

The applicant provided additional information in STD COL 3.8-5 to address COL Information Item 3.8-5. STD COL 3.8-5 addresses the inspection program for structures.

Supplemental Information

- STD SUP 17.6-1

The applicant provided additional information in STD SUP 17.6-1 to incorporate by reference NEI 07-02A. The applicant also identified where operational programs are described in the WLS COL FSAR, including a description of and milestones for the maintenance rule program.

- STD SUP 17.6-2

The applicant provided additional information in STD SUP 17.6-2 to incorporate condition monitoring of underground or inaccessible cables into the maintenance rule program.

License Condition

- Part 10, License Condition 6, "Operational Program Readiness"

This license condition states that the COL holder shall provide an operational program schedule to support NRC inspections.

17.6.3 Regulatory Basis

Commission regulations for the maintenance rule program include the requirements of 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," and 10 CFR 52.79(a)(15). The acceptance criteria associated with the relevant requirements of the NRC regulations for the MR are given in Section 17.6 of NUREG-0800.

The regulatory basis of the information incorporated by reference into WLS COL FSAR Section 17.6 is addressed in NUREG-1793 and its supplements for topical report NEI 07-02, Revision 3, which was transmitted to NEI in a letter from the NRC staff dated January 24, 2008.

SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," identifies schedule requirements and proposes a license condition to be satisfied by COL holders.

17.6.4 Technical Evaluation

The NRC staff reviewed Section 17.6 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the MR. The results of the staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the NRC staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The NRC staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the BLN Units 3 and 4 COL application. Any confirmatory items in the standard content material retain the numbers assigned in the VEGP SER.

AP1000 COL Information Items

- STD COL 3.8-5

The NRC staff reviewed STD COL 3.8-5 in WLS COL FSAR Section 17.6 related to the inspection program for structures. The staff's review of this program is documented in Section 3.8 of this SER.

The following portion of this technical evaluation section is reproduced from Section 17.6.4 of the VEGP SER:

> *The NRC staff reviewed conformance of Section 17.6 of the BLN COL FSAR, including the COL standard information item identified in Subsection 17.6.2, with the guidance in NUREG-0800, Section 17.6. The staff also compared it with RG 1.206, Section C.III.1, Chapter 17, C.I.17.6, "Description of the Applicant's Program for Implementation of 10 CFR 50.65, the Maintenance Rule."*
>
> *In addition, the NRC staff reviewed the COL standard information item identified in Subsection 17.6.2 above. In its review, the staff used NUREG-0800, Section 17.6, "Maintenance Rule," as guidance.*

<u>*Supplemental Information*</u>

- *STD SUP 17.6-1, which incorporated NEI 07-02A and identified where operational programs are described in the BLN COL FSAR, including a description of the MR program*

The applicant added the following text to Section 17.6 of the BLN COL FSAR:

> *This section incorporates by reference NEI 07-02A, "Generic FSAR Template Guidance for Maintenance Rule Program Description for Plants Licensed under 10 CFR Part 52," with the following supplemental information. See Table 1.6-201.*
>
> *Table 13.4-201 provides milestones for maintenance rule [MR] program implementation.*

The applicant indicated where, in the BLN COL FSAR, the programs listed in Subsection 17.X.3 of NEI 07-02A are described:

- *MR program (Section 17.6)*
- *QA program (Section 17.5)*
- *inservice inspection program (Sections 5.2 and 6.6)*
- *inservice testing program (Section 3.9)*
- *technical specifications surveillance test program (Chapter 16)*

The NRC staff endorsed NEI 07-02A, stating that it provides an acceptable method:

- *for complying with the requirement in 10 CFR 52.79(a)(15) that FSARs contain a description of the program and its implementation*

- *for monitoring the effectiveness of maintenance to meet the requirements of Section 50.65*

- *for satisfying the acceptance criteria of NUREG-0800, Section 17.6*

Because STD SUP 17.6-1 incorporates NEI 07-02A by reference and identifies the relevant operational programs and milestones, the staff finds that the applicant has provided sufficient information to fully describe the maintenance rule program. This provides reasonable assurance that the program, when implemented, satisfies the requirements of 10 CFR 50.65.

- *STD SUP 17.6-2*

In response to RAI 8.2-14, the applicant incorporated cable monitoring into its maintenance rule program. The program will monitor the condition of inaccessible or underground cables, including all those that support SSCs within the scope of 10 CFR 50.65. The staff documented its evaluation of the cable monitoring program in SER Section 8.2.4.

License Condition

- *Part 10, License Condition 6*

The applicant proposed a license condition to provide a schedule to support NRC inspection of operational programs including the MR program. The proposed license condition is consistent with the policy established in SECY-05-0197 and is acceptable.

17.6.5 Post Combined License Activities

For the reasons discussed in the technical evaluation section above, the NRC staff finds the following license condition acceptable:

- License Condition (17-1) – No later than 12 months after issuance of the COL, the licensee shall submit to the Director of the Office of New Reactors, a schedule that supports planning for and conduct of NRC inspections of the Maintenance Rule (MR) program. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until the MR has been fully implemented.

17.6.6 Conclusion

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to the MR program. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

STD SUP 17.6-1 incorporated NEI 07-02A by reference, identified where operational programs are described in the WLS COL FSAR, including a description of the MR program, and provided a schedule for implementation of the MR program. STD SUP 17.6-2 incorporated condition monitoring of underground or inaccessible cables into the maintenance rule program.

Based on its evaluation, the NRC staff concludes that the relevant information presented in Section 17.6 of the WLS COL FSAR meets the requirements of 10 CFR 50.65 and 10 CFR 52.79(a)(15), and is, therefore, acceptable.

18.0 HUMAN FACTORS ENGINEERING

18.1 Overview (No Corresponding Section in Regulatory Guide (RG) 1.206)

Section 18.1 of the William States Lee III Nuclear Station (WLS) combined license (COL) Final Safety Analysis Report (FSAR), Revision 11, incorporates by reference, with no departures or supplements, Section 18.1 of Revision 19 of the AP1000 Design Control Document (DCD). The Nuclear Regulatory Commission (NRC) staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793, "Final Safety Evaluation Report [FSER] Related to Certification of the AP1000 Standard Design," and its supplements.

18.2 Human Factors Engineering Program Management (Related to RG 1.206, Section C.I.18.1, "HFE Program Management")

18.2.1 Introduction

The Human Factors Engineering (HFE) Program Management plan describes the HFE program in sufficient detail to ensure that all aspects of the human-system interfaces (HSIs), procedures, staffing, and training are developed, designed, and evaluated on the basis of a structured top-down systems analysis using accepted HFE guidance.

18.2.2 Summary of Application

Section 18.2 of the WLS COL FSAR, Revision 11, incorporates by reference Section 18.2 of the AP1000 DCD, Revision 19.

In addition, in WLS COL FSAR Section 18.2.1.3, the applicant provided the following:

AP1000 COL Information Item

- WLS COL 18.2-2

The applicant provided additional information in WLS COL Information Item 18.2-2, addressing Emergency Operating Facility (EOF) and Technical Support Center (TSC) communications and HFE design.

License Condition

- License Condition 1, regarding the HFE inspections, tests, analyses and acceptance criteria (ITAAC).

[1] See Section 1.2.2 for a discussion of the staff's review related to verification of the scope of information to be included in a COL application that references a design certification (DC).

18.2.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for WLS COL 18.2-2 are given in Chapter 18 of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants."

The applicable regulatory requirements for WLS COL 18.2-2 are as follows:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 52.79(c)

The related acceptance criteria are as follows:

- NUREG-0711, "Human Factors Engineering Program Review Model," Revision 2, Section 2.4

- NUREG-0696, "Functional Criteria for Emergency Response Facilities"

18.2.4 Technical Evaluation

The NRC staff reviewed Section 18.2 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the HFE program management. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this safety evaluation report (SER) provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (Vogtle Electric Generating Plant [VEGP], Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from requests for additional information (RAIs).

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is

identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) contains evaluation material from the SER for the Bellefonte Nuclear Plant (BLN), Units 3 and 4 COL application.

The staff reviewed the information in the WLS COL FSAR:

AP1000 COL Information Item

- WLS COL 18.2-2

The WLS COL FSAR states that the EOF and TSC communications strategies and EOF and TSC human factors attributes are addressed in the emergency plan. The WLS Emergency Plan, Appendix 9, "Justification for Common EOF," states that the EOF meets all functional and design criteria provided in NUREG-0696 for an EOF with the exception of the EOF location. This captures all the guidance related to the HFE design. The TSC description contained in Appendix 10, "Technical Support Center Description" does not contain a similar statement and the description of the HFE design of the TSC is a generalization of the NUREG guidance. For example:

- The size of the TSC is described as:

  > The TSC provides working space, without crowding, for the personnel assigned to the TSC at the maximum level of occupancy. The working space is sized for a minimum of 25 persons. Minimum size of working space is exceeds [sic] 75 square feet per person.

  NUREG-0696 includes more specific guidance addressing space for TSC data system equipment, space for performing repair and maintenance activities, space for personnel access to functional displays, and space for unhindered access to communications equipment. HFE design includes these factors as part of the layout design for a control center.

- The Technical data and data system description for the TSC includes the following statement on HFE design:

  > Human Factors Engineering (HFE) is incorporated into the design of the TSC related to the display and availability of plant data.

  NUREG-0696 includes more specific guidance addressing trending capability and display characteristics.

The general statements contained in the appendix were not sufficient for the staff to draw a conclusion on the TSC HFE design. In RAI 18-1, the staff requested the applicant specifically identify the TSC HFE design requirements or reference the NUREG directly as is done for the EOF. In a letter dated December 14, 2011 the applicant proposed to add a paragraph to Appendix 10 of its Emergency Plan which explicitly states that the TSC meets the functional and design criteria provided in NUREG-0696, with the exception of the location of the TSC, as described and justified in departure WLS 18.8-1, "Emergency Response Facility Locations."

The staff found this change acceptable as it more clearly commits to meet the acceptable methods and criteria given in NUREG-0696 to meet regulatory requirements. The acceptability of WLS DEP 18.8-1 is addressed in Chapter 13 of this report. The inclusion of the proposed addition to the Emergency Plan was tracked as Confirmatory Item 18.2-1.

Resolution of Confirmatory Item 18.2-1

Confirmatory Item 18.2-1 is an applicant commitment to revise the WLS Emergency Plan to incorporate its response to RAI 18-1. The staff verified that Revision 3 of the WLS Emergency Plan reflected the change, as stated in their December 14, 2011 letter. As a result, Confirmatory Item 18.2-1 is now closed.

The WLS emergency plan, Appendix 9, "Justification for Common EOF," states that the WLS EOF will use the same facility shared by McGuire Nuclear Station, Catawba Nuclear Station, and Oconee Nuclear Station. Appendix 10, "Technical Support Center Description," states that WLS nuclear units 1 and 2 will share a common TSC. In RAI 18-2, the staff requested the applicant describe how data is differentiated between the different nuclear units**.** In a December 14, 2011 RAI response, the applicant explained that unit or facility specific information provided in the TSC and EOF include an on-screen identifier (whether projected on large screens for facility access or on individual monitors) that identifies the source facility/unit for the information. Duke Energy's operating fleet, which currently consists of three multiple unit sites with common TSCs and a common EOF for all sites, has utilized this method of data differentiation for a number of years with repeated success. The staff concludes that data differentiation between units and facilities is acceptable based on the use of accepted HFE principles and the success of these principles demonstrated by operating experience. As a result, RAI 18-2 is closed**.**

The effectiveness of human factors attributes and communications must be demonstrated as part of ITAAC closure for Emergency Planning following the same protocol as applied to the referenced COL. This protocol is described below.

The following portion of this technical evaluation section is reproduced from Section 18.2.4 of the VEGP SER:

> *The following portion of this technical evaluation section is reproduced from Section 18.2.4 of the BLN SER:*
>
> *In its September 2, 2008, response to RAI 18-3, the applicant stated that the scope of the HFE design includes implementation and verification of applicable EOF/Technical Support Center (TSC) displays consistent with the AP1000 HFE program. TR-136* [Technical Report] *(APP-GW-GLR-136, Revision 1, "AP1000 Human Factors Program Implementation for the Emergency Operations Facility and Technical Support Center") indicates that the Westinghouse DCD does not cover all aspects of the HSI design (such as panel layouts, room configuration, and indications/controls) for the EOF/TSC. The applicant states that the EOF/TSC functions and tasks that are not within the scope of the AP1000 HFE Program will be subject to HFE principles and practices as described in NUREG-0737, "Clarification of TMI [Three Mile Island] Action Plan Requirements."*

The staff was concerned that, since NUREG-0737 does not have HFE guidance comparable to that of NUREG-0711, EOF/TSC design elements would fall outside the scope of the HFE program. The applicant addressed this concern in its RAI 18-4 response dated February 23, 2009, stating that the HSI design will meet the data and availability criteria in NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, Section II.H, 'Emergency Facilities and Equipment,'" which states that the TSC and the EOF will be established in accordance with NUREG-0696.

The staff agrees that NUREG-0696 describes an acceptable method for meeting EOF/TSC requirements and contains guidance for managing the EOF/TSC HFE design based on the following:

- *NUREG-0696, Section 2.8, states, "The design of the TSC data system equipment shall incorporate human factors engineering with consideration for both operating and maintenance personnel."*

- *NUREG-0696, Section 4.7, states, "The design of the EOF data system equipment shall incorporate human-factors engineering with consideration for both operating and maintenance personnel."*

- *NUREG-0696, Section 4.8, states, "Human-factors engineering shall be incorporated in the design of the EOF." This section of the NUREG also addresses data availability and human factors design criteria.*

- *The AP1000 DCD includes a structured approach for identifying data needed to support the EOF/TSC functions.*

- *The guidance in NUREG-0696 addresses information usability. While some guidance is generic, the staff concludes APP-OCS-J1-002, "AP1000 HSI Design Guidelines," which is included by reference in Chapter 18 of the AP1000 DCD, is applicable to the definition of more explicit, measurable design acceptance criteria. Use of these guidelines will ensure that general design principles, such as "callup, manipulation, and presentation of data can be easily performed," and, "display formats shall present information so that it can be easily understood," will be subject to more explicit design acceptance criteria.*

Emergency planning drills and inspections provide repeated opportunities to identify improvements to HSIs. In the case of BLN, for which a common EOF will be used, EOF design improvements have already been implemented based on operating experience.

HFE design verification and validation (V&V) is a second area of NUREG-0711 guidance that is not being directly applied by the applicant. As an alternative, the applicant states in their RAI 18-4 response dated February 23, 2009, that V&V of the EOF HFE design is achieved by the evaluation of equipment and personnel performance during drills and exercises. The staff concludes that although the *specific guidance in NUREG- 0711 for V&V is not being applied, the alternative V&V approach provides reasonable assurance that the HFE aspects of the EOF and TSC will be acceptably designed based on the following:*

- *NUREG-0696 contains guidance on V&V. Section 9 states, "The design, development, qualification, and installation of the SPDS [safety parameter display system], TSC, EOF, and NDL [nuclear data link] facilities and systems shall be independently verified and validated by qualified personnel other than the original designers and developers."*

  The RAI 18-4 response indicates both equipment and personnel performance will be evaluated during drills and exercises.

- *Exercises and drills are conducted on a periodic basis, and therefore, provide repeated opportunities to test and improve the HSIs.*

- *The first exercise is included as an inspection, test, analysis and acceptance criterion (ITAAC) that ensures EOF/TSC functionality prior to fuel load. The BLN COL application Part 10, "Proposed License Conditions," Revision 1, Table 3.8-1, ITAAC contain the following inspections, tests and analyses:*

  > *ITAAC 1.1: An inspection of the control room, TSC, and CECC [Central Emergency Control Center] will be performed to verify that they have displays for retrieving facility system and effluent parameters in specific emergency action levels (EALs).*
  >
  > *ITAAC 8.1: A full-participation exercise (test) will be conducted within the specified time periods of Appendix E to 10 CFR Part 50.*

- *Exercises and drills are conducted in the actual facilities, (vice a simulator), allowing direct observation of the HSI.*

<u>Evaluation of Site-Specific Information Related to Standard Content</u>

WLS COL Application, Part 10, "Proposed License Conditions," Table 3.8-1 includes the following relevant site-specific ITAAC for WLS Units 1 and 2 that addresses a verification inspection to ensure functionality of the EOF, and TSC prior to fuel load:

> ITAAC 1.1: An inspection of the Control Rooms, Technical Support Center (TSC), and Emergency Operations Facility (EOF) will be performed to verify that they have displays for retrieving facility system and effluent parameters that are specified in the Emergency Classification and EAL scheme and the displays are functional.
>
> ITAAC 8.1: A full-participation exercise (test) will be conducted within the specified time periods of Appendix E, ["Emergency Planning and Preparedness for Production and Utilization Facilities"] to 10 CFR Part 50.

The staff found that WLS ITAAC 1.1 and WLS ITAAC 8.1 were comparable to those proposed by VEGP and concluded that the site-specific ITAAC provided an acceptable V&V approach to ensure functionality of the control room, EOF, and TSC from an HFE perspective. Therefore, the conclusions reached by the NRC staff related to VEGP COL 18.2-2 are directly applicable to the WLS COL application. The evaluation of these ITAAC from an emergency planning perspective is addressed in SER Section 13.3.

18.2.5 Post Combined License Activities

For the reason discussed in the technical evaluation section above, the staff proposes to include the following ITAAC proposed by the applicant to ensure functionality of the control room, EOF, and TSC from an HFE perspective.

- The Licensee shall perform the following ITAAC:
  - ITAAC 1.1: An inspection of the Control Rooms, Technical Support Center (TSC), and Emergency Operations Facility (EOF) will be performed to verify that they have displays for retrieving facility system and effluent parameters that are specified in the Emergency Classification and EAL scheme and the displays are functional.
  - ITAAC 8.1: A full-participation exercise (test) will be conducted within the specified time periods of Appendix E to 10 CFR Part 50.

18.2.6 Conclusion

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to HFE program management, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the requirements of 10 CFR 52.79, "Contents of applications; technical information in final safety analysis report," and meets the guidance in Chapter 18 of NUREG-0800. The staff based its conclusion on the following:

- WLS COL 18.2-2 is acceptable because the applicant will design the EOF/TSC in accordance with appropriate elements of the AP1000 HFE program and NUREG-0696.

18.3 Operating Experience Review (Related to RG 1.206, Section C.I.18.2, "Operating Experience Review")

Operating experience review (OER) identifies and analyzes HFE-related problems and issues in previous designs. In this way, negative features associated with previous designs may be avoided in the current one, while retaining positive features. This section describes the applicant's OER and how it was used to identify HFE-related safety issues. OER includes a summary discussion of the source materials, such as documents, event reports, and personnel interviews. OER-identified issues are included along with their resolution.

Section 18.3 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 18.3 of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

18.4 Functional Requirements Analysis and Allocation (Related to RG 1.206, Section C.I.18.3, "Functional Requirements Analysis and Function Allocation")

Functional requirements analysis and function allocation demonstrate that functions are allocated to human and system resources in a manner that takes advantage of human strengths and avoids human limitations. The scope includes identification and analysis of those functions that must be performed to satisfy the plant's safety objectives that is, to prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public.

Section 18.4 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 18.4 of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

18.5 AP1000 Task Analysis Implementation Plan (Related to RG 1.206, Section C.I.18.4, "Task Analysis")

Task analyses identify the specific tasks that are needed for function accomplishment and their information, control, and task support requirements. The analyses address how representative and important operations, maintenance, test, inspection, and surveillance tasks are selected, as well as the range of operating modes included in the analyses. This includes the use of probabilistic risk assessment (PRA)/human reliability analysis (HRA) for the identification of the risk-important human actions, including the monitoring and backup of automatic actions. The task analysis results are used as input to the design of HSIs, procedures, and training programs.

Section 18.5 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 18.5 of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

18.6 Staffing (Related to RG 1.206, Section C.I.18.5, "Staffing and Qualifications")

18.6.1 Introduction

Staffing and qualification analyzes the requirements for the number and qualifications of personnel in a systematic manner that includes a thorough understanding of task requirements and applicable regulatory requirements.

This section is coordinated with Section 13.1 of this SER, which also relates to organization and staffing. The staffing analysis is iterative in nature and discusses how the initial staffing goals have been reviewed and modified as the analyses associated with other HFE elements are complete. Staffing and qualifications are also shown to be in compliance with 10 CFR 50.54(m).

18.6.2 Summary of Application

Section 18.6 of the WLS COL FSAR, Revision 11, incorporates by reference Section 18.6 of the AP1000 DCD, Revision 19.

In addition, in WLS COL FSAR Section 18.6, the applicant provided the following:

AP1000 COL Information Item

- STD COL 18.6-1

The applicant provided additional information in Standard (STD) COL 18.6-1 to resolve COL Information Item 18.6-1, addressing staffing levels and qualifications of plant personnel.

18.6.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for STD COL 18.6-1 are given in Chapter 18 of NUREG-0800.

The applicable regulatory requirements for STD COL 18-1 are as follows:

- 10 CFR 52.79(c)
- 10 CFR 50.54(m)

The related acceptance criterion is as follows:

- NUREG-0711, Section 6.4

18.6.4 Technical Evaluation

The NRC staff reviewed Section 18.6 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to staffing and qualification. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) contains evaluation material from the SER for the BLN Units 3 and 4 COL application.

Although the staff concluded that the evaluation performed for the standard content is directly applicable to the WLS COL application, there was a difference in the information provided by the WLS applicant from that provided by the VEGP applicant regarding the plant operating experience. This difference is evaluated by the staff below, following the standard content material.

The following portion of this technical evaluation section is reproduced from Section 18.6.4 of the VEGP SER:

> *AP1000 COL Information Item*
>
> *The following portion of this technical evaluation section is reproduced from Section 18.6.4 of the BLN SER:*
>
> - *STD COL 18.6-1, addressing staffing level and qualification of plant personnel*.

The applicant provided additional information in STD COL 18.6-1 to resolve COL Information Item 18.6-1. COL Information Item 18.6-1 states:

> *Combined License applicants referencing the AP1000 design will address the staffing levels and qualifications of plant personnel including operations, maintenance, engineering, instrumentation and control technicians, radiological protection technicians, security, and chemists. The number of operators needed to directly monitor and control the plant from the main control room, including the staffing requirements of 10 CFR 50.54(m), will be addressed.*

The commitment was also captured as COL Action Item 18.6.3-1 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:

> *The COL applicant will address the staffing level and qualifications of plant personnel including operations, maintenance* [, engineering, instrumentation] *and control technicians, radiological protection technicians, security, and chemists. Specifically, the COL applicant will (1) address the staffing considerations in NUREG-0711, and (2) identify the minimum documentation that is necessary for the staff to complete the review*

Information pertaining to the staffing level and qualifications is contained in BLN COL FSAR Chapter 13 and is summarized here. The applicant provided the estimated staffing levels for different categories of personnel that are addressed by the HFE program in accordance with NUREG-0711. The minimum staffing level for control room personnel is also stated. Information about the staffing level of security personnel is contained in the separately submitted physical security plan. Qualification requirements of Technical Support Personnel, Nuclear Plant Personnel, and Security Personnel are also included.

The baseline level of staffing is derived from experience from current operating nuclear power plants. Iterative adjustments are implemented with input from other elements of the HFE program.

The NRC staff reviewed the resolution to COL Information Item 18.6-1 related to staffing and qualifications included under Section 18.6 of the BLN COL FSAR, Revision 1.

NUREG-0711 states that satisfying criterion 4 for the staffing and qualifications should be in part based on an operating experience review. The applicant addresses this in Chapter 13, Conduct of Operations, by stating:

> *The Tennessee Valley Authority (TVA) has over 30 years of experience in the design, construction and operation of nuclear generating stations. TVA has designed, constructed, and operates six nuclear units at three sites: Browns Ferry Nuclear Plant Units 1, 2, and 3; Watts Bar Nuclear Plant Unit 1; and Sequoyah Nuclear Plant Units 1 and 2.*

NUREG-0711, Criterion 1 states that the staffing and qualifications should address applicable guidance in NUREG-0800, Section 13.1 and 10 CFR 50.54.

Section 18.6 references BLN COL FSAR Section 13, which discusses staffing levels that meet the requirements in 10 CFR 50.54.

NUREG-0711, Criterion 2 states that the staffing analysis should determine the number and background of personnel for the full range of plant conditions including operational tasks, plant maintenance, and plant surveillance and testing.

Section 18.6 of the COL states that Table 13.1-201 of the COL application contains the estimated staffing levels for those categories of personnel that are addressed in NUREG-0711, as follows:

> *1) licensed operators, 2) shift supervisors, 3) non-licensed operators, 4) shift technical advisors, 5) instrumentation and control technicians, 6) mechanical maintenance technicians, 7) electrical maintenance technicians, 8) radiation protection technicians, 9) chemistry technicians, and 10) engineering support.*

The applicant states that the minimum level of control room staffing is also stated in Table 13.1-201 and meets the requirements of 10 CFR 50.54(m).

The staff reviewed the requirements of 10 CFR 50.54, which state:

> *A senior operator licensed pursuant to Part 55 shall be present at the facility or readily available on call at all times during its operations, and shall be present at the facility during initial start-up and approach to power, recovery from an unplanned or unscheduled shut-down or significant reduction in power, and refueling.*

This section of 10 CFR contains a table that describes the minimum staffing requirements in the control room for one, two and three unit sites. For example, a one unit site with one control room is required to maintain two Senior Operators, and two Operators at all times. Table 13.1-201 describes numbers for control room operators that meet these limits and, therefore, meet the requirements for operator staffing in 10 CFR 50.54.

NUREG-0711 states that the applicant should have systematically analyzed the need for the number and qualifications of personnel and have demonstrated a thorough understanding of task requirements and regulatory requirements. NUREG-0711 also references NUREG-0800, Section 13.1 that describes the roles and responsibilities for design and construction activities and pre-operational activities. NUREG-0711 also spells out specific acceptance criteria for providing the NRC with specific information about qualification levels of the staff. In Section 13.1 of the BLN COL FSAR, the applicant describes in detail the organizational structure of the AP1000 plant. The roles and qualifications described include: Management and Technical Support

Organization; Engineering; Quality Assurance; Chemistry; Radiation Protection; Fueling and Refueling Support; Training and Development; Maintenance Support; Operations Support; and Fire Protection. Each of these sections describes the applicant's commitment for maintaining qualified staff to carry out the responsibilities of each position. For example, in Section 13.1.1.2.1, "Engineering," the applicant states:

> *The engineering department consists of system engineering, design engineering, engineering programs, and safety and engineering analysis. These groups are responsible for performing the classical design activities as well as providing engineering expertise in other areas. Each of the engineering groups has a functional manager who reports to the manager in charge of engineering and site support.*

The applicant then describes the overall roles that the engineering department is responsible for, such as:

> *Support of plant operations in the engineering areas of mechanical, structural, electrical, thermal-hydraulic, metallurgy and materials, electronic, instrument and control and fire protection. Priorities for support activities are established based on input from the plant manager with emphasis on issues affecting safe operation of the plant.*

Review Criterion 3 in NUREG-0711 states that the staffing analysis should be iterative, meaning that staffing goals should be reviewed and modified as the analyses associated with other elements are completed. The applicant addresses this criterion by stating:

> *Iterative adjustments are implemented to the staffing, as necessary, based on findings and input from periodic reviews and staffing analysis. Input to this analysis includes information derived from the other elements of the human factors engineering program, particularly operating experience reviews, functional requirements analysis and function allocation, task analysis, human reliability analysis, human-system interface design, procedure development, and training program development.*

The staff finds this information sufficient for meeting the criteria for the level and qualification of staffing contained in NUREG-0711, NUREG-0800, and 10 CFR 50.54.

<u>Evaluation of Site-Specific Information Related to Standard Content</u>

In Section 13.1.1 of the VEGP COL FSAR, the applicant provided site-specific information regarding its operating experience that the staff considered to address the staffing and qualifications basis for NUREG-0711 Criterion 4. The applicant stated:

> *Southern Nuclear Operating Company, Inc. (SNC) has over 30 years of experience in the design, construction, and operation of nuclear generating plants. SNC, with its architectural engineering predecessor Southern Company Services, Inc., has designed, constructed, and currently operates six nuclear units at three sites: Edwin I. Hatch Nuclear Plant Units 1 and 2, Joseph M. Farley Nuclear Plant Units 1 and 2, and Vogtle Electric Generating Plant Units 1 and 2.*

The staff found the VEGP operating experience to be comparable to that described by BLN. Therefore, the staff finds this information sufficient for meeting the criteria for the level and qualification of staffing described in NUREG-0711, NUREG-0800, and 10 CFR 50.54, "Conditions of licenses."

<u>Evaluation of Site-Specific Information Related to Standard Content</u>

In Section 13.1.1 of the WLS COL FSAR, the applicant provided site-specific information regarding its operating experience that the staff considered to address the staffing and qualifications basis for NUREG-0711 Criterion 4. The applicant stated:

> Duke Energy has over 40 years of experience in the design, construction, and operation of nuclear generating stations. Duke Energy operates multiple nuclear units on three sites: McGuire Units 1 and 2, Catawba Units 1 and 2, and Oconee Units 1, 2, and 3.

The staff finds that the WLS operating experience is sufficient for meeting the criteria for the level and qualification of staffing described in NUREG-0711, NUREG-0800, and 10 CFR 50.54.

18.6.5 Post Combined License Activities

There are no post-COL activities related to this section.

18.6.6 Conclusion

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to staffing and qualification, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the acceptance criteria defined in NUREG-0711, Section 6.4. The staff based its conclusion on the following:

- STD COL 18.6-1 is acceptable because it meets the acceptance criteria described in NUREG-0711, NUREG-0800, and 10 CFR 50.54.

18.7 Integration of Human Reliability Analysis with Human Factors Engineering (Related to RG 1.206, Section C.I.18.6, "Human Reliability Analysis")

HRA is an integral activity of a complete PRA. HRA seeks to evaluate the potential for, and mechanisms of, human error that may affect plant safety. Thus, it is an essential element in achieving the HFE design goal of providing a design that will minimize personnel errors, allow their detection, and provide recovery capability.

Section 18.7 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 18.7 of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

18.8 Human-System Interface Design (Related to RG 1.206, Section C.I.18.7, "Human System Interface Design")

18.8.1 Introduction

HSI design describes the design process and scope, including the translation of function and task requirements into the detailed design of alarms, displays, controls, and other aspects of the HSI through the systematic application of HFE principles and criteria. It also describes the process by which HSI design requirements are developed and HSI designs are identified and refined.

18.8.2 Summary of Application

Section 18.8 of the WLS COL FSAR, Revision 11, incorporates by reference Section 18.8 of the AP1000 DCD, Revision 19.

In addition, in WLS COL FSAR Section 18.8, the applicant provided the following:

Departures

- WLS DEP 18.8-1

The applicant proposed a Tier 2 departure (DEP) from the AP1000 DCD related to the location of the TSC and Operational Support Center (OSC).

18.8.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for WLS DEP 18.8-1 are given in Chapter 18 of NUREG-0800.

The applicable regulatory requirements for WLS DEP 18.8-1 are as follows:

- 10 CFR Part 52, Appendix D, Section VIII, "Processes for changes and departures"
- 10 CFR 52.79(c)

18.8.4 Technical Evaluation

The NRC staff reviewed Section 18.8 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the HSI design. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the WLS COL FSAR:

Departures

- WLS DEP 18.8-1

HFE design implementation in the TSC is not location-dependent. Therefore, the proposed location of the TSC between the protected areas of Units 1 and 2 is acceptable from an HFE program perspective. HFE design elements applicable to the TSC are identified and implemented in accordance with AP1000 DCD, Chapter 18, which is addressed in Section 18.2.4 of this SER.

The TSC location has the potential to affect technical data availability, communications, power supply reliability, security, and habitability. The acceptability of this location relative to these attributes is addressed in Section 13.3 of this SER.

The OSC is not in the HFE program scope. Therefore, the OSC location change is not evaluated from an HFE program perspective. The OSC location, as it relates to emergency preparedness, is evaluated in Section 13.3 of this SER.

18.8.5 Post Combined License Activities

There are no post-COL activities related to this section.

18.8.6 Conclusion

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to HSI design, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the acceptance criteria defined in NUREG-0711, Section 8.4. The staff based its conclusion on the following:

Implementation of HFE design in the TSC is not location-dependent and the HFE design elements applicable to the TSC are in accordance with AP1000 DCD, Chapter 18.

18.9 Procedure Development (Related to RG 1.206, Section C.I.18.8, "Procedure Development")

Procedure development documents, in coordination with WLS COL FSAR Section 13.5, ensure that the HFE principles and criteria, along with other design requirements, are incorporated in developing procedures that are technically accurate, comprehensive, explicit, easy to use, and validated. The procedure development program addresses the requirements specified in 10 CFR 50.34(f)(2)(ii) and describes the procedure writer's guide that establishes the process for developing technical procedures. The writer's guide ensures that procedures are consistent in organization, style, and content, and it also specifies which procedures fall within the purview of the guide.

Section 18.9 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 18.9 of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding information related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

18.10 Training Program Development (Related to RG 1.206, Section C.I.18.9, "Training Program Development")

18.10.1 Introduction

Training programs help to provide reasonable assurance that plant personnel have the knowledge, skills, and abilities to properly perform their roles and responsibilities. The training program, as discussed in this section, is coordinated with the training discussions in WLS COL FSAR Section 13.2, and describes how the training program follows a systematic approach to training, and how it addresses the requirements of 10 CFR 50.120, "Training and qualification of nuclear power plant personnel," 10 CFR 52.79(a)(33), and 10 CFR Part 55, "Operators' Licenses."

18.10.2 Summary of Application

Section 18.10 of the WLS COL FSAR, Revision 11, incorporates by reference Section 18.10 of the AP1000 DCD, Revision 19.

In addition, in WLS COL FSAR Section 18.10, the applicant provided the following:

AP1000 COL Information Item

- STD COL 18.10-1

The applicant provided additional information in STD COL 18.10-1 to resolve COL Information Item 18.10-1, addressing the execution of a training plan.

18.10.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for STD COL 18.10-1 are given in Chapter 18, Section II.A.9 of NUREG-0800.

The applicable regulatory requirements for STD COL 18.10-1 are as follows:

- 10 CFR 52.79(c)

The related acceptance criteria are as follows:

- NUREG-0711, Section 10.4
- Nuclear Energy Institute (NEI) 06-13A, “Template for an Industry Training Program Description,” Revision 1

18.10.4 Technical Evaluation

The NRC staff reviewed Section 18.10 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff’s review confirmed that the information in the application and incorporated by reference addresses the required information relating to training program development. The results of the NRC staff’s evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure the staff’s findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) contains evaluation material from the SER for the BLN Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from Section 18.10.4 of the VEGP SER:

> *The following portion of this technical evaluation section is reproduced from Section 18.10.4 of the BLN SER:*
>
> *<u>AP1000 COL Information Item</u>*
>
> - *STD COL 18.10-1, addressing execution of a training plan*
>
> *The applicant provided additional information in STD COL 18.10-1 to resolve COL Information Item 18.10-1. COL Information Item 18.10-1 refers to Section 13.2, where the COL information item in Section 13.2.1 states:*
>
> > *Combined License applicants referencing the AP1000 certified design will develop and implement training programs for plant personnel. This includes the training program for the operations personnel who participate as subjects in the human factors engineering verification and validation. These Combined License applicant training programs will address the scope of licensing examinations as well as new training requirements.*
>
> *The commitment was also captured as COL Action Item 18.10.3-1 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:*
>
> > *With regard to the training program development, the COL applicant will: (1) address the training program development in NUREG-0711; (2) address relevant concerns identified in NUREG-1793; and (3) identify the minimum documentation that the COL applicant will provide to enable the staff to complete its review.*
>
> *The NRC staff reviewed the resolution to COL Information Item 18.10-1 related to staffing and qualifications included under Section 18.10 of the BLN COL FSAR, Revision 1. Section 18.10 in the BLN COL FSAR refers to Section 13.1, "Organizational Structure of Applicant," and Section 13.2, "Training," regarding the training program development. In Section 13.2 of the BLN COL FSAR, the*

applicant provided the referenced, NRC approved, NEI 06-13A [Revision 1], "Template for an Industry Training Program Description" to address COL Information Item 18.10-1. The applicant also noted that a systematic approach to training development will be conducted in accordance with the referenced staff approved WCAP-14655, "Designer's Input for the Training of the Human Factors Engineering Verification and Validation Personnel."

The applicant provided information for the operational programs relating to non-licensed plant staff training, reactor operator training, and reactor operator re-qualification, by referencing NEI 06-13A [Revision 1], "Template for an Industry Training Program Description."

NEI 06-13A was created to provide applicants with a generic program description for use with COL application submittals. In a letter dated March 7, 2007, the staff stated that the template was an acceptable means for describing reactor operator and non-licensed plant staff training programs. The staff finds this approach to be acceptable because NEI 06-13A addresses non-licensed plant staff training, reactor operator training, and reactor operator re-qualification.

18.10.5 Post Combined License Activities

There are no post-COL activities related to this section.

18.10.6 Conclusion

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to training program development, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and is sufficient to resolve COL Action Item 18.10.3-1. The staff based its conclusion on the following:

- COL Information Item 18.10-1, relating to training, appropriately references Section 13.2 "Training." In Section 13.2, the applicant has committed to using Westinghouse Commercial Atomic Power (WCAP)-14655 to ensure a systematic approach to training development, and the applicant has referenced the staff-endorsed NEI 06-13A, Revision 1.

- Information involving nonlicensed plant staff training, reactor operator training, and reactor operator requalification are acceptably addressed because the applicant referenced NEI 06-13A, Revision 1.

- The staff's review of the WLS training program is found in Sections 13.2 and 13.4 of this SER.

18.11 Human Factors Engineering Verification and Validation (Related to RG 1.206, Section C.I.18.10, "Verification and Validation")

Human factors V&V documents the V&V activities confirming that the HSI design conforms to HFE design principles and that it enables plant personnel to successfully perform their tasks to achieve plant safety and other operational goals.

Section 18.11 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 18.11 of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

18.12 Inventory (No Corresponding Section in RG 1.206)

The specific sensors, instrumentation, controls, and alarms that are needed to operate the various plant systems constitute the inventory. The instruments, alarms, and controls for each system are documented in the piping and instrumentation diagrams. The minimum inventory required to safely shut down the reactor and maintain it shutdown is also identified.

Section 18.12 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 18.12 of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

18.13 Design Implementation (Related to RG 1.206, Section C.I.18.11, "Design Implementation")

Design implementation verifies that the as-built design conforms to the verified and validated design that resulted from the HFE design process. The scope of the design implementation includes the following considerations:

- V&V of design aspects that cannot be completed as part of the HSI V&V program
- confirmation that the as-built HSI, procedures, and training conform to the approved design
- confirmation that all HFE issues in the tracking system are appropriately addressed

Section 18.13 of the WLS COL FSAR, Revision 11, incorporates by reference, with no departures or supplements, Section 18.13 of Revision 19 of the AP1000 DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section. The results of the NRC staff's technical evaluation of the

information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

18.14 <u>Human Performance Monitoring (Related to RG 1.206, Section C.I.18.12, “Human Performance Monitoring”)</u>

18.14.1 Introduction

Human performance monitoring is used to assure that no significant safety degradation occurs because of any changes that are made in the plant and to confirm that the conclusions that have been drawn from the integrated system validation remain valid over time. Human performance monitoring is a program that begins after plant operation commences. Therefore, the applicant describes the documentation to be maintained after the program is implemented. The objective of this review is to verify that the applicant has prepared a human performance monitoring strategy for ensuring that no significant safety degradation occurs because of any changes that are made in the plant.

The program describes: (1) a human performance monitoring strategy; (2) how it trends human performance relative to changes implemented in the plant after startup; and (3) how it demonstrates that performance is consistent with that assumed in the various analyses conducted to justify the changes.

The program provides for specific cause determination, trending of performance degradation and failures, and determination of appropriate corrective actions. Detailed implementation plans and procedures for human performance monitoring remain available for NRC review.

18.14.2 Summary of Application

Section 18.14 of the WLS COL FSAR, Revision 11, incorporates by reference Section 18.14 of the AP1000 DCD, Revision 19.

In addition, in WLS COL FSAR Section 18.14, the applicant provided the following:

<u>*AP1000 COL Information Item*</u>

- STD COL 18.14-1

The applicant provided additional information in STD COL 18.14-1 to resolve COL Information Item 18.14-1, addressing human performance monitoring after the plant is placed in operation.

18.14.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for STD COL 18.14-1 are given in Chapter 18, Section II A.12 of NUREG-0800.

The applicable regulatory requirements for STD COL 18.14-1 are as follows:

- 10 CFR 52.79(c)

The related acceptance criteria are as follows:

- NUREG-0711, Section 13.4

18.14.4 Technical Evaluation

The NRC staff reviewed Section 18.14 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to human performance monitoring. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure the staff's findings on standard content that were documented in the SER for the reference COL application (VEGP Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) contains evaluation material from the SER for the BLN Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from Section 18.14.4 of the VEGP SER:

The following portion of this technical evaluation section is reproduced from Section 18.14.4 of the BLN SER:

<u>AP1000 COL Information Item</u>

- *STD COL 18.14-1 (COL Action Item 18.13-1)*

The applicant provided additional information in STD COL 18.14-1 to resolve COL Information Item 18.14-1. COL Information Item 18.14-1 states:

> *Human performance monitoring applies after the plant is placed in operation, and is a Combined License Applicant responsibility.*

The commitment was also captured as COL Action Item 18.13-1 in Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:

> *The COL applicant is responsible for human performance monitoring after the plant is placed into operation. ~~The human performance monitoring process implements the guidance and methods as described in DCD Section 18.14 Reference 1 (NUREG-0711).~~*

The applicant noted that the human performance monitoring process implements the guidance and methods as described in DCD Section 18.14. The applicant defines a broad outline of the structure of the human performance monitoring process and the assurances that can be obtained through implementation of the process. The human performance monitoring process for risk-informed changes is integrated into the corrective action program, training program, and other programs as appropriate. The cause determination process is also defined. It states that monitoring strategies for human performance trending after the implementation of the design changes are capable of demonstrating that performance is consistent with that assumed in various analyses conducted to justify the changes. Risk-informed changes are screened commensurate with their safety importance to determine if the changes require monitoring.

The NRC staff reviewed the resolution of COL Information Item 18.14-1 relating to human performance monitoring included under Section 18.14 of the BLN COL FSAR, Revision 1.

The BLN COL FSAR describes the human performance monitoring program found in NUREG-0711. It also states:

> *The human performance monitoring process for risk-informed changes is integrated into the corrective action program, training program and other programs as appropriate. Identified human performance conditions/issues are evaluated for human factors engineering applicability.*

Criterion 5 of NUREG-0711 states:

> *As part of the monitoring program, it is important that provisions for specific cause determinations, trending of performance degradation and failures, and corrective actions be included. The cause determination should identify the cause of the failure or degraded performance to the extent that corrective action can be identified that would preclude the problem or provide adequate assurance that it is anticipated prior to becoming a safety concern.*

The applicant's use of cause investigation:

- *Identifies the cause of the failure or degraded performance to the extent that corrective action can be taken consistent with the corrective action program requirements.*
- *Addresses failure significance, which includes the circumstances surrounding the failure or degraded performance, the characteristics of the failure, and whether the failure is isolated or has generic or common cause implications.*
- *Identifies and establishes corrective actions necessary to preclude the recurrence of unacceptable failures or degraded performance in the case of a significant condition adverse to quality.*

The staff has determined that the information included in Section 18.14 of the BLN COL FSAR is consistent with criteria found in NUREG-0711 and is sufficient for the staff to consider COL Information Item 18.14-1 closed.

18.14.5 Post Combined License Activities

There are no post-COL activities related to this section.

18.14.6 Conclusion

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to human performance monitoring, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the acceptance criteria defined in NUREG-0711. The staff based its conclusion on the following:

- STD COL 18.14-1, addressing human performance monitoring after the plant is placed in operation, outlines a structured approach for accomplishing this monitoring.

19.0 PROBABILISTIC RISK ASSESSMENT (RELATED TO RG 1.206, SECTION C.III.1, CHAPTER 19, C.I.19, "PROBABILISTIC RISK ASSESSMENT AND SEVERE ACCIDENT EVALUATION")

Title 10 of the *Code of Federal Regulations* (10 CFR) 52.79, "Contents of applications; technical information in final safety analysis report," Subpart C, requires applicants to submit a description of the plant-specific probabilistic risk assessment (PRA) and its results. The PRA provides an evaluation of the risk of core damage and release of radioactive material associated with both internal and external events that can occur during plant operation at power or while shutdown.

Appendix 19A to this safety evaluation (SE) section evaluates the measures identified by the applicant needed to comply with requirements to address loss of large areas (LOLAs) of the plant due to explosions or fires from a beyond design basis event (BDBE). These requirements are in 10 CFR 50.54(hh)(2) and 10 CFR 52.80(d). It should be noted that the attachment to Appendix 19A (Attachment A), as well as some documents referenced in Appendix 19A, include security-related or safeguards information. Therefore, Attachment A to Appendix 19A and the references that include security-related or safeguards information are withheld from the public in accordance with 10 CFR 2.390, "Public inspections, exemptions, requests for withholding."

19.1–19.40, 19.42–19.54, 19.56–19.57, and Appendices 19A, 19B, 19C, and 19D, Probabilistic Risk Assessment

The William States Lee III Nuclear Station (WLS) combined license (COL) Final Safety Analysis Report (FSAR), Revision 11, incorporates by reference, with no departures or supplements, Sections 19.1 through 19.40, 19.42 through 19.54, 19.56, 19.57, and Appendices 19A, 19B, 19C, and 19D of the AP1000 Design Control Document (DCD) Revision 19:

19.1, "Introduction"
19.2, "Internal Initiating Events"
19.3, "Modeling of Special Initiators"
19.4, "Event Tree Models"
19.5, "Support Systems"
19.6, "Success Criteria Analysis"
19.7, "Fault Tree Guidelines"
19.8, "Passive Core Cooling System – Passive Residual Heat Removal"
19.9, "Passive Core Cooling System – Core Makeup Tanks"
19.10, "Passive Core Cooling System – Accumulator"
19.11, "Passive Core Cooling System – Automatic Depressurization System"
19.12, "Passive Core Cooling System – In-containment Refueling Water Storage Tank"
19.13, "Passive Containment Cooling"
19.14, "Main and Startup Feedwater System"
19.15, "Chemical and Volume Control System"
19.16, "Containment Hydrogen Control System"
19.17, "Normal Residual Heat Removal System"
19.18, "Component Cooling Water System"
19.19, "Service Water System"

19.20, “Central Chilled Water System”
19.21, “AC Power System”
19.22, “Class 1E DC and UPS System”
19.23, “Non-Class 1E DC and UPS System”
19.24, “Containment Isolation”
19.25, “Compressed and Instrument Air System”
19.26, “Protection and Safety Monitoring System”
19.27, “Diverse Actuation System”
19.28, “Plant Control System”
19.29, “Common Cause Analysis”
19.30, “Human Reliability Analysis”
19.31, “Other Event Tree Node Probabilities”
19.32, “Data Analysis and Master Data Bank”
19.33, “Fault Tree and Core Damage Quantification”
19.34, “Severe Accident Phenomena Treatment”
19.35, “Containment Event Tree Analysis”
19.36, “Reactor Coolant System Depressurization”
19.37, “Containment Isolation”
19.38, “Reactor Vessel Reflooding”
19.39, “In-Vessel Retention of Molten Core Debris”
19.40, “Passive Containment Cooling”

19.42, “Conditional Containment Failure Probability Distribution”
19.43, “Release Frequency Quantification”
19.44, “MAAP4.0 Code Description and AP1000 Modeling”
19.45, “Fission Product Source Terms”
19.46, Not used
19.47, Not used
19.48, Not used
19.49, “Offsite Dose Evaluation”
19.50, “Importance and Sensitivity Analysis”
19.51, “Uncertainty Analysis”
19.52, Not used
19.53, Not used
19.54, “Low Power and Shutdown PRA Assessment”

19.56, “PRA Internal Flooding Analysis”
19.57, “Internal Fire Analysis”

Appendix 19A, “Thermal Hydraulic Analysis to Support Success Criteria”
Appendix 19B, “Ex-Vessel Severe Accident Phenomena”
Appendix 19C, “Additional Assessment of AP1000 Design Features”
Appendix 19D, “Equipment Survivability Assessment”

The staff of the Nuclear Regulatory Commission (NRC) reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there are no outstanding issues related to these sections. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793, "Final Safety Evaluation Report [FSER] Related to Certification of the AP1000 Standard Design," and its supplements.

For the remaining sections of Chapter 19, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 19.0, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors," was the principal source of guidance for the review. NUREG-0800, Section 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," was also used. The acceptability of the risk to public health and safety was determined on the basis of the results and insights derived from the applicant's plant-specific internal events PRA, site-specific assessment of external events, and severe accident evaluations. The staff's evaluation of the remaining sections of Chapter 19 is described below.

19.41 Hydrogen Mixing and Combustion Analysis

In the course of a severe accident, the oxidation of the zirconium and other metals can generate a substantial amount of combustible gas in the reactor vessel. This gas will migrate to the containment. Section 19.41 presents the design features of the AP1000 containment that control the concentration of combustible gases, including hydrogen igniters. Section 19.41 of the WLS COL FSAR, Revision 11, incorporates by reference Section 19.41, "Hydrogen Mixing and Combustion Analysis," of the AP1000 DCD, Revision 19. Section 19.41 of the DCD provides a hydrogen analysis that quantifies the threat to containment integrity with and without hydrogen igniters.

By reference, Section 19.41 of the WLS COL FSAR incorporates Section 19.41 of the AP1000 DCD, "Hydrogen Mixing and Combustion Analysis." It includes an analysis that quantifies the threat of combustible gas to containment integrity, both with and without igniters (which are not safety-related).

In addition, in the WLS COL FSAR, the applicant provided the following:

Departures

- WLS DEP 6.2-1

The applicant provided additional information in Section 19.41 of the WLS COL FSAR about WLS DEP 6.2-1 related to changes to the acceptance criteria applied to a specific Inspection, Test, Analysis, and Acceptance Criteria (ITAAC) design commitment and associated inspection, test, or analysis in Tier 1 Table 2.3.9-3, Item 3 (for control of containment hydrogen concentration for beyond-design-basis accidents) to establish consistency with the current

[1] See Section 1.2.2 for a discussion of the staff's review related to verification of the scope of information to be included in a COL application that references a design certification (DC).

detailed design of the plant. This information, as well as related WLS DEP 6.2-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.4 of this safety evaluation report (SER).

The NRC staff reviewed Section 19.41 of the WLS COL FSAR and confirmed that the combination of the DCD and the COL application is sufficient. The staff's review confirmed that with this departure, the evaluation criteria are still satisfied. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

19.55 Seismic Margin Analysis

19.55.1 Introduction

The NRC staff reviewed Section 19.55 of the WLS COL FSAR, which incorporated Section 19.55 of the DCD with no departures or supplements.

The seismic analysis and design of the AP1000 plant is based on the certified seismic design response spectra (CSDRS) shown in AP1000 DCD Tier 1, Figures 1.0-1 and 1.0-2. These spectra are based on Regulatory Guide (RG) 1.60, “Design Response Spectra for Seismic Design of Nuclear Power Plants,” Revision 1, with an increase in the 25 Hertz (Hz) region to account for increased high-frequency ground motion at some prospective sites. The CSDRS has its dominant energy content in the frequency range of 2 to 10 Hz. Additional analyses were performed for five different site profiles, including a hard-rock, high-frequency (HRHF) site with spectra corresponding to those shown in AP1000 DCD Tier 1, Figures 1.0-3 and 1.0-4.

19.55.2 Summary of Application

Section 19.55 of the WLS COL FSAR, Revision 11, incorporates by reference Section 19.55 of the AP1000 DCD, Revision 19.

Departures

- WLS DEP 2.0-1

The applicant provided additional information in Section 19.55.6.3 of the WLS COL FSAR about WLS DEP 2.0-1 related to updated seismic hazards and updated site-specific foundation response spectra for WLS that exceed the AP1000 Certified Seismic Design Response Spectra (CSDRS). The staff‘s evaluation of WLS DEP 2.0-1 and supporting site-specific analysis is included in Section 3.7.2.4 of this report.

AP1000 COL Information Item

- WLS COL 19.59.10-6

In WLS COL FSAR, the applicant added WLS COL 19.59.10-6 and a new Section 19.55.6.3, “Site-Specific Seismic Margin Analysis.” This plant-specific COL item is in response to a new

COL Information Item 19.59.10-6 proposed for the AP1000 DCD in a letter from Westinghouse dated August 23, 2010, regarding confirmation that the seismic margin analysis (SMA) documented in the AP1000 DCD section is applicable to the WLS site. Specifically, WLS COL FSAR Section 19.55 describes features of the site and provides the applicant's basis for concluding that the seismic margin for WLS is bounded by the SMA for the certified design.

19.55.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the applicable regulatory requirements for the evaluation of plant-specific information evaluated in Section 19.55 of this safety evaluation report (SER) are as follows:

- 10 CFR 52.79(a)(46), "The final safety analysis report shall include…at a level of information sufficient to enable the Commission to reach a final conclusion on all safety matters that must be resolved…before issuance of a combined license…[a] description of the plant-specific PRA and its results."

- 10 CFR 52.79(d)(1), "If the combined license application references a standard design certification, then the…final safety analysis report need not contain information or analyses submitted to the Commission in connection with the design certification, provided, however, that the final safety analysis report must either include or incorporate by reference the standard design certification final safety analysis report and must contain, in addition to the information and analyses otherwise required, information sufficient to demonstrate that the site characteristics fall within the site parameters specified in the design certification. In addition, the plant-specific PRA information must use the PRA information for the design certification and must be updated to account for site-specific design information and any design changes or departures."

Interim staff guidance (ISG) in the form of DC/COL-ISG-1, "Interim Staff Guidance on Seismic Issues of High Frequency Ground Motion in Design Certification and Combined License Applications," provides clarifying guidance on implementation of the performance-based approach for determining site-specific ground motion. It also provides guidance on implementation of evaluation methodology to determine the effects of high-frequency ground motion.

DC/COL-ISG-3, "Probabilistic Risk Assessment Information to Support Design Certification and Combined License Applications," provides clarifying guidance regarding the scope and quality of PRAs being used to support COL applications, and documentation that must be submitted in support of these applications.

For external events analysis purposes, DC/COL-ISG-3 considers the requirements of 10 CFR 52.79(d)(1) met if the COL applicant compares the site's characteristics to those assumed in the bounding analyses to ensure that the site is enveloped. If the site is enveloped, the COL applicant need not perform further PRA evaluations for these external events. However, the COL applicant should perform site-specific risk evaluations to address any site-specific hazards for which a bounding analysis was not performed or that are not enveloped by the generic analyses. This is to ensure that vulnerabilities related to siting are addressed.

DC/COL-ISG-20, “Implementation of a Probabilistic Risk Assessment-Based Seismic Margin Analysis for New Reactors,” provides guidance on plant-specific updates of the DC PRA-based seismic margin evaluation for COL applications.

19.55.4 Technical Evaluation

The NRC staff reviewed Section 19.55 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff’s review confirmed that the information in the application and incorporated by reference addresses the required information relating to SMA. The results of the NRC staff’s evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the WLS COL FSAR:

AP1000 COL Information Item

- WLS COL 19.59.10-6

The staff’s review of the AP1000 PRA-based SMA is described in Section 19.1.5.1 of NUREG-1793 and its supplements. The AP1000 SMA estimated the seismic capacity of the AP1000 plant at which there is high confidence in low probability of failure (HCLPF value). Equipment needed to safely shut down the plant is evaluated against acceleration spectra characterized by the associated free-field peak ground acceleration (PGA), expressed in terms of g (the acceleration of gravity). Specifically, in a staff requirements memorandum (SRM) dated July 21, 1993, the Commission approved the following staff recommendation specified in SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” Section II.N, “Site Specific Probabilistic Risk Assessments and Analysis of External Events,” with a modification:

> PRA insights will be used to support a margins type assessment of seismic events. A PRA based seismic margins analysis will consider sequence level HCLPFs and fragilities for all sequences leading to core damage or containment failures up to approximately one and two thirds the ground motion acceleration of the design-basis SSE [safe shutdown earthquake].

A review-level earthquake (RLE) equal to 0.5 g was established in the AP1000 DCD for the SMA and used to demonstrate a margin over the SSE of 0.3 g.

The NRC staff reviewed the proposed additions to Section 19.55 of the WLS COL FSAR. Because the ground motion response spectrum (GMRS) for the WLS site (presented in WLS COL FSAR Figures 2.0-201 and 2.0-202) is bounded by the HRHF spectrum evaluated in the

AP1000 DCD, the staff finds that using the SMA provided in the DCD is conservative and acceptable.

19.55.5 Post Combined License Activities

There are no post-COL activities identified in this section.

19.55.6 Conclusion

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to site-specific features that may affect seismic margins in the WLS COL FSAR. The information provides sufficient basis to conclude that the incorporation of the SMA documented in the AP1000 DCD is acceptable. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the requirements of 10 CFR 52.79(a)(46) and 10 CFR 52.79(d)(1). The staff based its conclusion on the following:

- WLS DEP 2.0-1, related to updated seismic hazards and updated site-specific foundation response spectra for WLS that exceed the AP1000 CSDRS, is reviewed and found acceptable by the staff in Section 3.7.2.4 of this SER.

- WLS COL 19.59.10-6, as it relates to SMA, is acceptable based on the guidance in DC/COL-ISG-3 and -20.

19.58 <u>Winds, Floods, and Other External Events</u>

19.58.1 Introduction

Section 19.58 of the WLS COL FSAR discusses risks associated with external events other than earthquakes. The staff uses this information to confirm that the total risk represented by core damage frequency (CDF) and large release frequency (LRF) remains acceptably low when accounting for external events.

With respect to external events, the applicant's response to COL Information Item 19.59.10-2 may also affect WLS COL FSAR Section 19.58. Therefore, the staff's evaluation of this COL information item is discussed in Section 19.58.4 below.

19.58.2 Summary of Application

Section 19.58 of the WLS COL FSAR, Revision 11, incorporates by reference Section 19.58 of the AP1000 DCD, Revision 19.

In addition, in WLS COL FSAR Section 19.58, the applicant provided the following:

Supplemental Information

- WLS Supplement (SUP) 19.58-1

The applicant provided supplemental information to address a portion of COL Information Item 19.59.10-2 by adding text to the end of AP1000 DCD Section 19.58.3. WLS COL FSAR Table 19.58-201, "External Event Frequencies for WLS," documents the site-specific external events evaluation that has been performed for WLS. This table provides a general explanation of the evaluation and resultant conclusions and provides a reference to applicable sections of the COL where supporting information is located. The applicant concluded that the WLS Units 1 and 2 site is bounded by the high winds, floods and other external events analysis documented in DCD Section 19.58 and no further evaluations are required at the COL application stage.

19.58.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the applicable regulatory requirements for the evaluation of WLS SUP 19.58-1 are as follows:

- 10 CFR 52.79(a)(46), "The final safety analysis report shall include…at a level of information sufficient to enable the Commission to reach a final conclusion on all safety matters that must be resolved…before issuance of a combined license…[a] description of the plant-specific PRA and its results."

- 10 CFR 52.79(d)(1), "If the combined license application references a standard design certification, then the…final safety analysis report need not contain information or analyses submitted to the Commission in connection with the design certification, *provided, however*, that the final safety analysis report must either include or incorporate by reference the standard design certification final safety analysis report and must contain, in addition to the information and analyses otherwise required, information sufficient to demonstrate that the site characteristics fall within the site parameters specified in the design certification. In addition, the plant-specific PRA information must use the PRA information for the design certification and must be updated to account for site-specific design information and any design changes or departures."

DC/COL-ISG-3 provides clarifying guidance regarding the scope and quality of PRAs being used to support COL applications, and documentation that must be submitted in support of these applications.

For external events analysis purposes, DC/COL-ISG-3 considers the requirements of 10 CFR 52.79(d)(1) met if the COL applicant compares the site's characteristics to those assumed in the generic analyses to ensure that the site is bounded. If so, the COL applicant need not perform further PRA evaluations for these external events. However, the COL applicant should perform site-specific PRA evaluations to address any site-specific hazards for which a bounding analysis was not performed or that the prior analysis does not bound to ensure that no vulnerabilities due to siting exist.

19.58.4 Technical Evaluation

The NRC staff reviewed Section 19.58 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to winds, floods, and other external events. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the WLS COL FSAR:

Supplemental Information

- WLS SUP 19.58-1

The NRC staff reviewed WLS SUP 19.58-1 related to COL Information Item 19.59.10-2.

In support of the AP1000 DC amendment, and to address part of COL Information Item 19.59.10-2, the DC applicant submitted APP-GW-GLR-101, "AP1000 Probabilistic Risk Assessment Site-Specific Considerations." This technical report expanded Section 19.58 of the AP1000 DCD with descriptions of its analyses of selected external events at a hypothetical AP1000 site. The DC applicant gathered site-specific data for those external events hazards determined applicable to each of the sites proposing to build AP1000 plants. For each event, it used the most limiting of the parameters provided by the several sites to characterize the generic AP1000 site. This produced a set of bounding analyses for the selected external events. The DC applicant evaluated these limiting external events against the criteria of NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," suitably modified.

Section 19.58 of the AP1000 DCD provides an analysis of the capability of the AP1000 design to withstand external flooding, tornadoes, hurricanes, and other site-specific external events. The second portion of COL Information Item 19.59.10-2 in the AP1000 DCD makes the following statement:

> [The] Combined License applicant will confirm that the High Winds, Floods, and Other External Events analysis documented in Section 19.58 is applicable to the COL site. Further evaluation will be required if the COL site is shown to be outside of the bounds of the High Winds, Floods, and Other External Events analysis documented in Section 19.58.

In Section 19.59 of the WLS COL FSAR the applicant provided STD COL 19.59.10-2, which included the following paragraph:

> As discussed in Subsection 19.58.3, it has been confirmed that the Winds, Floods, and Other External Events analysis documented in DCD Section 19.58 is applicable to the site. The site-specific design has been evaluated and is consistent with the AP1000 PRA assumptions. Therefore, Subsection 19.58 of the AP1000 DCD is applicable to this design.

Staff Request for Additional Information

Although site-specific information at currently proposed AP1000 sites was considered in performing the generic analyses of AP1000 DCD Section 19.58, details were not made available to the staff in the initial application. The staff issued a request for additional information (RAI) for the applicant to provide sufficient information for the staff to conclude that the WLS site was bounded by the generic analysis (RAI 19-1).

In a letter dated October 17, 2008, the applicant responded to RAI 19-1 by describing the methodology used to develop the generic external event analysis and providing a table of external event frequencies for WLS. This table documents the site-specific external events evaluation that has been performed for WLS. It provides a general explanation of the evaluation and resultant conclusions.

Potential external events and hazards were first screened for applicability to the WLS site. For events that were judged applicable, the applicant developed an initiating event frequency and provided this information to Westinghouse for use in the bounding analysis of the generic AP1000 site. Westinghouse developed a limiting event to bound the severity and frequency of all reported events; a hypothetical site for the generic analysis was characterized by these limiting events.

To address the external events in the scope of the generic analysis, the applicant provided a comparison between the AP1000 DCD limiting events and site-specific events in the response to RAI 19-1. Table 1 in the RAI 19-1 response provides an assessment of external event applicability to the WLS site (with a brief justification), as well as the applicant's estimate of event frequency for relevant external events.

The staff independently compared these inputs to the event frequencies assumed in the AP1000 DCD.

The staff reviewed the data, the applicability justifications, and the basis for event frequency estimations in this table. Events that were bounded by the external events documented in the AP1000 DCD (no more frequent and no more damaging) required no additional evaluation. Events that are predicted to occur no more than once in ten million years can be screened because they occur so infrequently (frequency less than 1×10^{-7}/year). Events that may occur more frequently but less than once in a million years (frequency less than 1×10^{-6}/year) are assessed to determine that their consequences make a negligible contribution to core damage frequency (change CDF less than 1×10^{-8}/year). Other events, if any, must be explicitly evaluated and included in the plant-specific PRA.

A number of questions remained, and the staff issued several RAIs requesting additional details and clarification to allow the staff to confirm that the key site-related assumptions in the AP1000 DCD Section 19.58 external events analyses remain valid for the WLS site (RAIs 19-3 through 19-11, 19-13 and 19-15):

- RAI 19-3 requested: (a) the basis for screening; (b) assessment of risk from events that cannot be screened (to be reported in the FSAR); and (c) the basis for the numerical values generated.
- RAI 19-4 requested clarification of the frequency of extratropical cyclones.
- RAI 19-5 requested tornadoes reported in the FSAR be reclassified using the enhanced Fujita (EF) scale to allow direct comparison between the FSAR and the referenced DCD.
- RAI 19-6 requested additional discussion of the basis for the WLS assessment of external flooding.
- RAI 19-7 requested additional discussion of the analysis for commercial aircraft.
- RAI 19-8 requested the basis for screening of external fires.
- RAI 19-9 requested additional discussion of risk related to onsite chemical storage.
- RAI 19-10 requested discussion of risk related to nearby facilities.
- RAI 19-11 requested discussion of risk related to the release of toxic materials.
- RAI 19-13 requested the basis for determining that the loss of offsite power frequencies and recovery probabilities assumed in the AP1000 PRA bound the expected site-specific values for WLS.
- RAI 19-15 requested site-specific actions that must be performed to meet the requirements for regulatory treatment of nonsafety system (RTNSS) that may be needed more than 72 hours after a high wind or external flood event.

In a letter dated August 17, 2009, the applicant responded to these RAIs with the requested clarification and discussion. In addition, the applicant revised the table that had been submitted in response to RAI 19-1 and proposed to provide it in a plant-specific supplement to the WLS COL FSAR as Table 19.58-201, “External Event Frequencies for WLS.” It documents the basis for the applicant’s assessment of risks related to winds, floods, and other external events.

A summary of the staff's review of each of the external event categories follows.

High Winds

The applicant was expected to verify that the frequency of each of the 12 high wind categories at the proposed site is bounded by the frequency assumed in Section 19.58 of the AP1000 DCD or to show that the category does not contribute to risk.

Winds that would threaten safety-related SSCs

Because WLS safety-related structures, systems, and components (SSCs) are designed to withstand winds of 300 miles per hour (mph), the COL applicant should confirm the assumption that high wind events exceeding 300 mph (the design basis for the structures of the nuclear island) are extremely rare (frequency less than 1×10^{-7} per year). Subsequent to certification of the AP1000 design, the staff issued RG 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," Revision 1. This guide states that for the continental United States, the staff considers the highest tornado wind speed with a frequency as high as 1×10^{-7} to be 230 mph. The expected frequency of 300 mph tornadoes is significantly lower. Therefore, the staff considers such events at the WLS site to be screened from further analysis on the basis of negligible frequency.

High Winds—Tornadoes

The applicant was expected to verify that the frequency of each of the six tornado classes at the proposed site is bounded by the frequency assumed in Section 19.58 of the AP1000 DCD.

In response to RAI 19-2, the applicant stated that it found this external event category to be applicable to the WLS site. The applicant provided data on observed tornadoes striking Cherokee County, in which the site is located, and seven other nearby counties. The applicant used this data to estimate the frequency of each class of tornado (on the enhanced Fujita scale) using a methodology that is described in Table 19.58-201 of the WLS COL FSAR. For each class of tornado, the frequency is less than the values assumed in Section 19.58 of the AP1000 DCD.

The staff finds that the method used to calculate tornado frequencies was conservative and, therefore, acceptable. The staff concludes that the risk from tornados at the WLS site is bounded by the risk identified in the AP1000 DCD and that no further analysis is required.

High Winds—Hurricanes and Extratropical Cyclones

The applicant was expected to verify that the frequency of each of the five hurricane categories at the proposed site is bounded by the frequency assumed in Section 19.58 of the AP1000 DCD. In addition, risk associated with extratropical cyclones must be addressed.

In response to RAI 19-1, the applicant identified this external event category as applicable to the WLS site. In response to RAI 19-4, the applicant clarified the frequency of extratropical cyclones and stated that all events with winds below hurricane force had been screened out from further evaluation because all site structures are designed to withstand them: they do not contribute to risk. In response to RAI 19-15, the applicant stated that in accordance with the RAI 19-4 response, the event frequencies for external events associated with hurricanes are bounded by the limiting initiating event frequencies given in Table 3.0-1 of APP-GW-GLR-101.

The staff evaluated the method used to calculate hurricane frequencies and finds that it was realistic and acceptable. The staff concludes that the risk from hurricanes at the WLS site is bounded by the risk identified in the AP1000 DCD. In addition, applying the screening criteria documented in the certified design, the staff finds that the consequences of extratropical cyclones present a negligible contribution to risk. For that reason, no further analysis of risk from extratropical cyclones is required.

External Floods

The AP1000 DCD calls for a site-by-site evaluation of susceptibility to floods. The applicant is expected to verify that the frequency of external flooding at the proposed site is bounded by the frequency assumed in Section 19.58 of the AP1000 DCD. The DCD states that the AP1000 is protected against floods up to the plant grade, which, at the WLS site, is 590 feet above mean sea level.

In response to RAI 19-1, the applicant identified this external event category as applicable to the WLS site. In response to RAI 19-6, the applicant provided additional justification for screening external flooding from further risk analysis for the WLS site and cited studies documented in Chapter 2 of the WLS COL FSAR. Flooding due to surge, seiches, snow melt, ice effects, flood waves from landslides, and tsunamis was evaluated and determined not to be applicable to the WLS site. The probable maximum flood considered dam failure and coincident wind wave effects, and did not result in levels that could affect safety-related structures.

The staff's evaluation of the applicant's hydrologic analyses is presented in Section 2.4 of this SER. The staff concludes that the applicant has demonstrated that consequential flooding from external sources is so unlikely that it can be screened from further risk analysis.

Transportation and Nearby Facility Accidents—Aviation Accidents

The applicant was expected to demonstrate that it is bounded by Section 19.58 of the AP1000 DCD by limiting impact frequencies to 1.2×10^{-6} per year by small aircraft and 1.0×10^{-7} per year by large, commercial aircraft. The bounding analysis for a small aircraft in the AP1000 DCD assumes that the impact would result in a loss of offsite power initiating event with subsequent loss of nonsafety-related systems. Larger (commercial) aircraft may have the capacity to challenge some safety-related SSCs, although the spatial separation of redundant safety-related system trains provides confidence that required functions will be maintained.

In response to RAI 19-1, the applicant identified this event category as applicable to the WLS site, and referenced WLS COL FSAR Section 3.5.1.6, which provides details of aircraft impact analysis. The applicant determined that the total probability of aircraft accidents that hit safety-related structures is less than 1.0×10^{-7} per year. The applicant stated that the calculated event is based on the general aviation crash rate and that this event frequency is bounded by the limiting value of 1.21×10^{-6} events/year for small aircraft in APP-GW-GLR-101. In response to RAI 19-7, the applicant provided additional justification for why the commercial aircraft impact frequency is less than 1.0×10^{-7} per year for the site. The applicant determined that there are only two airways that require further review. The estimated frequency of an accident arising from commercial aircraft would be less than 1.0×10^{-7} per year. On this basis, aircraft hazards were screened from further analysis.

The staff's evaluation of the applicant's aircraft impact assessment is presented in Sections 2.2 and 3.5 of this SER. The staff concludes that aviation accidents provide a negligible contribution to risk of core damage and that no further evaluation of risk from these accidents is required.

Transportation and Nearby Facility Accidents—Marine Accidents

In response to RAI 19-1, the applicant found that this event category was not applicable to the WLS site.

The staff finds that because there is no commercial shipping or barge traffic on waterways near the site, marine accidents need not be considered for the WLS site.

Transportation and Nearby Facility Accidents—Pipelines

In response to RAI 19-1, the applicant states that the pipelines within five miles of the WLS site do not pose a credible hazard and references WLS COL FSAR Section 2.2.3.1.1.2. In Section 2.2 of the WLS COL FSAR, the applicant concluded that the safe standoff distance for an explosive hazard is less than the distance from the site boundary to the nearby facilities.

Because the limiting event evaluated for pipeline-related explosion in the AP1000 DCD was a 30-inch pipe at a distance of 5800 feet from the plant, the applicant states that explosion hazards due to pipeline accidents can be screened from further evaluation.

The staff finds that because the risk from pipeline gas release is bounded by the AP1000 analysis, no further evaluation is required.

Transportation and Nearby Facility Accidents—Rail Accidents

In response to RAI 19-1, the applicant found that the safe standoff distance for an explosive hazard (based on trinitrotoluene equivalency) is less than the distance from the site boundary to the nearest railway. Unconfined vapor clouds of various combustible materials released at this distance were also determined not to result in any significant damage to the plant.

In response to RAI 19-11, the applicant referenced WLS COL FSAR Section 2.2.3.1.3, which provides a more detailed analysis of this scenario.

The staff's assessment of these accidents is documented in Chapter 2 of this SER. The staff concludes that they do not contribute to risk of core damage and that no further evaluation of risk from these accidents is required.

Transportation and Nearby Facility Accidents—Truck Accidents

Hazards from trucks were evaluated in the same manner as railway accidents. The safe standoff distance for an explosive hazard is less than the distance from the site boundary to the nearest highway.

Additionally, in response to RAI 19-11, the applicant referenced WLS COL FSAR Section 2.2.3, which provides a more detailed analysis of explosions and combustible material releases from trucks.

The staff's assessment of these accident analyses is documented in Chapter 2 of this SER. The staff concludes that they do not contribute to risk of core damage and no further evaluation of risk from truck accidents is required.

Transportation and Nearby Facility Accidents—Nearby Facilities

Section 19.58.2.3 of the AP1000 DCD, "Transportation and Nearby Facility Accidents," indicates that this section discusses events that "consist of accidents related to transportation near the nuclear power plant and accidents at industrial and military facilities in the vicinity." RAI 19-10 was issued requesting additional information about the toxic and explosive hazards associated with nearby facilities.

In response to RAI 19-10, the applicant referenced Section 2.2 of the WLS COL FSAR, where the military and industrial facilities within five miles of the plant are identified, the inventories of hazardous materials associated with each one are documented, and the potential consequences of release are evaluated. The applicant found the potential consequences to have a negligible effect on safety.

The staff's assessment of the applicant's analysis is documented in Chapter 2 of this SER. Because accidents at nearby facilities do not have consequences that contribute to risk, the staff finds that they can be screened from further analysis.

External Fires

The AP1000 DCD calls for the applicant to "reevaluate the qualitative screening of external fires" and perform a risk assessment if it cannot be demonstrated that the frequency of hazard is less than 1×10^{-7} per year. The NRC issued RAI 19-8 to request documentation of this reevaluation or assessment in the WLS COL FSAR.

External fires are discussed in WLS COL FSAR Chapter 2. On the basis of the distance separating the plant from potential external fires, the applicant concluded that safe operation of the plant is not jeopardized by external fires. In response to RAI 19-8, the applicant proposed to present the risks associated with external fires in WLS COL FSAR Table 19.58-201.

The staff's assessment of the applicant's analysis is documented in Chapter 2 of this SER. Because external fires do not contribute to risk, the staff finds that no further evaluation of risk from external fires is required.

Toxic Chemical Releases

The consequences of toxic chemical releases from stationary sources onsite and within five miles of the WLS site as well as mobile sources were described in WLS COL FSAR Chapters 2 and 6. The applicant reported that such releases did not pose a credible risk to control room operators.

The staff's evaluations of these analyses are documented in Chapters 2 and 6 of this SER. Because the staff concluded that the release of toxic chemicals from identified external sources did not challenge control room habitability, it does not contribute to plant risk and no further evaluation is required.

Major Depots and Storage Areas

In Chapter 2 of the WLS COL FSAR, the applicant identified three mines within five miles of the site and assessed the associated risk under this category. Explosives are not used at any of these mines and no other potential hazards are associated with these mining activities.

The staff's evaluation of this analysis is documented in Chapter 2 of this SER. Because the staff concluded that there is no hazard associated with them, they do not contribute to plant risk and no further evaluation is required.

Summary

On the basis of this additional information, the staff confirmed that for all external events that contribute to risk, the parameters used for the AP1000 DCD external events analysis bound the reported parameters of the WLS site. The staff concludes that the incorporation of AP1000 DCD Section 19.58 by reference with plant-specific supplemental information is acceptable, resolving RAIs 19-1, 19-3 through 19-11, 19-13, and 19-15. The staff confirmed that the proposed FSAR updates discussed above were appropriately incorporated into the WLS COL FSAR.

19.58.5 Post Combined License Activities

There are no post-COL activities related to this section.

19.58.6 Conclusion

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to winds, floods, and other external events, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

The staff concludes that the relevant information presented in WLS SUP 19.58-1 is consistent with the requirements of 10 CFR 52.79(a)(46) and 10 CFR 52.79(d)(1) and is, therefore, acceptable.

19.59 PRA Results and Insights

19.59.1 Introduction

This section describes the use of the PRA in the design process. It also provides an overall summary of PRA results, including those from the following analyses:

- full power, internal events PRA (both Level 1 and Level 2, providing information on CDF and LRF)
- shutdown and low power events PRA (both Level 1 and Level 2 PRA, with information on CDF and LRF)
- internal flooding assessment (both Level 1 and Level 2 PRA, with information on CDF and LRF for both full power and shutdown/low power conditions)
- internal fire assessment (both Level 1 and Level 2 PRA, with information on CDF and LRF for both full power and shutdown/low power conditions)
- SMA

In addition, this section discusses key insights from the PRA. It describes those plant features that are important to risk. It also provides information on where the PRA was used to support the certification of the AP1000 design, such as the assessment of design alternatives and scoping of the reliability assurance program.

19.59.2 Summary of Application

Section 19.59 of the WLS COL FSAR, Revision 11, incorporates by reference Section 19.59 of the AP1000 DCD, Revision 19.

In addition, in WLS COL FSAR Section 19.59.10.5, the applicant provided the following:

Departure

- WLS DEP 6.3-1

The applicant provided additional information in Section 19.59 of the WLS COL FSAR about WLS DEP 6.3-1 related to quantifying the duration that the passive residual heat removal system heat exchanger can maintain safe shutdown conditions. This information, as well as related WLS DEP 6.3-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.1 of this SER.

AP1000 COL Information Items

- STD COL 19.59.10-1

The applicant provided additional information in STD COL 19.59.10-1 to address COL Information Item 19.59.10-1. This item will evaluate any differences between the as-built plant and the certified design to confirm that seismic margins remain adequate.

- STD COL 19.59.10-2

The applicant provided additional information in STD COL 19.59.10-2 to address COL Information Item 19.59.10-2. The portion of this item dealing with evaluation of the as-built plant for conformance to the design modeled in the AP1000 PRA was originally identified in Revision 15 of the AP1000 DCD as a COL applicant's responsibility. It was subsequently identified as a COL holder's responsibility.

The portion of COL Information Item 19.59.10-2 dealing with the site-specific PRA for external events remains the responsibility of the COL applicant and is discussed in Section 19.58 of this SER.

- STD COL 19.59.10-3

The applicant provided additional information in STD COL 19.59.10-3 to address COL Information Item 19.59.10-3. This item will evaluate any differences between the as-built plant and the certified design to confirm that there are no significant adverse changes to the internal fire and internal flood analysis results.

- STD COL 19.59.10-4

The applicant provided additional information in STD COL 19.59.10-4 to address COL Information Item 19.59.10-4. The COL applicant states that severe accident management guidance (SAMG) is implemented on a site-specific basis.

- STD COL 19.59.10-5

The applicant provided additional information in STD COL 19.59.10-5 to address COL Information Item 19.59.10-5. This item, thermal lag assessment of the as-built equipment required to mitigate severe accidents, must be completed prior to initial fuel loading (for equipment that has not been tested at severe accident conditions).

- WLS COL 19.59.10-6

In WLS COL FSAR Revision 4, the applicant added WLS COL 19.59.10-6 to reflect a COL information item that is part of the DC amendment. The applicant confirmed that the SMA documented in the AP1000 DCD section is applicable to the WLS site. This COL information item is evaluated in SER Section 19.55.4.

Section 19.59 of the WLS COL FSAR adds Section 19.59.10.6 to include the following:

Supplemental Information

- STD SUP 19.59-1

The applicant provided the following supplemental information, discussing the processes for:

– maintaining the PRA to reflect the as-built, as-operated plant

– upgrading the PRA to incorporate improved methodologies and other information, as well as ensuring that it continues to meet the required NRC-endorsed consensus standards

– maintaining proper quality controls on the PRA, including computer codes used to support PRA quantification

– maintaining the PRA documentation current

– using the PRA in applications, including those that support decision making

In addition, the applicant describes where the WLS PRA is expected to provide input to other programs and processes.

License Conditions

- Part 10, License Condition 2

The proposed license condition identifies required actions that cannot be accomplished until a license is granted. It provides milestones for their completion.

- Part 10, License Condition 6

The proposed license condition requires submittal of a schedule to support NRC inspections of operational programs, including those related to implementation of SAMG.

19.59.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793 and its supplements.

In addition, the following regulations apply to Sections 19.59.10.5 and 19.59.10.6 of the WLS COL FSAR:

- 10 CFR 50.71(h)(1), "No later than the scheduled date for initial loading of fuel, each holder of a combined license under subpart C of 10 CFR Part 52 shall develop a level 1 and a level 2 probabilistic risk assessment (PRA). The PRA must cover those initiating events and modes for which NRC-endorsed consensus standards on PRA exist one year prior to the scheduled date for initial loading of fuel."

- 10 CFR 50.71(h)(2), "Each holder of a combined license shall maintain and upgrade the PRA required by paragraph (h)(1) of this section. The upgraded PRA must cover initiating events and modes of operation contained in NRC-endorsed consensus standards on PRA in effect one year prior to each required upgrade. The PRA must be upgraded every four years until the permanent cessation of operations under 10 CFR 52.110(a) of this chapter."

- 10 CFR 52.79(a)(46), "The final safety analysis report shall include…at a level of information sufficient to enable the Commission to reach a final conclusion on all safety matters that must be resolved…before issuance of a combined license:…[a] description of the plant-specific probabilistic risk assessment (PRA) and its results."

- 10 CFR 52.79(a)(38), "The final safety analysis report shall include…at a level of information sufficient to enable the Commission to reach a final conclusion on all safety matters that must be resolved…before issuance of a combined license:…a description and analysis of design features for the prevention and mitigation of severe accidents…."

- 10 CFR 52.79(d)(1), "If the combined license application references a standard design certification, then the…final safety analysis report need not contain information or analyses submitted to the Commission in connection with the design certification, *provided, however*, that the final safety analysis report must either include or incorporate by reference the standard design certification final safety analysis report and must contain, in addition to the information and analyses otherwise required, information sufficient to demonstrate that the site characteristics fall within the site parameters specified in the design certification. In addition, the plant-specific PRA information must use the PRA information for the design certification and must be updated to account for site-specific design information and any design changes or departures."

NUREG-0800 provides the following guidance:

- Section 19.0, Section III.1.C provides guidance for reviewing a COL application referencing a DC, with emphasis on documented assumptions and insights from the PRA.

- Section 19.0, Section III.3 provides guidance for reviewing COL action items.

- Section 19.1 provides information regarding the review of the technical adequacy of a design-specific, site-specific PRA.

Additional guidance is found in the following documents:

- RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 1, provides guidance on determining whether a PRA provides an adequate basis for issuing a COL.

- DC/COL-ISG-3 clarifies the staff's expectations for information to be included in the COL application.

- SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," establishes expectations for reporting scheduled implementation of operational programs.

19.59.4 Technical Evaluation

The NRC staff reviewed Section 19.59 of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this review topic.[1] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the PRA results and insights. The results of the NRC staff's evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (Vogtle Electric Generating Plant, Units 3 and 4 (VEGP)) were equally applicable to the WLS COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Section 1.2.3 of this SER provides an explanation of why the standard content material from the SER for the reference COL application (VEGP) includes evaluation material from the SER for the Bellefonte Nuclear Plant (BLN), Units 3 and 4 COL application.

The following portion of this technical evaluation section is reproduced from Section 19.59.4 of the VEGP SER:

AP1000 COL Information Items

- *STD COL 19.59.10-1*

The NRC staff reviewed STD COL 19.59.10-1, which is related to the seismic margin evaluation found in Section 19.55 of the AP1000 DCD, incorporated by reference into the BLN COL FSAR. RAI 19-1 requested justification of an apparent difference between STD COL 19.59.10-1 and the corresponding information item in the DCD. The applicant revised BLN COL FSAR Section 19.59.10.5 as follows:

> *The requirements to which the equipment is to be purchased are included in the equipment specifications. Specifically, the equipment specifications include:*
>
> 1. *Specific minimum seismic requirements [are] consistent with those used to define the Table 19.55-1 [high confidence, low probability of failure] HCLPF values. This includes the known frequency range used to define the HCLPF by comparing the required response spectrum (RRS) and test response spectrum (TRS). The range of frequency response that is required for the equipment with its structural support is defined.*
>
> 2. *Hardware enhancements that were determined in previous test programs and/or analysis programs will be implemented.*

This is consistent with the AP1000 DCD, and is therefore acceptable to the staff. As a result, the staff considers RAI 19-1 to be closed.

STD COL 19.59.10-1 states that this should be completed prior to initial fuel load, rather than at the time of the COL application. The required comparison cannot be performed until completion of fabrication, installation, and construction of SSCs, and the as-built review of the seismic margin evaluation.

The NRC staff concluded in Section 19.1.5.1 of NUREG-1793 that the methodology for calculating the HCLPF values complied with the relevant regulatory requirements, based on the certified seismic design response spectra (CSDRS). The staff concludes that it is acceptable to complete the final verification of seismic margins when the walkdowns are performed after the plant is built.

- STD COL 19.59.10-2

As noted in SER Section 19.59.2 above, this COL information item has two parts. The first part requires the COL holder to compare the as-built plant to the design used as the basis for the AP1000 PRA and DCD Table 19.59-18 (which was incorporated by reference into Chapter 19 of the applicant's FSAR). The COL holder must update the site-specific PRA to reflect differences if they potentially result in a significant increase in CDF or LRF.

Revisions to 10 CFR Part 52 and related rules were issued after the initial AP1000 DC, but prior to the submittal of the WLS COL application. Two of them, 10 CFR 52.79(d)(1) and 10 CFR 50.71(h), require that a COL application provide a description of a site-specific PRA, and that this PRA will, by fuel load, meet those industry consensus PRA standards endorsed by the NRC at least one year prior to the scheduled fuel load date. Additional guidance was provided in DC/COL-ISG-3, which states, "PRA maintenance should commence at the time of application for both DC and COL applicants. This means that the PRA should be updated to reflect plant modifications if there are changes to the design." DC/COL-ISG-3 also clarifies the staff position on what constitutes a significant change in PRA results.

The staff requested clarification in RAI 19-2 of how the WLS PRA will be updated to account for WLS site-specific information by fuel load. It also requested a definition of a "significant increase."

In response to RAI 19-2, the applicant indicated that the PRA would be updated as described in WLS COL FSAR Section 19.59.10.6. PRA updating will include evaluation of as-built plant differences, departures from the certified design, and a plant-specific review of all the PRA insights and assumptions as documented in AP1000 DCD Table 19.59-18. The applicant revised WLS COL FSAR Section 19.59.10.6 to clarify that any differences found would be evaluated and that the plant-specific PRA model would be modified as necessary to reflect both the plant-specific design and PRA-based insights.

The staff requested in RAI 19-12 that the applicant discuss the basis for concluding that the site-specific systems described in the COL application (e.g., raw water system, turbine building closed cooling water system) and modeled in the WLS PRA are consistent with the assumptions made in the development of initiating event frequencies and support system failure probabilities in the AP1000 PRA.

In response to RAI 19-12, the applicant indicated that the site-specific systems described in the COL application (e.g., raw water system, turbine building closed cooling water system, circulating water system) are designed as Class E systems and have no safety-related function and do not contain sufficient radioactive material such that a release could exceed applicable limits. The applicant also stated that the plant-specific PRA-based insight differences will be evaluated and the plant-specific PRA model modified as necessary to account for the plant-specific design and, any design changes or departures from the DC PRA.

The staff agrees that the applicant's response meets the expectations of 10 CFR 52.79(d)(1) regarding the requirement for a site-specific PRA, as well as the additional guidance described in DC/COL-ISG-3. STD COL 19.59.10-2 now states that this should be completed prior to initial fuel load, rather than at the time of the COL application. The required updates cannot be finalized until completion of fabrication, installation, and construction.

The NRC staff concluded in Section 19.1.9 of NUREG-1793 that the quality and completeness of the AP1000 PRA are adequate and satisfy the regulatory requirements. The methodology for upgrading and updating the plant-specific PRA described in the WLS COL FSAR satisfies the guidance of RG 1.200, and is, therefore, acceptable to the staff. The staff concludes that it is acceptable to update the plant-specific PRA when walkdowns are performed after the plant is built. This is consistent with the 10 CFR 50.71(h) requirement that the plant-specific PRA reflect the risk profile of the as-built, as-operated plant.

The second portion of this COL information item involves a review of site-specific external events to confirm that they are bounded by the external events addressed in the generic risk assessment for the AP1000 design. The staff's evaluation of this review is documented in Section 19.58 of this SER.

The following portion of this technical evaluation section is reproduced from Section 19.59.4 of the VEGP SER:

- *STD COL 19.59.10-3*

In response to RAI 19-20, the applicant proposed a change to its response to STD COL 19.59.10-3 to the effect that plant-specific internal fire and internal flood analysis will be evaluated and the analysis modified as necessary to account for the plant-specific design, and any design changes or departures from the certified design.

The staff reviewed STD COL 19.59.10-3, which is related to the internal fire and internal flood analyses evaluation included under Sections 19.56 and 19.57 of the AP1000 DCD, incorporated by reference in the BLN COL FSAR.

The NRC staff discussed, in Sections 19.1.5.2 and 19.1.5.3 of NUREG-1793, the methodology for assessing the risk from internal fire and floods, respectively. In Section 19.1.9, the staff concluded that the quality and completeness of the AP1000 PRA are adequate and satisfy the applicable regulatory requirements. Because the as-built configuration cannot be assessed until construction is complete, the staff finds that it is acceptable to update internal fire and flood analyses if the need to do so is identified when walkdowns are performed after the plant is built.

In a letter dated April 15, 2009 (ML091100173), the applicant proposed to revise its response to STD COL 19.59.10-1 through 19.59.10-3 and to revise License Condition 2 to conform to the revised wording of these three STD COL items. The staff identifies incorporation of these changes as Confirmatory Item 19.59-1.

<u>Resolution of Standard Content Confirmatory Item 19.59-1</u>

Confirmatory Item 19.59-1 required the applicant to revise the proposed License Condition 2 (in Part 10 of the application) to reflect the revised wording of STD COL 19.59.10-1 through 19.59.10-3. The NRC staff verified that the proposed License Condition 2 in Part 10 of the application was updated to reflect the above. As a result, Confirmatory Item 19.59-1 is resolved.

The following portion of this technical evaluation section is reproduced from Section 19.59.4 of the BLN SER:

- *STD COL 19.59.10-4*

The AP1000 DCD closed this COL information item with respect to the development of the SAMG. The COL holder will implement the AP1000 SAMG.

For STD COL 19.59.10-4 in Section 19.59.10 of the BLN COL FSAR, the applicant states, "The AP1000 Severe Accident Management Guidance (SAMG) from APP-GW-GLR-070, Reference 1 of DCD Section 19.59, is implemented on a site-specific basis." In Table 1.8-202 of the BLN COL FSAR, the applicant identifies this as a COL holder item. In response to RAI 19-3, the applicant revised its response to STD COL 19.59.10-4 in the BLN COL FSAR. The staff found this response incomplete and issued RAI 19-21.

In a letter dated April 15, 2009 (ML091100173), in response to RAI 19-21, the applicant proposed to revise License Conditions 2 and 6 to conform to the revised FSAR wording. Specifically, the applicant proposed to revise License Condition 2, Item 19.59.10-4 to reflect the fact that the SAMG development had been completed in the AP1000 DCD. In addition, the applicant proposed to revise License Condition 6 (Operational Program Readiness in Part 10 of the BLN COL application) to include a schedule for the implementation of site-specific SAMG, thereby supporting NRC inspections of operational programs in the period between issuance of a COL and authorization to load fuel in accordance with 10 CFR 52.103. This is consistent with the staff position documented in SECY-05-0197, and therefore, acceptable to the staff. The staff identifies the incorporation of these changes as Confirmatory Item 19.59-2.

<u>Resolution of Standard Content Confirmatory Item 19.59-2</u>

Confirmatory Item 19.59-2 required the applicant to revise the proposed License Condition 2 (in Part 10 of the application), item 19.59.10-4, to reflect that the SAMG development was completed in the AP1000 DCD. In addition, the confirmatory item required that the applicant revise the proposed License Condition 6 to [include] *a schedule for the implementation of site-specific SAMG. The NRC staff verified that the proposed License Conditions 2 and 6 in Part 10 of the application were updated to reflect the above. As a result, Confirmatory Item 19.59-2 is resolved.*

The following portion of this technical evaluation section is reproduced from Section 19.59.4 of the BLN SER:

- *STD COL 19.59.10-5*

The AP1000 DCD, Revision 17, changed the wording of COL Information Item 19.59.10-5 to clarify which equipment requires thermal lag assessment. STD COL 19.59.10-5 in Chapter 19 of the BLN COL FSAR, as well as the COL holder item listed in License Condition 2 (Part 10 of the BLN COL application) have been revised to conform with the AP1000 DCD.

The NRC staff concluded, in Section 19.2.3.3.7.3 of NUREG-1793, that the equipment and instrumentation identified as required to mitigate severe accidents meets the guidance of SECY-93-087 and 10 CFR 50.34(f). In addition, the staff required that the COL applicant referencing the AP1000 certified design perform a thermal response assessment of as-built equipment used to mitigate severe accidents. Since the as-built equipment and configuration are not available until after the COL is issued, the staff concludes that it is acceptable to complete thermal lag assessments prior to fuel load.

<u>COL Action Items from Chapter 19 of NUREG-1793</u>

The staff compared COL information items in Chapter 19 of the AP1000 DCD with the COL action items from NUREG-1793. The staff identified differences between them, which resulted in two RAIs:

<u>RAI 19-6</u>

Two items from NUREG-1793 relate to the training of operators to respond to certain conditions during shutdown. The first calls for the COL applicant to train operators to quickly close containment hatches and penetrations in the event of an accident during Modes 5 or 6. This must be completed before boiling begins in the reactor coolant system (RCS).

The BLN COL FSAR cited APP-GW-GLR-040, "Plant Operations, Surveillance, and Maintenance Procedures." This is the template document for AP1000 procedure generation. The applicant also noted that BLN COL FSAR Section 13.2 incorporates by reference NEI 06-13, "Template for an Industry Training Program Description." Sections 1.1.1.1, 1.1.1.2, 1.1.2, and 1.2.1 of this document focus on training for operations during shutdown, including abnormal and emergency operations. Technical Specification 3.6.8 provides direction for maintaining containment closure capability prior to steaming during Modes 5 and 6, and it is expected that operators will be well versed in technical specification requirements.

The staff finds that this is an acceptable way to ensure that operators will be prepared to close containment hatches in the event of an accident during Mode 5 or 6.

The second calls for operator training in the use of the wide range pressurizer level indication to cross-check the safety-related narrow range hot-leg level instruments. This is to avoid inadvertent over-draining of the RCS, particularly during reduced inventory operation. The staff reviewed Table 19.59-18, "AP1000 PRA-Based Risk Insights." Item 62 of the table explicitly states, "It is important to maximize the availability of the non-safety-related wide range pressurizer level indication during RCS draining operations during cold shutdown. Procedures and training must be developed to encompass this item." BLN COL 19.59.10-2 includes verification of every item in this table by the COL holder, prior to fuel load. This is accomplished by comparing each item to the as-built (and as operated) plant.

The staff finds this to be an acceptable way to confirm that operators are adequately trained on the use of wide range pressurizer level indication as a cross-check on the safety-related narrow range hot-leg level instruments. Therefore, RAI 19-6 is closed.

RAI 19-7

The staff sought more specific information about compensatory measures used to maintain adequate internal fire and flooding detection and suppression capability during maintenance activities that may impair these features.

The applicant responded by indicating that compensatory measures for fire protection are addressed in BLN COL FSAR Section 9.5.1.8.1.2, which describes use of a permit system that controls and documents inoperability of fire protection systems and equipment, and establishes requirements to initiate proper notifications and compensatory actions, such as fire watches, when the inoperability of any fire protection system or component, such as detectors or suppression devices, is identified. The staff reviewed the cited section of the BLN COL FSAR, and found that it adequately addresses situations when maintenance activities potentially impair fire detection and suppression equipment.

The applicant also responded that flooding detection and suppression equipment, such as sump level indicators, are identified as specific design features in BLN COL FSAR Sections 3.4 and 9.3.5. The most important ones, containment sump level indicators, are controlled by technical specification limiting conditions for operations (LCOs) with required actions and completion times. In addition, flood control in other places is managed by a floor drain system, which provides level detection, as well as manual or automatic pump down of the sumps, which collect water entering the floor drains. Administrative procedures described in BLN COL FSAR Section 13.5.1 control maintenance *activities and provide for equipment control and, if needed, compensatory action when maintenance activities impair flooding control equipment.*

The staff reviewed the references provided by the applicant and finds the applicant's responses provide adequate compensatory action; therefore, RAI 19-7 is closed.

<u>Supplemental Information</u>

- *STD SUP 19.59-1*

The applicant provided supplemental information in BLN COL FSAR Section 19.59.10.6, "PRA Configuration Controls." The applicant discusses how the BLN plant-specific PRA is developed and maintained to reflect the as-built and as-operated plant, as well as how it will be used to support other programs.

The applicant committed to upgrade the Level 1 and Level 2 PRA prior to fuel load to cover those initiating events and modes of operation set forth in NRC-endorsed consensus standards on PRA that are in effect one year prior to the scheduled date of the initial fuel load. In addition, upgrades are completed at least once every four years. This is consistent with 10 CFR 50.71(h) and, therefore, acceptable to the staff.

In addition, the applicant committed to monitor various information sources for changes or new information that could affect the model assumptions or quantification. Plant-specific design, procedure, and operational changes are reviewed for risk impact. A screening process determines whether a PRA update should be performed more frequently, and includes consideration of whether the changes affect the PRA insights. If the changes warrant a PRA update, the update is made as soon as practicable consistent with the importance of the change and the applications being used. Otherwise, changes are tracked and incorporated in the next regularly scheduled update. This is consistent with RG 1.200, Revision 1, and therefore acceptable to the staff.

PRA quality assurance (QA) provisions ensure that personnel involved in PRA are qualified, work is reviewed independently, documentation is adequately controlled, and upgrades to the PRA are peer-reviewed. When assumptions, analyses, or information used previously are changed or determined to be in error, potential impacts to the PRA model are tracked. If errors are found in the PRA model, they are tracked and appropriate corrective action governed by procedures is taken. This is consistent with RG 1.200 and, therefore, acceptable to the staff.

The PRA provides input to various programs and processes, such as implementation of the maintenance rule, reactor oversight process, the reliability assurance program, the program for regulatory treatment of non-safety systems, and the motor-operated valve (MOV) program. The staff agrees that a *plant-specific, site-specific PRA, based on the generic PRA for the AP1000 and maintained as described in the BLN COL FSAR, is an appropriate model to provide input to each of these risk-informed activities.*

19.59.5 Post Combined License Activities

The license condition language in this section has been clarified from previously considered language. In a letter dated March 22, 2016 (ADAMS Accession No. ML16084A099), the applicant did not identify any concerns with the clarified license condition language. The changes do not affect the staff's above analysis of the conditions, and therefore, for the reasons discussed in the technical evaluation section above, the staff finds the following license conditions acceptable:

- License Condition (19-1) – The licensee shall review differences between the as-built plant and the design used as the basis for the AP1000 SMA prior to initial fuel load. The licensee shall perform a verification walkdown to identify differences between the as-built plant and the design. The licensee shall evaluate any differences and shall modify the seismic margin analysis as necessary to account for the plant-specific design and any design changes or departures from the certified design. The licensee shall compare the as-built SSC HCLPFs to those assumed in the AP1000 seismic margin evaluation prior to initial fuel load. The licensee shall evaluate deviations from the HCLPF values or assumptions in the seismic margin evaluation due to the as-built configuration and final analysis to determine if vulnerabilities have been introduced.

- License Condition (19-2) – Before initial fuel load, the licensee shall review differences between the as-built plant and the design used as the basis for the AP1000 probabilistic risk assessment (PRA) and the AP1000 DCD, Rev. 19, Table 19.59-18. The licensee shall evaluate the plant-specific PRA-based insight differences and shall modify the plant-specific PRA model as necessary to account for the plant-specific design and any design changes or departures from the design certified in Rev. 19 of the AP1000 DCD.

- License Condition (19-3) – Before initial fuel load, the licensee shall review differences between the as-built plant and the design used as the basis for the AP1000 internal fire and internal flood analysis. The licensee shall evaluate the plant-specific internal fire and internal flood analyses and shall modify the analyses as necessary to account for the plant-specific design and any design changes or departures from the design certified in Rev. 19 of the AP1000 DCD.

- License Condition (19-4) – No later than 12 months after issuance of the COL, the licensee shall submit to the Director of NRO, or the Director's designee, a schedule for implementation of the site-specific severe accident management guidelines. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until this license condition has been fully implemented. The schedule shall identify the implementation of the site-specific severe accident management guidelines (before startup testing).

- License Condition (19-5) – Prior to initial fuel load, the licensee shall perform a thermal lag assessment of the as-built equipment listed in Tables 6b and 6c in Attachment A of APP-GW-GLR-069, “Equipment Survivability Assessment,” to provide additional assurance that this equipment can perform its severe accident functions during environmental conditions resulting from hydrogen burns associated with severe accidents. This assessment is required only for equipment used for severe accident mitigation that has not been tested at severe accident conditions. The licensee shall assess the ability of the as-built equipment to perform during accident hydrogen burns using the environment enveloping method or the test based thermal analysis method described in Electric Power Research Institute (EPRI) NP-4354, “Large Scale Hydrogen Burn Equipment Experiments.”

19.59.6 Conclusion

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff’s review confirmed that the applicant addressed the required information relating to PRA results and insights, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements. In addition, WLS DEP 6.3-1, related to quantifying the duration that the passive residual heat removal system heat exchanger can maintain safe shutdown conditions, is reviewed and found acceptable by the staff in Section 21.1 of this SER.

The staff concludes that the relevant information presented in Section 19.59 of the WLS COL FSAR is consistent with the requirements of 10 CFR 52.79(a)(46) and 10 CFR 52.79(d)(1) and is, therefore, acceptable.

Appendix 19E Shutdown Evaluation

Appendix 19E presents the design features of the active systems and passive safety-related systems that address the issues of shutdown risk and shutdown safety. It also evaluates the design features with respect to their ability to reduce or mitigate the consequences of events that can occur during shutdown, including discussions of the following:

- Systems designed to operate during shutdown
- Shutdown operations (including maintenance insights, risk management, and Emergency Response Guidelines (ERGs))
- Safety analyses and evaluations for shutdown operations
- Chapter 16, “Technical Specifications”
- Shutdown risk evaluations (including shutdown PRA results and fire/flood risk)
- Consistency with the guidance in NUREG-1449

Appendix 19E of the WLS COL FSAR, Revision 11, incorporates by reference Appendix 19E, “Shutdown Evaluation,” of the AP1000 DCD, Revision 19. Appendix 19E of the DCD provides a shutdown evaluation and includes Sections 19E.2.3, “Passive Core Cooling System,” 19E.4.3, “Decrease in Heat Removal by the Secondary System,” and 19E.4.10.2, “Shutdown Temperature Evaluation.”

In addition, in the WLS COL FSAR, the applicant provided the following:

Departures

- WLS DEP 3.2-1 and WLS DEP 6.3-1

The applicant provided additional information in Appendix 19E of the WLS COL FSAR about WLS DEP 3.2-1 and WLS DEP 6.3-1 related to design modifications to the condensate return portion of the Passive Core Cooling System and quantifying the duration that the passive residual heat removal heat exchanger can maintain safe shutdown conditions, respectively. This information, as well as related WLS DEP 3.2-1 and WLS DEP 6.3-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.1 of this SER.

- WLS DEP 7.3-1

The applicant provided additional information in Appendix 19E of the WLS COL FSAR about WLS DEP 7.3-1 related to required design changes for the PMS source range neutron flux doubling logic to comply with the requirements of IEEE Std. 603-1991, Clause 6.6. This information, as well as related WLS DEP 7.3-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.5 of this SER.

The NRC staff reviewed Appendix 19E of the WLS COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL application represents the complete scope of information relating to this section. The NRC staff’s review confirmed that the applicant addressed the required information to satisfy the evaluation criteria. There is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. The results of the NRC staff’s technical evaluation of the information incorporated by reference in the WLS COL application are documented in NUREG-1793 and its supplements.

Appendix 19F Malevolent Aircraft Impact

Appendix 19 F of the WLS FSAR addresses the requirements related to 10 CFR 50.150, “Malevolent Aircraft Impact.” In FSAR Appendix 19F, the applicant incorporated by reference Appendix 19F of the AP1000 DCD, Revision 19.

In 2016, the staff concluded an inspection of the Westinghouse Electric Company, LLC, to examine recent design changes and the resolutions of the 2010 notice of violations with respect to 10 CFR 50.150 (ADAMS Accession No. ML102980583). The April 19, 2016 inspection report identified two issues with the existing AP1000 aircraft impact assessment (AIA) and the AP1000 DCD (ADAMS Accession No. ML16099A049).

The first issue involved the crediting of the Auxiliary Building in the AIA as a key design feature for protecting the integrity of the spent fuel pool and for protecting from physical damage the equipment needed to maintain core cooling. However, only the spent fuel pool integrity credit was translated into Appendix 19F of the AP1000 DCD incorporated by reference by the WLS COL applicant. Since the AP1000 DCD was missing the information about the Auxiliary Building credit to protect core cooling equipment from physical damage, the WLS COL application also omits this citation of the Auxiliary Building as a key design feature relied upon to ensure core cooling capability.

The second issue involved the fire damage spread in certain plant areas not following the methodology in NEI 07-13, “Methodology for Performing Aircraft Impact Assessments for New Plant Designs,” Revision 7. Fire protection features with specific ratings cited in the NEI 07-13 guidance had not been incorporated into Appendix 19F or Appendix 9A of the AP1000 DCD, and thus not incorporated into the WLS COL application.

At the conclusion of the inspection, the staff found the revised AIA acceptable, including the addition of specific pressure-rated fire doors.

To address and capture the missing information identified in the April 19, 2016, inspection report, the staff proposes the following license condition. This license condition would allow the staff to conclude that the WLS Units 1 and 2 would be constructed and operate in compliance with the requirements of 10 CFR 50.150:

- License Condition (19-6) – At the first annual update of the WLS FSAR required by 10 CFR 50.71(e) DEC shall include the following changes based on inspection findings from NRC Inspection Report No. 99900404/2015-203:

  a) Revise Appendix 19F.4.1, “Malevolent Aircraft,” to include the Auxiliary Building as a key design feature that also protects from physical damage the core cooling credited to meet 10 CFR 50.150(b)(2).

  b) Revise DCD drawings to show the 5 psid and 3 hour fire rated doors that have been added to the inner portion (annulus side) of the shield building in accordance with final markups used to satisfy NRC Inspection Report No. 99900404/2015-203 and 50.150 (a)(1). The DCD figures listed below are to be revised:

     1. Figure 1.2-7 - Nuclear Island General Arrangement Plan at Elevation 107'-2" & 111'-0"
     2. Figure 1.2-10 - Nuclear Island General Arrangement Plan at El. 135'-3"
     3. Figure 9A-1 (Sheet 5 of 16) - Nuclear Island Fire Areas Plan at Elevation 100'-0" & 107'-2"
     4. Figure 9A-1 (Sheet 7 of 16) - Nuclear Island Fire Area Plan at Elevation 135'-3"
     5. Figure 12.3-1 (Sheet 6 of 16) - Radiation Zones, Normal Operations/Shutdown Nuclear Island, Elevation 100'-0" & 107'-2"

6. Figure 12.3-1 (Sheet 8 of 16) - Radiation Zones, Normal Operations/Shutdown Nuclear Island, Elevation 135'-3"
7. Figure 12.3-2 (Sheet 6 of 15) - Radiation Zones, Post-Accident Nuclear Island, Elevation 100'-0" & 107'-2"
8. Figure 12.3-2 (Sheet 8 of 15) - Radiation Zones, Post-Accident Nuclear Island, Elevation 135'-3"
9. Figure 12.3-3 (Sheet 6 of 16) - Radiological Access Controls, Normal Operations/Shutdown Nuclear Island, Elevation 100'-0" & 107'-2"
10. Figure 12.3-3 (Sheet 8 of 16) - Radiological Access Controls, Normal Operations/Shutdown Nuclear Island, Elevation 135'-3"

The license condition part (a) requires the applicant to include, as an update to the applicant's UFSAR Appendix 19F, the Auxiliary Building as a structure to protect core cooling equipment from structural physical damage in addition to its role of protecting the spent fuel pool integrity as analyzed in the aircraft impact assessment. Therefore, the staff finds that with the incorporation of this change, the applicant meets the requirements of 10 CFR 50.150(b)(2) which require applicants to describe in their FSAR how each key design feature meets the acceptance criteria credited in 10 CFR 50.150(a) because UFSAR Appendix 19F will reflect that the Auxiliary Building is credited to protect from physical damage the core cooling equipment in the AIA.

The license condition part (b) requires the applicant to incorporate, as an update to the applicant's UFSAR, those design changes contained in the identified figures to be revised, and within Westinghouse's Design Change Proposal APP-GW-GEE-2450, "Relocation of AIA Blast Doors and Addition of Shielding Doors to Annulus Personnel Access Portals," Revision 0. Specifically, those changes which address, in part, the specific 3-hour fire rated door additions and their proper pressure ratings. The staff reviewed these proposed changes during the Inspection 99900404/2015-203 and found them acceptable in accordance with the guidance in NEI 07-13, Revision 7. Therefore, the staff finds that with the incorporation of these changes, the applicant meets the requirements of 10 CFR 50.150(b)(1), which require the applicant to identify and describe in the FSAR those key design features required to satisfy 10 CFR 50.150(a)(1), because the revised figures will identify and describe the added key design features (i.e., fire doors).

Appendix 19.A

LOSS OF LARGE AREAS OF THE PLANT DUE TO EXPLOSIONS OR FIRES

19.A.1 Introduction

Duke Energy Carolinas, LLC (Duke Energy), in Part 9 of the William States Lee III Nuclear Station, Units 1 and 2 (WLS) COL application submitted the Loss of Large Areas of the Plant Due to Explosions or Fire Mitigative Strategies Description and Plans (MSD).

In the submittal, the applicant describes how the requirements to address loss of large areas (LOLAs) of the plant due to explosions or fires from a beyond-design basis event (BDBE) are met. These requirements are in Title 10 of the *Code of Federal Regulations* (10 CFR) 52.80(d) and 10 CFR 50.54(hh)(2). It should be noted that the attachment to this safety evaluation (SE) section (Attachment A), as well as some documents referenced in this SE section, include security-related or safeguards information, and are not publicly available.

The provisions of 10 CFR 52.80(d) require an applicant for a combined operating license (COL) to submit a description and plans for implementation of the guidance and strategies intended to maintain or restore core cooling, containment, and spent fuel pool (SFP) cooling capabilities under the circumstances associated with the LOLAs of the plant due to explosions or fire as required by 10 CFR 50.54(hh)(2).

The provisions of 10 CFR 50.54(hh)(2) require licensees to develop and implement guidance and strategies for addressing the LOLAs of the plant due to explosions or fires from a BDBE. Specifically, guidance and strategies are intended to maintain or restore core cooling, containment, and SFP cooling capabilities including:

- fire fighting
- operations to mitigate fuel damage
- actions to minimize radiological release

19.A.2 Summary of Application

Duke Energy's submission of its "Loss of Large Areas of the Plant Due to Explosions or Fire – Mitigative Strategies Description and Plans" incorporated the full, non-redacted version of the MSD, including changes identified in response to NRC requests for additional information (RAIs), in Part 9 of the WLS COL application. The redacted version of this MSD is incorporated in Part 11 of the WLS COL application. The applicant stated that the LOLA mitigative strategies, including implementation of operational and programmatic aspects of responding to loss of large area events, would be implemented prior to initial fuel load.

License Conditions

- Part 10, License Condition 6

The applicant proposed a license condition in Part 10 of the WLS COL application to provide a schedule to support the NRC's inspection of operation programs including the programmatic elements of responding to an event associated with a loss of large areas of the plant due to explosions or fire, prior to initial fuel load.

19.A.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed in NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," and its supplements.

The applicable regulatory requirements for loss of large areas of the plant due to explosions or fires are as follows:

- 10 CFR 50.54(hh)(2)
- 10 CFR 52.80(d)

The applicable regulatory guidance include Interim Staff Guidance (ISG) DC/COL-ISG-016, "Compliance with 10 CFR 50.54(hh)(2) and 10 CFR 52.80(d) Loss of Large Areas of the Plant due to Explosions or Fires from a Beyond-Design Basis Event" (not publically available), which provides an acceptable means of meeting the requirements of 10 CFR 50.54(hh)(2) and 10 CFR 52.80(d). The ISG-016 references the February 25, 2005, guidance letter (not publically available) to operating reactor licensees for Phase 1 and the Nuclear Energy Institute (NEI) document NEI 06-12, "B.5.b Phase 2 & 3 Submittal Guideline," Revision 3, for Phases 2 and 3 (not publically available). The DC/COL-ISG-016 takes exception to a few areas of NEI 06-12, and provides additional clarification and enhancement of NEI 06-12 and the staff's guidance letter issued February 25, 2005, based on NRC inspections of operating reactor implementation. The DC/COL-ISG-016 has two attachments: Attachment 1 is titled, "Supplementary Guidance for Implementing Mitigation Strategies," and Attachment 2 is titled, "Experience Gained from Implementation of Temporary Instruction 2515/171 at Currently Licensed Power Reactor Sites and Related Staff Positions."

19.A.4 Technical Evaluation

The staff reviewed the applicant's submittal consistent with the requirements of 10 CFR 52.80(d) and 10 CFR 50.54(hh)(2). The staff also used the guidance in DC/COL-ISG-016 to perform its review. The DC/COL-ISG-016 references the February 25, 2005, guidance letter for Phase 1, and NEI 06-12 for Phases 2 and 3. A discussion of the staff's technical evaluation of the WLS Units 1 and 2 submittal is found in Attachment A to Appendix 19.A.

The WLS COL applicant provided the LOLA event evaluation via a three-phased approach similar to existing plants and consistent with the NEI 06-12 guidance, Phases 1, 2, and 3. The applicant's MSD was written at the programmatic level for licensing approval, and the implementation details and documentation will be made available for inspection by the NRC prior to initial fuel load. In response to NRC staff RAIs, the applicant submitted additional information to clarify the MSD. The applicant's responses to these RAIs are evaluated by the NRC staff in Attachment A to this SE section. Revisions to Parts 9 and 11 of the WLS COL application, including modifications made to the MSD are discussed in detail in Attachment A to Appendix 19A of this SER.

In its submittal of the MSD, the applicant provided a Mitigative Strategies Table (MST), which follows the template guidance in Appendix D to NEI 06-12. The MST addresses various areas and issues pertinent to loss of large areas and describes commitments, including completion dates, for areas that are best resolved closer to the completion of building WLS Units 1 and 2. All commitments made in the submittal will be implemented prior to the initial fuel load of the units.

The MST addresses the three phases considered in NEI 06-12. The phases as described in the guidance documents can be mapped to the regulatory requirements and are as follows:

- Phase 1 – Fire Fighting Response Strategy
- Phase 2 – Spent Fuel Pool Cooling
- Phase 3 – Reactor Core Cooling and Fission Product Release Mitigation

Phases 1, 2, and 3 of NEI 06-12 are similar to the three areas included as part of the requirements in 10 CFR 50.54(hh)(2): fire fighting, operations to mitigate fuel damage, and actions to minimize radiological release. However, the three phases are categorized differently. In 10 CFR 50.54(hh)(2), the category of operations to mitigate fuel damage includes both the reactor core and the spent fuel pool, and the category of actions to minimize radiological release is separate. In NEI 06-12, spent fuel pool and reactor core cooling are found in separate phases, and reactor core cooling and fission product release mitigation are combined. Despite the change in the categorization of the phases in NEI 06-12 and the areas of the regulatory requirements, the staff finds all of the necessary information is included in the submittal.

The guidance for Phases 1, 2, and 3 suggests development of certain strategies or processes to mitigate the consequences of a LOLA event. The applicant addressed all of these suggested strategies or processes. In evaluating each plant specific mitigating strategy against its functional objective[1], the staff weighed whether the strategy reasonably can be expected to successfully provide spent fuel pool cooling, or maintain or restore the key safety functions necessary to protect the reactor core and containment. The staff's review considered the expected effectiveness of strategies and the ease and timeliness of strategy implementation.

While some strategies needed to meet 10 CFR 50.54(hh)(2) can be developed and implemented in the near future, some strategies and planning efforts cannot be effectively determined or implemented until the plant is further along in construction. To identify such commitments for future action, the applicant documented areas that would be more appropriately completed prior to the initial fuel load. The staff reviewed the commitments made by the applicant in its submittal and is satisfied that the timing of all procedural or strategy development was appropriately scheduled prior to the initial fuel load.

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[1] As used here, the functional objective is the basic description of the capabilities of the conceptual strategy(s) as proposed for Phase 2 and 3 by NEI and accepted by NRC.

The MSD has been reviewed by the NRC staff for content using DC/COL-ISG-016, and found to include all strategies considered essential for such a program, and is acceptable. The staff finds that the regulatory requirements of 10 CFR 52.80(d) and 10 CFR 50.54(hh)(2) are met.

License Conditions

- Part 10, License Condition 6

In RAI 19-95, the staff asked Vogtle Electric Generating Plant (VEGP) to provide a draft license condition to be added to Part 10 of the VEGP Units 3 and 4 COL application related to implementation of mitigative strategies and to submitting schedules to support planning for and conduct of NRC inspections. In its response dated May 24, 2010, VEGP provided a license condition in Part 10 of the VEGP COL application to provide a schedule to support the NRC's inspection of operational programs, including the programmatic elements of responding to an event associated with a loss of large areas of the plant due to explosions or fire, prior to initial fuel load. Although this program is not identified as an operational program in SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," the proposed license condition is consistent with the policy established in SECY-05-0197 for operational programs in general, and is acceptable. WLS endorsed this response as standard material in a letter dated November 4, 2010. Thus, this RAI is closed.

- Managing MSD Commitments

In RAI 19-96, the staff asked VEGP to describe its plans for managing changes to the commitments included in the MSD. In its response dated May 24, 2010, VEGP included a revision to the MSD that states that commitments in the MSD will be captured in the licensee's commitment management program and managed in accordance with the guidance in NEI 99-04, Revision 0, "Guidelines for Managing NRC Commitment Changes," July 1999. This is similar to the approach followed by the operating fleet licensees commitments made under Section B.5.b of the 2002 Interim Compensatory Measures. In its November 4, 2010 letter, WLS endorsed this response as standard material.

The NRC staff reviewed specific commitments in the MSD and used these commitments as the basis for the staff's safety conclusion. The staff finds that a commitment management program conforming to the guidance in NEI 99-04, Revision 0, is appropriate for managing the commitments contained in the MSD. However, the staff is proposing that a license condition be included requiring the licensee to use a commitment management program which conforms to the guidance in NEI 99-04, Revision 0. Subsequently, the staff decided that the most appropriate way to handle the commitments and maintenance of the MSD was to insure that the licensee maintains the guidance and strategies developed in accordance with 10 CFR 50.54(hh)(2). This language was included in the staff proposed License Condition 19.A-1. Thus, this RAI is closed.

19.A.5 Post Combined License Activities

The license condition language in this section has been clarified from previously considered language. In a letter dated March 22, 2016 (ADAMS Accession No. ML16084A099), the applicant did not identify any concerns with the clarified license condition language. The changes do not affect the staff's above analysis of the conditions, and therefore, for the reasons discussed in the technical evaluation section above, the staff finds the following license condition acceptable:

- License Condition (19.A-1) – No later than 12 months after issuance of the COL, the licensee shall submit to the Director of NRO, or the Director's designee, a schedule for implementation of the operational and programmatic elements of the mitigative strategies for responding to circumstances associated with loss of large areas of the plant due to explosions or fire. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until each license condition has been fully implemented. The schedule shall identify the completion of or implementation of the operational and programmatic elements of the mitigative strategies for responding to circumstances associated with loss of large areas of the plant due to explosions or fire developed in accordance with 10 CFR 50.54(hh)(2) (before initial fuel load).

19.A.6 Conclusion

The NRC staff reviewed the information provided by the applicant under 10 CFR 52.80(d) and concludes that the applicant has adequately followed the guidance of DC/COL-ISG-016; NEI 06-12; and the February 25, 2005, guidance letter. The staff finds that the applicant provided sufficient information at the COL application stage, including commitments made in the WLS COL application, to meet the requirements of 10 CFR 52.80(d) and to provide reasonable assurance that the requirements in 10 CFR 50.54(hh)(2) will be met prior to the initial fuel load of WLS Units 1 and 2, respectively.

Attachment A to Appendix 19.A, "Loss Of Large Areas of the Plant Due to Explosions or Fires"

Evaluation of the William States Lee III Nuclear Station Units 1 and 2 Submittal to Meet the Requirements of 10 CFR 50.54(hh)(2)

Introduction

This attachment documents the evaluation by the Nuclear Regulatory Commission (NRC) of submittals made by Duke Energy Carolinas, LLC (Duke Energy) (the applicant) regarding how the William States Lee III Nuclear Station (WLS) Units 1 and 2 design and mitigation strategies meet the requirements of 10 CFR 50.54(hh)(2) and 10 CFR 52.80(d). The provisions of 10 CFR 50.54(hh)(2) require nuclear power plant utilities to develop and implement guidance and strategies for addressing the loss of large areas (LOLAs) of the plant due to explosions or fires from a beyond-design-basis event. Specifically, licensees must develop and implement guidance and strategies intended to maintain or restore core cooling, containment, and spent fuel pool (SFP) cooling capabilities under the circumstances associated with LOLAs of the plant due to explosions or fire. The staff's review is based on DC/COL-ISG-016 and its referenced documents.

Section 1.2.3 of this safety evaluation report (SER) provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the design certification (DC) and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (Vogtle Electric Generating Plant (VEGP), Units 3 and 4) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP Mitigative Strategies Description and Plans (MSD) to the WLS MSD. In performing this comparison, the staff considered changes made to the WLS MSD (and other parts of the COL application, as applicable) resulting from requests for additional information (RAIs).

- The staff compared VEGP COL FSAR Section 9.5.1 and Appendix 9A to the WLS COL FSAR Section 9.5.1 and Appendix 9A.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were either endorsed or responded to in the same manner in those instances where NRC staff issued RAIs to WLS that mirrored the VEGP RAIs. Attachment A, Tables 19-A-1 and 19-A-2 contain a cross-reference of the VEGP to WLS RAIs.

- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. Any confirmatory items in the standard content material retain the numbers assigned in the VEGP SER.

Although the staff concluded that the evaluation performed for the standard content is directly applicable to the WLS COL application, there were differences in the information provided by the WLS applicant from that provided by the VEGP applicant. These differences are identified and evaluated by the staff either within the standard content material below or immediately following.

The following portion of this technical evaluation section is reproduced from Attachment A to Appendix 19.A of the VEGP SER:

> *General Information on How the VEGP Units 3 and 4 Design and Mitigation Strategies Meet 10 CFR 50.54(hh)(2)*
>
> *The following section discusses issues and areas that are general in nature and applicable to the applicant acceptably meeting 10 CFR 50.54(hh)(2):*
>
> *The guidance in Nuclear Energy Institute's (NEI) 06-12 states that the pump(s) charging the fire protection system (FPS) header and the alternating current (ac)-independent portable pump should be housed in structures at least 100 yards from the nearest target building. The applicant's May 29, 2009, MSD submittal states that the pumps meet that criterion. The applicant has committed that the portable pump will have sufficient pump head and flow rate to deliver the flow necessary for the mitigating strategies for which it is credited. In response to request for additional information (RAI) 19-30, the applicant stated it would modify the submittal to stipulate that the detailed design of the portable pump would consider friction losses in the piping system and the needed pump head so that appropriate water flow is supplied to the SFP. This will be accomplished prior to the initial fuel load. The staff considers this RAI resolved pending the incorporation of this change in a future revision of the MSD. This is* ***Confirmatory Item 19.A-2****.*
>
> *Resolution of Standard Content Confirmatory Item 19.A-2*
>
> *Confirmatory Item 19.A-2 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-2 is now closed.*
>
> *As specified in NEI 06-12, the applicant committed to provide an ac-independent portable pump, which will have a fuel tank capacity of at least 12 hours. The portable pump is relied on by a number of mitigation strategies to meet 10 CFR 50.54(hh)(2) requirements. The applicant stated that the portable pump will deliver sufficient flow (as defined by the particular credited strategy), with sufficient head to overcome line losses between the source of water for the pump and delivery to the intended destination (e.g., the SFP) by the chosen path (e.g., through piping to wall-mounted nozzles surrounding the SFP). It can draw water from sources such as the redundant FPS storage tanks (each a minimum of 300,000 gallons), ancillary water storage tank (780,000 gallons), the condensate storage tank (325,000 gallons), the demineralized water storage tank*

(100,000 gallons), or the cooling tower basin (greater than 5,000,000 gallons [per unit]*).*

The guidance in DC/COL-ISG-016 discusses the importance of clearly labeling or tagging plant equipment that would be used to implement B.5.b strategies. In RAI 19-17 the staff asked the applicant to address this issue. In its responses dated October 29, 2009, and December 23, 2009, the applicant stated it would modify its submittal to add a commitment to provide direction on unique identification of equipment, clearly marking the LOLA-specific equipment with reflective signs or other designators, and mapping the guidance to the items required to implement the LOLA strategies. This work will be completed prior to initial fuel load. The staff considers this to meet the guidance in DC/COL-ISG-016 and considers this RAI resolved pending the incorporation of this change in a future revision of the MSD. This is ***Confirmatory Item 19.A-3****.*

Resolution of Standard Content Confirmatory Item 19.A-3

Confirmatory Item 19.A-3 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-3 is now closed.

In RAI 19-20 the staff asked the applicant to commit to perform a walk-through of written procedures identified as applicable to 10 CFR 50.54(hh)(2) prior to the initial fuel load. In its response dated October 29, 2009, the applicant stated it would modify its submittal to add a commitment to perform a walk-through of the procedures that will be accomplished prior to the initial fuel load. The staff considers this commitment meets the guidance of DC/COL-ISG-016 and considers this RAI resolved pending the incorporation of this change in a future revision of the MSD. This is ***Confirmatory Item 19.A-4****.*

Resolution of Standard Content Confirmatory Item 19.A-4

Confirmatory Item 19.A-4 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-4 is now closed.

The guidance in NEI 06-12 specifies that evaluators, decision makers, and implementers (as well as operators) be appropriately trained for a LOLA event. In RAI 19-70 the staff asked the applicant to modify its submittal to address training for evaluators, decision makers, and implementers, not just plant licensed operators. In its response dated October 29, 2009, the applicant stated it would modify the submittal to add a commitment that training material for evaluators, decision makers, and implementers will be developed using the Systematic Approach to Training (SAT). The applicant stated that training on mitigation strategies is incorporated into initial and requalification licensed operator training programs. This training will be completed prior to initial fuel load. The staff finds the applicant followed the guidance in NEI 06-12, and *considers this RAI resolved, pending the incorporation of this change in a future revision of the MSD. This is **Confirmatory Item 19.A-5**.*

Resolution of Standard Content Confirmatory Item 19.A-5

Confirmatory Item 19.A-5 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-5 is now closed.

The staff asked the applicant in RAI 19-16 to address how the strategies and procedures proposed for core cooling and other areas apply for a plant in the shutdown mode. In its response dated November 13, 2009, the applicant stated the strategies discussed in the Mitigative Strategies Table (MST) in the MSD are flexible enough to be applicable in the shutdown mode. In its amended response on February 5, 2010, the applicant noted that the same key safety functions are required when the AP1000 is in shutdown mode as when it is in operation, and the same mitigating strategies also apply. The staff considers this RAI resolved.

The February 25, 2005, guidance letter calls for an evaluation of ways to limit the spread of combustible liquids. In response to RAI 19-65, dated November 13, 2009, as revised February 5, 2010, the applicant stated that the AP1000 equipment relied on for safe shutdown is located inside the Nuclear Island, which consists of the shield building and the auxiliary building. Also, the AP1000 design is extensively compartmentalized into separate fire areas and fire zones. The three-hour fire barriers that provide compartmentalization are reinforced concrete walls, many of which are 24-inches thick. The design is further compartmentalized to provide complete separation of the radiological and non-radiological portions of the Nuclear Island. In addition, the applicant stated that analysis of structural components did not predict perforation due to impact of a commercial airplane, and therefore assessment of the effects of burning jet fuel on equipment in the containment was not required. The design and location of 3-hour fire barriers, including fire doors and security doors, within the auxiliary building are key AP1000 design features for the protection of equipment to manually actuate the passive core cooling system from the impact of a large, commercial aircraft. The applicant's assessment credited the design and location of fire barriers (including doors) as described in Appendix 9A of VEGP COL FSAR Chapter 9 to limit the effects of internal fires created by the impact of a large, commercial aircraft. The staff considers this RAI resolved.

The February 25, 2005, guidance letter states that equipment relied upon to implement the strategies required by 10 CFR 50.54(hh)(2) should be maintained and periodically tested to ensure it will operate when called upon. In RAI 19-98, the staff asked the applicant to discuss the maintenance program and controls for equipment credited for meeting 10 CFR 50.54(hh)(2). In its response dated May 24, 2010, the applicant stated that maintenance activities for mitigative strategies equipment that is also used to support plant operations will be covered by established maintenance and testing requirements for the systems that include this equipment. The applicant stated it will revise the MSD to state that *additional maintenance activities will be developed to cover active equipment that is only used to support mitigative strategies and is not used otherwise to support plant operations. The maintenance activities will include periodic surveillance checks, start and run checks, and flow tests. The staff finds this acceptable because a maintenance program will exist for all active equipment credited in the strategies, and considers this RAI resolved, pending the incorporation of this change in a future revision of the MSD. This is **Confirmatory Item 19.A-6**.*

<u>Resolution of Standard Content Confirmatory Item 19.A-6</u>

Confirmatory Item 19.A-6 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-6 is now closed.

*Guidance in NEI 06-12 specifies that an applicant include guidance or information in site procedures for operations staff or offsite resources so that they can mitigate or restore core cooling following a LOLA event. Since many LOLA procedures can only be completed as plant construction nears completion, in RAI 19-11 the staff asked the applicant to provide commitments to develop and implement these procedures prior to initial fuel load. In its response dated October 29, 2009, the applicant committed to develop and implement these procedures. The staff considers this RAI resolved, pending the incorporation of this change in a future revision of the MSD. This is **Confirmatory Item 19.A-7**.*

<u>Resolution of Standard Content Confirmatory Item 19.A-7</u>

Confirmatory Item 19.A-7 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-7 is now closed.

NRC inspections at operating reactors have shown that some mitigating strategies may require connecting portable equipment, such as fire hoses or electrical devices that may not normally be connected or tested. They might also require connections between onsite equipment and equipment provided by offsite responders. In RAI 19-103, the staff asked the applicant to modify the commitment made in its October 29, 2009, response to RAI 19-20 by revising the commitment to confirm by engineering evaluation or a demonstration prior to the initial fuel load that: (1) hoses can be connected to each other and to pumps, adapters, and fittings; (2) electrical cables, connectors, and jumpers are compatible; and, (3) fire hoses and nozzles can be attached to lifting devices and/or secured in place as needed. In its response dated May 24, 2010, the applicant stated that the response to RAI 19-90 includes a revised commitment to verify these things. Additionally, in RAI 19-90, the staff asked the applicant to address suction supply piping. Various performance attributes in NEI 06-12 specify that an applicant should have an adequate amount of suction supply piping to allow the portable pump to be located as proposed in the mitigation strategies. In its response dated May 5, 2010, the applicant proposed a revision *to the submittal and committed, as part of the walk-through that will be performed for each procedure to validate the guidance, to verify prior to initial fuel load that hose sizes and lengths, pumping capability, and availability of supply piping adequately support mitigative strategies. The staff finds that this commitment meets the guidance of NEI 06-12. The staff considers RAIs 19-90 and 19-103 resolved, pending the incorporation of these changes in a future revision of the MSD. This is* ***Confirmatory Item 19.A-8****.*

Resolution of Standard Content Confirmatory Item 19.A-8

Confirmatory Item 19.A-8 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-8 is now closed.

The guidance in NEI 06-12 specifies that an applicant address competing demands on the FPS, if the FPS is simultaneously required to fight a fire and also provide a suction source for SFP makeup, SFP spray, core cooling, containment cooling, or fission product release reduction. In RAI 19-14 the staff asked the applicant to address this concern for LOLA events. In its response dated October 29, 2009, the applicant stated that Section 5.1.4 of the MSD addresses fire protection management and states that procedures and guidelines will be developed to manage the FPS, including isolation of fire headers inside structures. In response to other RAIs (e.g. RAI 19-25) the applicant addressed pump flow and pump head adequacy as well as the capability of the FPS to operate for two hours using the diesel-driven fire pump, which is designed to provide sufficient water to supply the largest sprinkler header in the auxiliary building, the annex building, or the radwaste building; plus 500 gallons per minute (gpm) to feed fire hoses; and an additional 500 gpm that can provide makeup to the SFP. The staff considers that the guidance of NEI 06-12 is met and considers this RAI resolved.

The guidance in NEI 06-12 specifies that an applicant address how it evaluated the effect of proposed strategies and procedures on the safety and security of the plant. This guidance was provided in recognition that strategies and procedures implemented in isolation can result in unintended consequences. In RAI 19-15 the staff asked the applicant to address this issue. In its response dated December 23, 2009, the applicant stated that it would modify the submittal to add a commitment that LOLA procedures and guidance will be walked down and validated prior to initial fuel load, and any negative impacts on security and/or operations that are identified will be corrected. The staff found this acceptable and considers this RAI resolved, pending the incorporation of this change in a future revision of the MSD. This is ***Confirmatory Item 19.A-9****.* [WLS, in its response dated March 31, 2010, stated that the procedures and guidance that implement the LOLA strategies are focused on maintaining both nuclear safety and site security. Also, the applicant stated that the procedures and guidance will utilize a validation process similar to the process for Severe Accident Management Guidelines (SAMGs) to ensure that safety and security functions are established and maintained. The staff found this acceptable, because it meets the guidance in NEI 06-12 that an applicant addresses how it evaluated the effect of proposed strategies and procedures on the safety and security of the plant. The staff considers this RAI resolved.]

Resolution of Standard Content Confirmatory Item 19.A-9

Confirmatory Item 19.A-9 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-9 is now closed.

Guidance in NEI 06-12 specifies that an applicant discuss its search for area(s) in the plant where a LOLA event could cause damage such that both SFP cooling and reactor core cooling are simultaneously affected. In its response dated December 23, 2009, to RAI 19-64, [WLS response dated March 31, 2010, to RAI 19-79] *the applicant stated that there were no areas identified in the AP1000 design where, based on the guidance in NEI 06-12, both the cooling for the SFP and for the reactor core could be affected. The primary components and water sources required for core cooling are contained within containment, and the primary components and water sources required for SFP cooling are located outside of containment. The staff finds the applicant's response to RAI 19-64* [WLS RAI 19-79]*, proposing to revise the MSD to state that there are no areas where a LOLA event would cause a simultaneous loss of coolant accident (LOCA) in the RCS and a loss of SFP cooling, meets the guidance of NEI 06-12. The staff considers this RAI resolved, pending the incorporation of this change in a future revision of the MSD. This is* ***Confirmatory Item 19.A-10****.*

Resolution of Standard Content Confirmatory Item 19.A-10

Confirmatory Item 19.A-10 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-10 is now closed.

Evaluation of WLS RAIs

In addition to the site-specific RAIs, the NRC staff issued RAIs to the WLS applicant that mirrored the RAIs issued to the VEGP applicant. For other RAIs, the WLS applicant endorsed the VEGP applicant's responses. Specifically, the following RAIs issued to the WLS applicant correspond to RAIs issued to the VEGP applicant.

WLS RAI	WLS RAI Response Date	Corresponding VEGP RAI
19-17	3/31/2010	19-11
19-20	3/31/2010	19-14
19-21	3/31/2010	19-15
19-22	3/31/2010	19-16
19-23	3/31/2010	19-17
19-25	3/31/2010	19-20
19-30	3/31/2010	19-25

19-35	3/31/2010	19-30
19-65	3/31/2010	19-65
19-70	3/31/2010	19-70
19-79	3/31/2010	19-64(b)
N/A	[a]	19-90
N/A	[a]	19-98
N/A	[a]	19-103

[a] Endorsed 11/4/2010

The NRC staff compared the responses provided by the WLS applicant to the responses provided by the VEGP applicant, and concluded that the responses are essentially identical. Therefore, the conclusions reached by the NRC staff regarding general information on how the VEGP Units 3 and 4 design meets 10 CFR 50.54(hh)(2) are applicable to WLS.

The following portion of this technical evaluation section is reproduced from Attachment A to Appendix 19.A of the VEGP SER:

Phase 1 - Summary of Technical Information for Fire Fighting

The Phase 1 assessment under 10 CFR 50.54(hh)(2) focuses on enhancement of a plant's fire fighting response capability to respond to a LOLA event. Enhancement is based, in part, on pre-thinking how plant operations staff, other plant staff (including security), the utility, and nearby resources (such as local fire departments and law enforcement) can work together to mitigate a LOLA event. Also included are consideration of communication enhancement, and command and control.

The details of the VEGP Units 3 and 4 firefighting capabilities for design basis fire events are provided in Section 9.5.1 and Appendix 9A in the VEGP COL FSAR. These sections provide the following details regarding the fire protection program, including the FPS and the plant's fire brigade. The FPS is designed to be able to fight a design basis fire and simultaneously meet the needs of equipment relied upon in meeting the requirements of 10 CFR 50.54(hh)(2) for either SFP cooling or reactor core cooling. The FPS draws water from two large fire water storage tanks. The water is pumped by either of two 100-percent capacity pumps (motor-driven or diesel-driven). The FPS is sized so that it contains sufficient water for two-hour operation of the largest sprinkler system plus a 500 gpm manual hose stream allowance to support fire suppression activities plus 500 gpm flow for makeup to [reactor core] *or spray of 400 gpm to the SFP. The underground fire water yard main loop supply piping has post indicator valves (exterior to structures) that allow sectionalized control and isolation of portions of the loop, standpipe, and hose stations. The motor-driven and diesel-driven pumps are in separate buildings. The FPS diesel-driven pump and water storage tanks are located greater than 100 yards from all target areas. There are redundant flow paths that are capable of supplying water from the yard main loop to each building.*

One of the Phase 1 firefighting strategies is to develop an alternate means to charge (i.e., fill and pressurize) the FPS yard main loop in the event the normal *supply source (e.g., a fire water storage tank) is lost. The MSD submittal describes the underground yard ring header as designed to provide a minimum of two external connections that can be used to connect an external water source and pumping capability. For VEGP Units 3 and 4, the external water sources are identified as the cooling tower basins, condensate storage tanks, and other tanks with stored water. The portable pump or a fire truck will be used to take suction from the basins or tanks to pressurize the ring header*.

The applicant identified the Production Warehouse and Fire Training Facility [WLS identified the Training Center and the Maintenance Support Building] *as the staging areas for the firefighting and operations staff. The applicant stated that the staging areas will be greater than 100 yards from all target areas and that additional staging areas will be established as necessary, and documented in its procedures. These staging areas would support offsite responders and a large number of vehicles. The applicant will also provide staging areas for triage of mass casualties. For VEGP Units 3 and 4, the applicant identified the primary assembly area for evacuated and responding plant personnel as the Training Facility* [WLS identified its Maintenance Support Building]*, and an alternate assembly area as the Ebenezer Church parking lot* [WLS identified Warehouses 1 and 2]*.*

The applicant identified multiple onsite fire brigade equipment locations, each of which contains the necessary equipment for the brigade to dress out, communicate, and respond to a fire. Its procedures will maximize the survivability of fire brigade personnel by relocating them to a safe location if timely pre-event notification is made. The applicant identified the following training for firefighting personnel at VEGP Units 3 and 4. Fire brigade personnel will receive accelerant-fed fire training, and training on the coordinated fire response between onsite and offsite fire responders to help delineate roles and responsibilities during a response to a LOLA event. Site familiarization training will be provided to local offsite responders. Table top exercises postulating a LOLA event will be conducted prior to the initial fuel load and periodically thereafter involving offsite fire responders, onsite fire brigade, and operations staff.

The potential use of air-lifted resources located within two hours of the site was considered by the applicant and is documented in VEGP Units 1 and 2 letters NL-05-0946, 05-1859, 06-0333, and 06-0736. [For WLS, the applicant stated that in the event of a LOLA, the Emergency Operations Facility (EOF) would be available and staffed with personnel. This resource could be used by the applicant to secure aircraft and transport equipment as required to mitigate the event. The EOF maintains a list of support services and equipment that is available through other Duke sites and outside agencies.]

The applicant stated that its command and control (C&C) protocols are in place and specify that the site maintains overall command authority for onsite firefighting actions so that firefighting priorities are defined by operations personnel (i.e., a licensed operator) and then communicated to the offsite incident commander. Its procedures integrate the onsite LOLA response *strategies into the C&C protocols. The applicant stated that protocols have been or will be established with offsite resources (including at the county and state level) to obtain assistance and/or resources with skills and equipment not possessed by VEGP. A total 2-hour (door-to-door) response time was used by the applicant to screen potential assistance.*

The applicant identified the following communication strategies. Callout procedures will be in place, and plant response personnel will be notified via auto dialers, supplemented by pagers if there is a LOLA event. Radio interoperability is achieved between onsite and offsite responders by pairing site personnel with radios with offsite responders or by issuing site radios to offsite responders. Radios are located far enough from target areas for their survival and in an area where their access is convenient to offsite responders.

The applicant stated that the VEGP Units 3 and 4 design provides communication equipment, (such as radios, cell phones, etc.) to facilitate the response to a LOLA event and that radios are particularly important in implementing command and control. This design also includes the addition of items that would support internal communication systems(s) such as repeaters, antennas, backup power sources, leaky coax cables, etc.

<u>*Staff Evaluation of Fire Fighting Capabilities*</u>

The primary guidance for meeting the expectations of Phase 1 is provided by the February 25, 2005, letter to currently operating reactor licensees. The letter discusses firefighting guidance, which encompasses various firefighting response and support strategies including command and control, incident response training, and assurance of adequate resources to mitigate a LOLA event.

In its submittal, the applicant identified the Production Warehouse and Fire Training Facility [WLS identified its Training Center or the Maintenance Support Building] *as the staging areas for the firefighting and operations staff. In RAI 19-47, the staff requested that the applicant clarify whether and where dispersal will be directed by procedure for each level of threat warning. In its response dated December 23, 2009, the applicant stated that the imminent threat procedure for VEGP Units 1 and 2 has guidance for the relocation of fire brigade personnel based on the level and timing of the threat warning, and the VEGP Units 3 and 4 imminent threat procedure will have similar relocation guidance. The applicant stated it would modify the submittal to stipulate the applicant would develop procedures that will include a requirement for dispersion of plant staff and list the locations of the staging areas. The staff finds the applicant's response follows the guidance of the February 25, 2005, letter and considers this RAI resolved, pending the incorporation of this change in a future revision of the MSD. This is* ***Confirmatory Item 19.A-11****.* [WLS, in its response dated March 31, 2010, stated that procedures maximize survivability of the fire brigade personnel by relocating them to a safe location, such as the Training Center or the Maintenance Support Building, given pre-event notification. The staff finds this acceptable, because the applicant’s response follows the guidance of the February 25, 2005 letter that an applicant have procedures for the relocation of fire brigade personnel based on the level and timing of the threat warning. The staff considers this RAI resolved.]

<u>Resolution of Standard Content Confirmatory Item 19.A-11</u>

Confirmatory Item 19.A-11 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-11 is now closed.

In its submittal, the applicant discussed how it met the guidance in the February 25, 2005, letter that helps ensure the licensee is aware of nearby offsite organization resources and how these resources fit into the coordinated response strategy including the use of "specialized capabilities" such as debris removal equipment (bulldozers, large cranes, etc.) and specialized firefighting equipment (e.g., aqueous film forming foam (AFFF)). In RAI 19-50, the staff requested that the applicant provide a commitment that prior to initial fuel load it will re-evaluate offsite organizations, including associated memoranda of understanding (MOUs), that could significantly enhance needed skills, equipment, or abilities should a LOLA event occur. The applicant stated it would modify the submittal to include this commitment in its October 29, 2009, response. The staff finds that the applicant met the guidance of the February 25, 2005, letter and considers this RAI resolved, pending the incorporation of this change in a future revision of the MSD. This is ***Confirmatory Item 19.A-12****.*

<u>Resolution of Standard Content Confirmatory Item 19.A-12</u>

Confirmatory Item 19.A-12 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-12 is now closed.

In its submittal, the applicant discussed how it met the guidance in the February 25, 2005, letter that addresses the concern that a LOLA event could lead to a large enough number of casualties that it would challenge the established onsite medical/health services. In RAI 19-56, the staff requested that the applicant describe the location of the specific triage areas, and discuss their approximate size and capability to handle mass casualties and personnel assembly. In its response dated December 23, 2009, the applicant stated it would modify the submittal to add a commitment establishing procedures and guidance prior to initial fuel load with criteria for determining an appropriate site for triage area(s). The staff finds that the applicant met the guidance of the February 25, 2005, letter and considers this RAI resolved, pending the incorporation of this change in a future revision of the MSD. This is ***Confirmatory Item 19.A-13****.*

<u>Resolution of Standard Content Confirmatory Item 19.A-13</u>

Confirmatory Item 19.A-13 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-13 is now closed.

In its submittal, the applicant discussed how it met the guidance in the February 25, 2005, letter to identify a location(s) to be used for congregating offsite responding personnel. In RAI 19-57, the staff requested that the applicant document its criteria for these assembly area location(s). In its response dated December 23, 2009, the applicant stated it would modify the submittal to stipulate that the Training Facility is greater than 2 miles away from the plant site target areas and the Ebenezer Church is greater than 5 miles away [WLS stated its Maintenance Support Building and Warehouses 1 and 2 are greater than 100 yards away from the plant site target areas.] *These assembly areas will not be impacted by a LOLA event because of their distance from the plant. The staff finds the locations of the assembly areas meet the guidance of the February 25, 2005, letter and considers this RAI resolved, pending the incorporation of this change in a future revision of the MSD. This is* ***Confirmatory Item 19.A-14****.*

Resolution of Standard Content Confirmatory Item 19.A14

Confirmatory Item 19.A-14 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-14 is now closed.

In its submittal, the applicant discussed how it met the guidance to develop a means to supply the fire protection ring header using off-site resources and a portable pump. Guidance in NEI 06-12 specifies that an applicant provide at least two locations, at least 100 yards apart, where the fire header can be charged by the portable pump drawing from a water source. The staff evaluated the adequacy of water sources identified by the applicant, the proposed means for feeding the fire protection ring header, and procedure development. The staff finds the applicant's submittal meets the guidance of the February 25, 2005, letter.

In its submittal, the applicant discussed its assessment of mutual aid firefighting assets and discussed how it met the guidance to identify airlifted resources (personnel and equipment) for firefighting. The staff evaluated whether the applicant appropriately studied the use of airlifted resources via regional or local airports (in addition to reliance on ground response) and whether the evaluation looked beyond a 30-minute response time. In addition, the NRC considered whether there were additional regional or local airports not considered by the applicant. The applicant identified several facilities with airlift capability within a 2-hour response time. The identification and evaluation of these airports is documented in VEGP Units 1 and 2 letters NL-05-0946, NL-05-1859, NL-06-0333, and NL-06-0736. The staff finds the applicant has followed the guidance of the February 25, 2005, letter.

Memorandum of Understanding with Mutual Aid Responders

In its submittal, the applicant discussed how it met the guidance of the February 25, 2005 letter to revisit existing MOUs with mutual aid responders. In RAI 19-51, the staff requested the applicant to (a) describe in the table, WLS Units 1 and 2's expectation of the time it will take for Class B fire extinguishing equipment/supplies (e.g., AFFF) to be provided door to door, and (b) if any MOU is currently in force, identify the MOU and equipment and resources associated with each in force MOU at the time. In its response dated, March 31, 2010, the applicant stated that South Carolina law requires that state, county, and municipal governments cooperate in developing and maintaining a plan for mutual assistance. The applicant stated as a signatory to the Mutual Aid Agreement, Cherokee County can request additional assistance to support firefighting activities. Also, the applicant stated they will maintain Class B fire extinguishing equipment/supplies on site, and additional equipment can be provided by Cherokee County. The applicant stated the City of Spartanburg can be contacted for further assistance and resources. The applicant stated that based on the close proximity of firefighting resources, airlifted resources are not required. The staff finds that the applicant met the guidance of the February 25, 2005, letter and considers this RAI resolved.

The following portion of this technical evaluation section is reproduced from Attachment A to Appendix 19.A of the VEGP SER:

> *In RAI 19-61, the staff requested the applicant to provide assurance that protective measures and equipment have been specifically considered. This was to include the emergency planning (EP) program, which ensures that on-site and off-site responders will be protected in areas that have suffered a loss of a radiation barrier(s) and areas with a complete or partial loss of structural support capability. Assurance considerations included a list of equipment (e.g., poles for structural support, anti-radiation blankets for radiation protection, etc.) readily available onsite or easily attainable from off-site venders within 30 minutes. In its response dated October 29, 2009, the applicant stated that in the event of a LOLA, the Emergency Director (ED) will utilize any materials onsite that could be used to erect temporary radiation shields for responders.* [WLS stated that in the event of a LOLA, the Emergency Coordinator has the authority and ability to utilize on-site materials and requisition materials from offsite sources to minimize radiological risk and exposure to on-site and off-site responders.] *Lead blankets, different length and size poles, steel framing, wooden and steel studs, dry wall, and other materials are typically in the warehouse and will be available to the ED. Also, the ED can request materials from offsite suppliers if it is determined that sufficient materials cannot be found onsite. The staff finds this response acceptable because it documents onsite/offsite resources that can protect responders to a LOLA event. The staff considers this RAI resolved.*
>
> *In its submittal the applicant discussed the role other regional resources would play in a fully preplanned mobilization effort. In RAI 19-52, the staff requested that the applicant clarify what was meant by the on-site firefighting equipment being staged in "appropriate locations obviating the need for pre-staging equipment at local fire departments." The staff also requested that the applicant provide the criteria used in determining why a chosen location is "appropriate."*

In its response dated October 29, 2009, the applicant stated that these "appropriate locations" are the dress-out locations for the fire brigade. The applicant stated it would modify the submittal to stipulate that these locations will be identified and established when the VEGP Units 3 and 4 procedures/guidance for mitigative strategies are developed. As stated in the response to RAI 19-48, Burke County EMA has extensive firefighting resources, and they will bring those resources to the plant in response to a LOLA event. [The WLS applicant stated that for WLS Units 1 and 2, on-site firefighting equipment is staged in appropriate locations (areas that are more than 100 yards from key target areas). The applicant committed to identifying and establishing the appropriate locations when the WLS Units 1 and 2 procedures/guidance documents for mitigative strategies are developed.] *The applicant's response met the guidelines of the February 25, 2005, guidance letter* [for staging firefighting equipment and the guidance in NEI 06-12 for those locations being appropriate because they are more than 100 yards from key target areas]*. The staff considers this RAI resolved, pending the incorporation of this change in a future revision of the MSD. This is* ***Confirmatory Item 19.A-16****.*

Resolution of Standard Content Confirmatory Item 19.A-16

Confirmatory Item 19.A-16 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-16 is now closed.

In RAI 19-53, the staff requested that the applicant provide a commitment to update the coordination with offsite local fire departments and other regional firefighting aid organizations. In its response dated October 29, 2009, the applicant stated it would modify the submittal to add a commitment to update the coordination prior to the initial fuel load for VEGP Units 3 and 4. The staff finds this response meets the guidance of NEI 06-12 and considers this RAI resolved, pending the incorporation of this change in a future revision of the MSD. This is ***Confirmatory Item 19.A-17****.*

Resolution of Standard Content Confirmatory Item 19.A-17

Confirmatory Item 19.A-17 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-17 is now closed.

The following portion of this technical evaluation section is reproduced from Attachment A to Appendix 19.A of the VEGP SER:

In RAI 19-45, the staff requested that the applicant state if procedures will require the firefighting brigade to disperse at least 100 yards from all target areas and specifically not to return to target areas. In addition, the staff requested that the applicant address or commit to address the minimum performance-based number of fire brigade and operations personnel you plan to locate at each *staging area. In its response dated December 23, 2009, the applicant stated that the fire brigade members will be directed to the staging areas identified in the MST in the MSD, and remain there until directed to respond. These staging areas are located more than 100 yards from target areas. The required number of fire brigade members (as defined in Section 13.1.2.1.5 of the VEGP COL FSAR) will respond to the appropriate staging areas. The staff finds the applicant's response follows the guidance of the February 25, 2005, letter and considers this RAI resolved.*

*The submittal addressed provisions for controlling a large number of emergency response vehicles that may arrive at the plant should a LOLA event occur. The staff reviewed whether: (1) staging areas for the offsite responders were identified; (2) provisions were made with the Local Law Enforcement Agency (LLEA) to ensure no restrictions are placed on emergency vehicle arrivals; (3) site familiarization or training was provided to the LLEA; (4) plant procedures document protocols with LLEA were or would be in place; and (5) sufficient dosimetry would be available for arriving offsite assets. In RAI 19-54, the staff requested that the applicant further discuss the provisions for controlling a large number of emergency response vehicles that may arrive at the plant in response to a LOLA event. The applicant was also requested to provide the criteria for selecting each satisfactory staging area(s). In its response dated October 29, 2009, the applicant stated it would modify the submittal to add a commitment to establish and document staging areas prior to initial fuel load. It also provided the selection criteria, which are based on the expected volume and type of vehicles and the proximity of the areas to the plant. The applicant stated it has coordinated with LLEAs to ensure that access of responders is controlled. Local law enforcement is provided with site familiarization training (overall layout of site, access points, staging areas, etc.) and is kept informed of site LOLA response strategies. Protocols for interacting with LLEA are documented in site procedures. An evaluation will be performed by the applicant to determine the number of dosimeters needed for initial arriving offsite response personnel expected to be involved during a LOLA event. This resultant number of dosimeters will be staged at a location that is expected to survive a potential LOLA event. The staff finds the guidance of the February 25, 2005, letter is met and considers this RAI resolved, pending the incorporation of this change in a future revision of the MSD. This is **Confirmatory Item 19.A-18**.*

Resolution of Standard Content Confirmatory Item 19.A-18

Confirmatory Item 19.A-18 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-18 is now closed.

The applicant's submittal discussed meeting the guidance on command and control functions needed to ensure responding assets follow preplanned strategies. The staff evaluated whether: (1) command and control protocols are proceduralized; (2) the licensee maintains overall command authority at all times; (3) onsite and offsite response strategies are factored into the command and *control protocol; and (4) the licensee provides technical assistance to the offsite responders. In RAI 19-49, the staff requested that the applicant discuss how the procedures provide a framework for providing technical direction to offsite responders. In its response dated October 29, 2009, the applicant stated it would revise the MSD to state that the firefighting priorities will be defined by operations. Command and control procedures will specify that the site maintains overall command authority for onsite firefighting actions at all times to ensure that firefighting priorities, as defined by operations (i.e., a licensed operator), are communicated to the incident commander. The individual providing direction will be a licensed operator* [For WLS, a Fire Brigade Commander, or Shift Manager]. *The staff considers that the guidance of the February 25, 2005, letter is met and considers this RAI resolved, pending the incorporation of this change in a future revision of the MSD. This is* ***Confirmatory Item 19.A-19****.*

Resolution of Standard Content Confirmatory Item 19.A-19

Confirmatory Item 19.A-19 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-19 is now closed.

The applicant's submittal addressed communication enhancements. Guidance in the February 25, 2005, letter applies specifically to the communications with the offsite responders for firefighting purposes. The applicant stated that plant response personnel are notified via autodialers and supplemented by pagers. Radio interoperability between onsite and offsite responders is achieved by pairing site personnel holding site radios with offsite responders. Radios needed to support firefighting response are provided in an appropriate location. These radios are not radios also earmarked for operational recovery response. At least one location where firefighting response radios are stored is expected to survive a potential LOLA event. At least one location where firefighting response spare batteries and chargers are stored is expected to survive a potential LOLA event. The criteria used to select the appropriate locations for staging the radios and batteries will be included in the VEGP MSD. The staff finds these strategies meet the guidance in the February 25, 2005, letter. However, in RAI 19-24 the staff asked the applicant, based on NEI 06-12 guidance, to provide a commitment that communication for responders who have to enter robust buildings following a LOLA event will be enhanced. In its response dated October 29, 2009, the applicant proposed modifications to the submittal to more clearly state that it would provide these enhancements. The staff considers this RAI resolved, pending the incorporation of this change in a future revision of the MSD. This is ***Confirmatory Item 19.A-20****.*

Resolution of Standard Content Confirmatory Item 19.A-20

Confirmatory Item 19.A-20 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-20 is now closed.

*In RAI 19-*55*, the staff requested that the applicant (a) make a commitment to determine the number of radios needed to support firefighting responders, (b) provide the criteria to determine staging locations for radio(s), (c) explain why only one firefighting response radio needs to survive a potential LOLA event at VEGP Units 3 and 4, (d) address how and where spare batteries or chargers are to be provided/stored,* [and] *(e) discuss how its communication enhancements are designed to deal with concurrent loss of offsite power at the site* [WLS RAI 19-57 is a modified version of VEGP Units 3 and 4 RAI 19-55 as follows: the staff requested that the WLS applicant (a) make a commitment to perform an evaluation to determine the number of radios needed to support firefighting responders, (b) address how and where spare batteries or chargers are to be provided/stored, and (c) discuss how its communication enhancements are designed to deal with concurrent loss of offsite power at the site]*. In its response dated October 29, 2009* [WLS response dated 3/31/2010]*, the applicant provided the criteria to determine the number and staged location of radios. The applicant stated one firefighting response radio is sufficient to initiate communications between the onsite fire brigade commander and the responding fire fighters. The applicant indicated that chargers and spare batteries are stored at the security pavilion* [WLS Maintenance Support Building] *and also in the dress-out areas* [WLS Training Building]*. Each of these areas is greater than 100 yards from the target areas. The hand-held communications equipment operates using batteries and the repeater has secondary power that comes from an uninterruptible power supply (UPS). The applicant stated it would modify the submittal to add a commitment that the number and locations of radios would be included in LOLA procedures and guidance documents. The staff considers that this response meets the guidance in the February 25, 2005, letter, and considers this RAI resolved, pending the incorporation of this change in a future revision of the MSD. This is* ***Confirmatory Item 19.A-21****.*

Resolution of Standard Content Confirmatory Item 19.A-21

Confirmatory Item 19.A-21 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-21 is now closed.

In RAI 19-74 the staff asked the applicant to discuss plans on how communications will be restored following a LOLA event. In its response dated October 29, 2009, the applicant stated it would modify the submittal to stipulate that efforts to restore communications will be focused on the most rapid success path, the most effective communication pathway, and the ability to communicate with the broadest set of resources to ensure the most effective response. The staff found this response acceptable, and considers this RAI resolved, pending the incorporation of this change in a future revision of the MSD. This is ***Confirmatory Item 19.A-22****.*

Resolution of Standard Content Confirmatory Item 19.A-22

Confirmatory Item 19.A-22 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-22 is now closed.

The staff evaluated the level and type of training of response personnel. In its submittal, the applicant addressed providing fire brigade personnel with training on accelerant-fed fires. The concern for accelerant-fed fires is related to the potential for them to greatly exceed what has been analyzed under 10 CFR Part 50.48, "Fire protection" (which assumes fires will not affect multiple fire areas). The applicant's submittal stated that fire brigade personnel receive accelerant-fed fire training, site fire brigade personnel are trained in the application of firefighting foam, training is provided to the fire brigade on the coordinated fire response between onsite and offsite fire responders (including interface with operations) to help delineate roles and responsibilities during response, and site familiarization training with local offsite responders is offered on a recurring basis. Additionally, training of offsite responders is conducted to enhance the understanding of the coordinated response strategies for a LOLA event. Drills are conducted on large liquid fires (involving offsite fire responders, onsite fire brigade, and operations staff). In RAI 19-58, the staff requested that the applicant discuss the training of fire brigade personnel, including any tabletop exercises that will be conducted prior to the initial fuel load and periodically, as required. The applicant was also asked whether training on fighting Class B fires will be provided to brigade members. In its response dated October 29, 2009, the applicant committed that the tabletop exercise will be conducted prior to the initial fuel load and periodically, as required. In addition, the applicant said it would modify the MSD to state that training on fighting a Class B fire will be provided to the fire brigade members. The staff considers that the guidance of the February 25, 2005, letter is met by this submittal and considers the RAI resolved, pending the incorporation of this change in a future revision of the MSD. This is ***Confirmatory Item 19.A-23****.*

<u>*Resolution of Standard Content Confirmatory Item 19.A-23*</u>

Confirmatory Item 19.A-23 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-23 is now closed.

In RAI 19-21, the staff asked the applicant to provide the specifics by which it determined that the separation between critical equipment such as portable pump and radios and target areas was acceptable. In its response dated December 23, 2009, the applicant stated that equipment credited in multiple mitigative strategies will be located a minimum of 100 yards from target areas or spatially separated to meet the requirements of Phase 3 in NEI 06-12. The RAI also asked the applicant to state if it would measure distance from a target area to mitigating equipment by taking the distance from the exterior wall of the target area to the closest exterior wall of the structure surrounding the mitigating equipment. In its response to RAI 19-25, the applicant addressed the staff's *concern in RAI 19-21 indicating it would measure the distance in this manner. The staff considers the responses to meet the criteria of NEI 06-12 and considers this RAI resolved, pending the incorporation of this change in a future revision of the MSD. This is* ***Confirmatory Item 19.A-24****.*

Resolution of Standard Content Confirmatory Item 19.A-24

Confirmatory Item 19.A-24 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-24 is now closed.

In RAI 19-25 the staff asked the applicant to address how the AP1000 design would simultaneously fight a fire while providing 500 gpm injection to the SFP. In its response dated December 23, 2009, the applicant clarified and stated it would modify the MSD to stipulate that the diesel-driven fire pump is designed to provide sufficient water to supply the largest sprinkler header in the auxiliary building, the annex building, or the radwaste building; plus 500 gpm to feed fire hoses; and an additional 500 gpm, which can provide makeup to the SFP. The staff considers this response meets the guidance of NEI 06-12 and considers this RAI resolved, pending the incorporation of this change in a future revision of the MSD. This is ***Confirmatory Item 19.A-25****.*

Resolution of Standard Content Confirmatory Item 19.A-25

Confirmatory Item 19.A-25 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-25 is now closed.

Evaluation of WLS RAIs

The NRC staff issued RAIs to the WLS applicant that mirrored the RAIs issued to the VEGP Units 3 and 4 applicant. Specifically, the following RAIs issued to the WLS applicant correspond to RAIs issued to the VEGP applicant.

WLS RAI	WLS RAI Response Date	Corresponding VEGP RAI
19-20	3/31/2010	19-14
19-26	3/31/2010	19-21
19-29	3/31/2010	19-24
N/A	[a]	19-25
19-49	3/31/2010	19-45
19-50	3/31/2010	19-47
19-52	3/31/2010	19-49
19-53	3/31/2010	19-50
19-54	3/31/2010	19-52
19-55	3/31/2010	19-53
19-56	3/31/2010	19-54
N/A	N/A	19-56
19-58	3/31/2010	19-57
19-59	12/1/2009	19-58
19-61	3/31/2010	19-61
19-73	3/31/2010	19-74

[a] Endorsed 11/4/2010

The NRC staff compared the responses provided by the WLS applicant to the responses provided by the VEGP applicant and concluded that the responses are essentially identical. Therefore, the conclusions reached by the NRC staff regarding Phase 1 firefighting are applicable to WLS.

Evaluation of Site-specific Response to Standard Content

In addition to the site-specific RAIs, the applicant identified in Section 4.1 of the WLS MSD, additional external water sources for feeding the fire protection ring header not identified by VEGP. The external water sources identified by VEGP for feeding the fire protection ring header were the cooling tower basins, condensate storage tanks, and other tanks with stored water. WLS has identified the passive containment cooling ancillary water storage tank, the demineralized water tank, Pond “A,” Pond “B,” the local Cherokee County Municipal Water System (Draytonville Water District), and the waste water retention basins, in addition to the sources identified by VEGP, as external sources for providing water to the fire protection ring header. Because WLS has identified additional water inventory above that found at VEGP, the staff finds these differences do not affect its standard content conclusions.

The following portion of this technical evaluation section is reproduced from Attachment A to Appendix 19.A of the VEGP SER:

Phase 2 - Summary of Technical Information for Spent Fuel Pool Cooling

The objective of the SFP cooling strategies proposed by the applicant is to assure cooling of the spent fuel in the SFP should the water level of the pool be lowered or the pool emptied due to a LOLA event. Consideration is given to diversity of equipment, as defined in NEI 06-12, between normal and alternate *strategies. The SFP for the AP1000 is located in the "southern[1]" portion of the auxiliary building. None of the walls of the SFP are exposed to the exterior of the building. The top of active fuel in the SFP is approximately 7.75 feet above grade.*

NEI 06-12 identifies various strategies applicants should implement to mitigate damage to fuel in the SFP caused by a beyond-design basis LOLA event. The applicant developed mitigative strategies based on NEI 06-12 guidance. The applicant stated that VEGP Units 3 and 4 have design enhancements, such as the integral spray nozzles in the SFP, to facilitate the implementation of these measures in a manner to reduce reliance on operator actions compared to operating facilities.

The first strategy specified in NEI 06-12 for Phase 2 is to provide a diverse internal SFP makeup source in addition to normal makeup sources to the pool. For the AP1000 design, normal makeup is provided by the demineralized water system (DWS). The DWS pumps water into the SFP through the spent fuel pool cooling system (SFS) normal return line. The applicant identified the alternative source as from the FPS through the "West" spray header in the SFP. The flow rate to the pool will be at least 500 gpm, with water drawn from the fire water storage tank(s) that have a total capacity of 815,000 gallons. The applicant identified the equipment and support systems for this pathway. The applicant states that these pathways are physically and electrically diverse (as defined in NEI 06-12) from the normal path for SFP makeup (i.e., share no piping or components). The alternate pathway does not require operators to use fire hoses to implement SFP refill.

The second strategy specified in NEI 06-12 for Phase 2 is a diverse means of providing at least 500 gpm makeup to the SFP using a power-independent pumping capability that is external to the structure housing the SFP. The applicant stated the AP1000 design provides the capability through a combination of two flow paths. The first path uses gravity flow from the passive containment cooling water storage tank (PCCWST) (located on the roof of the shield building) to supply the "east" spray header for the SFP. The second path uses a portable ac-independent pump that takes suction from the fire water yard main loop supply piping and then it pumps water to the SFP through a flanged connection exterior to the auxiliary building that leads to the SFP makeup line. The applicant's submittal discusses the flow paths in detail, including that there are no power sources, piping, or equipment for these flow paths that are shared with the normal SFP makeup flow path. The applicant stated these flow paths meet the definition of "diverse" defined in NEI 06-12.

The third strategy specified in NEI 06-12 guidance is to establish a flexible means of providing at least 200 gpm spray to the SFP using a power-independent pumping capability. Operating plants typically address this requirement by manually routing fire hoses from the exterior of the building to the

[1] Terms dealing with the orientation of buildings or their contents may reference "north," "south," "east," or "west," referring to the relation of the structure, system, or component to the "north" orientation assumed in the AP1000 design documents.

SFP elevation. The pumping source for operating plants is a diesel-powered pump connected to the fire water supply system yard ring header via hydrants. Concerns with this approach include area access, radiation exposure to plant operators, difficulties with carrying hoses up stairwells to the SFP floor elevation, and time constraints. The applicant stated the AP1000 design includes two permanently installed SFP spray headers, each capable of providing greater than 400 gpm spray flow to the SFP. This eliminates some of the concerns associated with operating reactor strategies. Additionally, the AP1000 external spray strategy uses three spray methods. The first spray method uses gravity to feed water to the east spray header from the PCCWST until the portable pump can be connected to the external flanged connection. The second spray method uses the diesel-driven fire pump to provide spray flow from the FPS through the west spray header. The third spray method uses the portable pump taking suction from a fire hydrant, using water from the fire water storage tanks. The portable pump sends water through the flanged connection that is connected to piping that penetrates the east portion of the auxiliary building at the ground level and delivers the water to the east spray header. The applicant's submittal discusses the flow paths in detail. The applicant stated that the two spray headers provide redundancy and spatial separation to avoid damage to both during an event. Piping and valves for the SFP east spray header are located in separate rooms of the auxiliary building from the piping and valves for the SFP west spray header, thereby ensuring adequate spatial separation of the two spray systems. The rooms are separated by 24-inch thick concrete walls. No plant electrical equipment is needed to establish these flow paths through the SFP east or west spray headers. Consequently, the applicant stated that the three spray methods are diverse, as defined by the NEI 06-12 guidance.

The applicant stated the spray system consists of two spray headers, each with 16 fixed spray nozzles recessed into the east and west walls of the SFP. They are located at an approximate elevation of 134'-9" and are angled downward. The fixed spray nozzles will be preferentially directed to areas of the pool where high decay heat fuel may be staged or stored to supply the necessary spray flux for cooling. An evaluation of the fuel assembly decay heat loads and corresponding spray flow requirements, compared with the calculated spray flux pattern will be performed by the applicant to confirm adequate spray coverage in the SFP. Plant procedures will apply the results of this evaluation to direct that fuel assemblies may only be staged or stored in locations that are provided adequate spray flux to remove their decay heat.

In addition to the three strategies described, the applicant stated that VEGP Units 3 and 4 have added SFP makeup strategies. Consistent with NEI 06-12, new plants typically have several additional potential water sources and flow paths to provide makeup water to the spent fuel pool beyond those discussed in Sections 6.1, 6.2.1, and 6.2.2 of the MSD. Specifically for the AP1000 design, the applicant identified the following water sources and flow paths that provide makeup water to the SFP:

- *The chemical and volume control system (CVS) nominally provides 100 gpm of makeup to the SFP. This water is pumped using the CVS* *makeup pumps. Makeup from the CVS to the SFP is provided through a 3-inch CVS line that connects to a 10-inch SFS line.*

- *The passive containment cooling system (PCS) recirculation pumps can take suction from the passive containment cooling ancillary water storage tank (PCCAWST) to provide makeup to the SFP through either the east spray header or the PCS makeup line. Either of these flow paths provides an estimated flow of 200 gpm.*

- *A Class III hose station in the vicinity of the SFP is charged through the FPS, and can provide an estimated flow of 250 gpm through a 2-1/2 inch fire hose. The fire hose can be routed directly to the SFP.*

- *The internal SFP makeup strategy can be modified to use the motor-driven fire pump in lieu of the diesel-driven fire pump. This flow path will take suction from either fire water storage tank to provide approximately 600 gpm to the west spray header.*

- *The in-containment refueling water storage tank (IRWST) can provide water to the SFP using the SFS pumps. Permanent piping from the IRWST can be aligned to the SFS pump suction to provide approximately 1200 gpm through the SFP cooling return line.*

- *The IRWST can also be aligned to the suction of the normal residual heat removal system (RNS) pumps to inject approximately 1500 gpm of makeup through the normal RNS return line to the SFP.*

The applicant states that the primary sources of water for the alternate makeup to the SFP are the PCCWST, PCCAWST, fire water storage tank, fire water/clearwell storage tank, demineralized water storage tank, condensate storage tank, the boric acid storage tank, and the cooling tower basin.

The SFP leakage control strategies provided in NEI 06-12 are based on operational actions. The applicant stated the SFP for the AP1000 is designed to have thick, heavily reinforced concrete walls and floor. The AP1000 SFP is located above the two waste holdup tank rooms. Both of these rooms are designed to be water tight. Water tight doors are used for each waste holdup tank room so that if the pool were to drain from damage to its floor the drainage would be contained within one of the rooms. This would prevent the water level in the SFP from draining below the top of fuel in the SFP. Nonetheless, the applicant stated that appropriate SFP leakage control strategies and procedures are included in technical support center and emergency operations facility guidance for the responding group to consider during a LOLA event as possible control methods.

Staff Evaluation of Spent Fuel Pool Cooling

The staff evaluated the capabilities of the VEGP Units 3 and 4 design to provide alternate SFP makeup and to spray the SFP including evaluation of the flow *rates and the ability of the applicant to implement the strategies in a timely manner. The staff evaluated the applicant's submittal on SFP cooling against the guidance of NEI 06-12 and DC/COL-ISG-016.*

The applicant's response to RAI 19-81 addresses the staff's questions in RAIs 19-81, 19-82, 19-83, and 19-84. The staff asked the applicant in RAI 19-83 to address personnel actions to provide diverse makeup and spray to the SFP. In its response dated May 5, 2010, the applicant stated the response to RAI 19-81 provided this information. The applicant stated that the actions required to manipulate the valves needed to implement these strategies can be accomplished within two hours. The staff concurs with the applicant's referral to RAI 19-81, and considers that the guidance in NEI 06-12 is met by the response to RAI 19-81. The staff considers RAI 19-83 resolved.

The staff asked the applicant in RAI 19-84 to address the flow rates associated with the normal makeup flow path and each of the diverse flow paths. This information was provided by the applicant in response to RAI 19-81 in its May 5, 2010 letter and later clarified in its May 28, 2010 letter. The staff reviewed the flow rates and found they met the guidance in NEI 06-12. The staff considers RAI 19-84 resolved.

The first strategy reviewed by the staff was the alternate internal makeup capability to the SFP. In this strategy, as outlined in NEI 06-12, the alternate makeup pathway is to provide at least 500 gpm injection to the SFP using piping and equipment that are diverse from the normal pathway. In its May 29, 2009, submittal, the applicant described several pathways that can inject into the SFP from sources internal to the auxiliary building. In RAI 19-81 the staff asked the applicant to clarify the diverse pathways that were credited with providing injection or spray into the SFP, including a description of the valves, pumps, significant electrical equipment (including instrumentation), hoses, and piping in the pathways. In its response dated May 5, 2010, the applicant provided a single alternate makeup path that provided at least 500 gpm flow to the SFP and was diverse as defined by NEI 06-12. The applicant also proposed to modify Sections 6.1 and 6.2 of the MSD to describe the pathways from the water sources to the final injection into the SFP. The description addressed valves, pumps, tanks, piping and other significant equipment, flow rates, power requirements, operator actions, and the degree of separation/diversity. These proposed revisions address the changes associated with RAIs 19-82, 19-83, and 19-84. Strategy tables were supplied that summarize the key aspects for the SFP Makeup-Internal Strategy (Table 6-1), SFP Makeup-External Strategy (Table 6-2), and SFP Spray-External Strategy (Table 6-3). A simplified diagram was also supplied to aid in understanding the injection and spray flow paths. The staff reviewed the applicant's RAI responses regarding the alternate internal makeup capability. The staff determined that these responses provided adequate identification of the flow path and meets the guidance of NEI 06-12. The staff considers RAI 19-81 resolved.

The staff reviewed the physical and electrical separation (as defined in NEI 06-12) of the pathways. In RAI 19-82 the staff asked the applicant to *address whether or not the power supplies are diverse, as defined in NEI 06-12, for equipment that must function to inject or spray water into the SFP. In its response dated May 5, 2010, the applicant stated that as discussed in its response to RAI 19-94, in the same letter, the normal makeup to the SFP is provided by the Demineralized Water System. All valves in the alternate flow path are manually actuated, and the only significant electrical equipment required to establish the flow path is the use of one of the two Demineralized Transfer Pumps. The power source for the pumps is the normal plant power system. The response to RAI 19-81, in the same letter, provides a summary of the electrical equipment used to establish flow paths for the diverse internal and external SFP makeup strategies and spray strategy (i.e., Tables 6-1 through 6-3 in the MSD). These proposed changes to the MSD demonstrate that the electrical equipment needed is diverse and the pathways meet the definition of "diverse." The applicant's response meets the guidance in NEI 06-12 and the staff considers RAI 19-82 and RAI 19-94 resolved.*

Tracking proposed changes to the MSD in response to RAIs 19-81, 19-82, 19-83, and 19-84 is ***Confirmatory Item 19.A-26.***

Resolution of Standard Content Confirmatory Item 19.A-26

Confirmatory Item 19.A-26 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-26 is now closed.

The second strategy evaluated by the staff was the establishment of a flexible means of external SFP makeup of at least 500 gpm using a portable, power-independent pumping capability. The staff reviewed the pathways (one of which used gravity injection from the PCCWST and the other which used the portable pump) to determine if they provided flow and found the flow adequate and the pathways physically and electrically diverse as defined by NEI 06-12. The flow paths should provide at least 500 gpm of makeup flow to the SFP as stated in NEI 06-12. Although the use of the PCCWST does not rely on use of a portable pumping capability, the capability is power-independent and spatially separated. Use of the PCCWST meets one of the Commission's objectives of reducing the reliance on operator actions. The staff considers that gravity injection from the PCCWST meets the intent of the NEI 06-12 guidance.

The third strategy evaluated by the staff was the establishment of a flexible means of SFP spray using a portable, power-independent pumping capability. The staff reviewed the flow paths to determine if they are physically and electrically diverse as defined in NEI 06-12. The two paths each should provide at least 400 gpm of spray flow to the SFP, due to VEGP 3 and 4's SFP loading pattern. In its response by letter to RAI 19-81 dated May 5, 2010, clarified by letter on May 28, 2010, and further clarified in RAI 19-69 by letter on August 13, 2010, the applicant provided a description of the flow paths credited for providing at least 400 gpm spray from each header to the SFP. The staff reviewed these pathways and found that they could provide at least 400 gpm *spray flow from each header to the pool. The pathways use the portable pump or diesel-driven FPS pump to inject flow into either the "east" or "west" SFP spray headers. As a result of the loading pattern chosen by the applicant and the heavy reliance on the use of emergency spray cooling when the fuel is initially offloaded from the core both headers are needed to provide 800 gpm of flow to the SFP. The permanently discharged spent fuel assemblies will continue to cool following discharge. If it is desired to go to a single header for spray coverage or otherwise reduce spray capabilities, a confirmation of the stored fuel assembly decay heat loads and corresponding spray flow requirements is performed prior to moving fuel assemblies into SFP storage locations or reducing spray capabilities. The staff finds that the two spray headers are diverse as defined in NEI 06-12. The applicant stated at the onset of a LOLA event which results in the drain down of the SFP, the AP1000 design has the capability of providing passive spray/makeup to the pool initially through the "east" header from the PCCWST. In RAI 19-85, and further clarified in RAI 19-69, the staff asked the applicant to discuss its plans to demonstrate that the spray headers will deliver an effective spray to the SFP of at least 800 gpm or 400 gpm, whichever is appropriate depending on the SFP loading pattern at the time. In its response dated May 5, 2010, the applicant proposed to modify the MSD by adding a commitment that preoperational tests will be performed prior to initial fuel load to verify the required flow rates through each spray header and that adequate coverage of the pool is provided. The staff considers this acceptable and considers RAI 19-69, RAI 19-81, and RAI 19-85 resolved, pending the incorporation of these changes in a future revision of the MSD. This is* ***Confirmatory Item 19.A-27****.*

Resolution of Standard Content Confirmatory Item 19.A-27

Confirmatory Item 19.A-27 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-27 is now closed.

Guidance in NEI 06-12 specifies that the implementing guidance for Phase 2 should include steps to assist plant staff in determining whether to use the external strategy in the makeup or spray modes. In RAI 19-92 the staff asked the applicant to address this. In its response dated May 5, 2010, the applicant stated that it would modify the submittal to add a commitment that such guidance will be included in makeup and spray strategies prior to initial fuel load. The staff agrees that the flexible means of providing SFP cooling credited by the applicant meets the criteria of NEI 06-12. The staff therefore considers this RAI resolved, pending the incorporation of this change in a future revision of the MSD. This is ***Confirmatory Item 19.A-28****.*

Resolution of Standard Content Confirmatory Item 19.A-28

Confirmatory Item 19.A-28 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff *verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-28 is now closed.*

Guidance in NEI 06-12 calls for fuel that has been unloaded from the reactor and has the greatest decay heat to be distributed throughout the SFP in a particular pattern to lessen the chance that fuel uncovery will lead to an exothermic reaction and a potential zirconium fire. The applicant's submittal of May 29, 2009, stated the VEGP Units 3 and 4 design did not need to constrain SFP loading patterns. This was because there is spray cooling available and the rooms under the SFP are designed to be water tight so that if the water in the pool were to drain out and fill these rooms, the water level in the pool would not drop below the top of fuel in the SFP. The staff found this approach unacceptable and in RAI 19-69 asked the applicant to consider dispersing the hot fuel as a defense-in-depth measure, because the damage assumed under 10 CFR 50.54(hh)(2) is non-mechanistic and can come from any number of beyond design basis security events. In Supplement 1 to RAI 19-69, the staff also asked the applicant to provide a detailed technical basis for the conclusion that the SFP will not drain down below half the height of the fuel assemblies based on a scenario that does not credit the function of the water tight doors, as well as to consider committing to implementing the defense-in-depth SFP loading strategy originally presented in RAI 19-69 if this conclusion cannot be supported. In Supplement 2 to RAI 19-69, the staff requested additional information regarding the diverse external strategy for maintaining SFP cooling capabilities, including SFP dispersal pattern(s) and SFP spray system design and operation. Also, the staff noted that the applicant's strategy may deviate from industry and regulatory guidance, and that the primary focus should be on compliance with the regulatory requirements to maintain and restore SFP cooling capabilities. In its response dated August 13, 2010, the applicant stated that during refueling outages, the complete core (157 fuel assemblies) may be staged in a uniform pattern in Region 1 and Region 2 of the pool, in accordance with spray flux and criticality requirements. The applicant stated that since this configuration does not allow for air cooling to prevent a zirconium fire in a LOLA event with a complete loss of SFP inventory, the prevention of a zirconium fire will be accomplished by using the installed spray system on both sides of the pool. VEGP 3 and 4's loading pattern strategy requires a heavy reliance on the use of spray. A detailed description of the spray system to be utilized by the applicant due to the loading pattern chosen is already provided in this section of the SER, under the description of the third strategy evaluated by the staff for the establishment of a flexible means of SFP spray using a portable, power-independent pumping capability.

Although air cooling is neither credited nor necessary to meet the regulatory requirements of 10 CFR 50.54(hh)(2) for the AP1000 design, the capability to provide air cooling exists. The applicant stated that the capabilities to augment natural circulation air cooling of fuel assemblies in the pool in the unlikely event that the SFP inventory, makeup, and spray capabilities are lost include the availability of limited downcomer space resulting from open areas within and around the spent fuel racks, and passive ventilation of the bulk air space in the SFP area. The design features supporting passive ventilation would be enabled *by opening the roll-up doors to the staging area and the personnel doors in the stairway leading to the SFP operating deck, thereby allowing outside air to flow into the SFP area. The hot air over the pool surface would then exit the building through engineered relief panels that are designed to open manually or automatically through the activation of fusible links.*

The staff finds that the strategy of providing sufficient spray to remove decay heat is an acceptable approach for minimizing the chance that fuel uncovery will lead to an exothermic reaction and a potential zirconium fire. The staff considers this RAI resolved, pending the incorporation of these changes in a future revision of the MSD. This is ***Confirmatory Item 19.A-29****.*

Resolution of Standard Content Confirmatory Item 19.A-29

Confirmatory Item 19.A-29 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-29 is now closed.

Guidance in NEI 06-12 specifies that flexible hoses used to deliver flow to the SFP be secured in some manner. In RAI 19-86 the staff asked the applicant to address this issue. In its response dated May 5, 2010, the applicant proposed to revise Section 6.0 of the MSD to state that guidance documents will include instructions for securing flexible hoses. The staff considers the guidance in NEI 06-12 met and considers this RAI resolved, pending the incorporation of this change in a future revision of the MSD. This is ***Confirmatory Item 19.A-30****.*

Resolution of Standard Content Confirmatory Item 19.A-30

Confirmatory Item 19.A-30 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-30 is now closed.

Guidance in NEI 06-12 calls for an applicant to describe additional site-specific SFP makeup strategies using the strategies identified in Attachment A to the NEI letter from Marvin Fertel to Luis Reyes on Closure of Phase 2, January 24, 2006. In RAI 19-88 the staff asked the applicant to provide this list. In its response dated May 5, 2010, the applicant proposed to add a new section to the MSD (Section 6.2.3), which will list six additional strategies beyond those previously identified for the VEGP Units 3 and 4 design. This meets the guidance in NEI 06-12, and the staff considers this RAI resolved, pending the incorporation of this change in a future revision of the MSD. This is ***Confirmatory Item 19.A-31****.*

Resolution of Standard Content Confirmatory Item 19.A-31

Confirmatory Item 19.A-31 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff *verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-31 is now closed.*

In RAI 19-71, the staff asked the applicant to provide a list of onsite resources that could be used by the emergency response officer (ERO) to reduce or stop leakage from a damaged SFP or other pool holding spent fuel and their general location. In its response, the applicant indicated that planning for VEGP Units 1 and 2 has already developed a list of materials that are available onsite to reduce or stop leakage from a damaged pool and that it would modify its submittal to include the list of onsite resources. These materials include sheets of plywood, steel plates, aluminum plates, Plexiglas, sand bags, and inflatable bladders. The staff finds this meets the guidance in NEI 06-12 and this RAI is considered resolved, pending the incorporation of this change in a future revision of the MSD. This is ***Confirmatory Item 19.A-32****.*

Resolution of Standard Content Confirmatory Item 19.A-32

Confirmatory Item 19.A-32 was determined by staff not to be necessary. As a result, Confirmatory Item 19.A-32 is now closed and RAI 19-71 is resolved.

Evaluation of WLS RAIs

The NRC staff issued RAIs to the WLS applicant that mirrored the RAIs issued to the VEGP applicant. For other RAIs, the WLS applicant endorsed the VEGP applicant's responses. Specifically, the following RAIs issued to the WLS applicant correspond to RAIs issued to the VEGP applicant.

WLS RAI	WLS RAI Response Date	Corresponding VEGP RAI
19-69	[b]	19-69
19-71	3/31/2010	19-71
N/A	[a]	19-81
N/A	[a]	19-82
N/A	[a]	19-83
N/A	[a]	19-84
N/A	[b]	19-85
N/A	[a]	19-86
N/A	[a]	19-88
N/A	[a]	19-92
N/A	[a]	19-94

[a] Endorsed 11/4/2010
[b] Endorsed 4/25/2011

The NRC staff compared the responses provided by the WLS applicant to the responses provided by the VEGP applicant, and concluded that the responses are essentially identical. Therefore, the conclusions reached by the NRC staff regarding Phase 2 SFP cooling are applicable to WLS.

The following portion of this technical evaluation section is reproduced from Attachment A to Appendix 19.A of the VEGP SER:

Phase 3 - Summary of Technical Information for Core Cooling Mitigation Should a LOLA Event Occur

One aspect of the requirements in 10 CFR 50.54(hh)(2) is the need for an applicant to provide strategies and procedures to help assure core cooling should a LOLA event occur. In order to develop guidance and strategies to maintain or restore core cooling, the applicant identified a list of key safety functions that are to be protected should a LOLA event occur. The list identified by the applicant is similar to the generic list provided in NEI 06-12. The methodology used by the applicant to identify strategies to maintain or restore these key safety functions is similar to the method used by currently operating reactor utilities (i.e., NEI 05-07, Revision 1) to identify reactor core cooling mitigation strategies. The applicant identified each key safety function (including necessary support equipment such as power, cooling water, ventilation, etc.). After identifying the key safety functions, the applicant identified the normal, primary and alternative means of meeting the key safety functions. The applicant identified the minimal set of equipment for both the primary and alternate means.

Next, the applicant identified the physical locations of equipment in the systems and support systems needed to satisfy each key safety function. The applicant compared the physical locations of the primary and alternate means to the separation criteria in NEI 06-12. To meet the separation criteria, a strategy must meet the criteria depicted in Figure 4.2, "Separate Building Criteria," or Figure 4.3, "Nearby Building Criteria, External Threats" of NEI 06-12, or must meet the criteria in both Figure 4.4, "Same Face External Threats (all Elevations)" and Figure 4.5, "Internal Threat Separation Criteria," of NEI 06-12. Figures 4.2, 4.3, and 4.4 address damage from the exterior of the structures (e.g., a truck bomb or aircraft impact). Figure 4.5 addresses damage caused by fires and explosions that originate from within the confines of a structure. The applicant stated that the separation of the primary and alternate means meet the criteria of NEI 06-12.

The applicant identified the key safety functions for core cooling. For each key safety function, it identified the normal means of providing the key safety function, and the primary and alternate means to meet each key safety function if the normal means is not available. The key safety functions, normal means, primary means, and alternate means were identified as follows:

Key Safety Function - Reactor Coolant System (RCS) Makeup

The applicant identified RCS makeup as a key safety function. The applicant stated the normal means of providing RCS makeup for the AP1000 design is the CVS (e.g., if a cooldown of the RCS results in shrinkage in RCS volume). The applicant has identified the primary means of meeting RCS makeup if CVS is unavailable is core makeup tank (CMT) A. CMT B is the alternate means of providing RCS makeup. The CMTs are part of the passive core cooling system (PXS).

Key Safety Function – Safety Injection

The applicant identified safety injection as a key safety function. The applicant stated that a LOLA event inside containment is complex due to the open nature of the containment structure, the proximity of equipment needed to ensure safe shutdown, and the array of possibilities for LOLA events. In general for a pressurized-water reactor (PWR), safety injection is only needed if there is a loss of coolant accident or a steam line break. The applicant stated that the normal means of meeting the key safety function, safety injection, for the AP1000 design is through the PXS. If a LOLA event occurs in a location inside containment where a LOCA would not occur, there is no need for safety injection. If a LOLA event were to occur in a location that could cause a LOCA and drain the IRWST, the containment would be flooded, and the AP1000 design should remain in a safe cooldown state. If a LOLA event were to occur near the IRWST or any of its injection piping such that it damaged that equipment, the containment would be flooded and the AP1000 should remain in a safe cooldown state. The applicant identified the following primary and alternate means. If a LOLA event initiates a LOCA without causing the IRWST to fail or the injection piping to rupture, the primary means of providing safety injection involves CMT A, Accumulator A, two "east" steam generator compartment 4th stage automatic depressurization system (ADS) valves, the IRWST, two "south" injection squib valves, two "south" recirculation squib valves, and associated piping. The alternate means uses CMT B, Accumulator B, and two "west" steam generator compartment 4th stage ADS valves, the IRWST, two "north" injection squib valves, two "north" recirculation squib valves, and associated piping. The equipment for the primary means is located in the "southern" half of containment and the equipment for the alternate means is located in the "northern" half of containment with the exception of the 4th stage ADS squib valves. The two 4th stage squib valves credited for the primary means are located in the "east" steam generator compartment, and the two 4th stage ADS valves for the alternate means are in the "west" steam generator compartment. They are separated by two 24-inch, steel-lined, concrete walls. The applicant stated the physical separation meets the criteria in NEI 06-12.

The applicant stated that the only equipment credited that requires power in the event of a LOCA are the squib valves, which receive power and control from the protection and safety monitoring system (PMS) divisions and the Diverse Actuation System (DAS). All other equipment is passive. Either the PMS or the DAS can arm and fire the squib valves. The applicant stated that PMS and DAS are spatially separated except in the area of the squib valves themselves. If the LOLA event were to occur in that area, safety injection would be initiated from the other safety injection means or through gravity drain through the break in the squib valve line. The applicant stated that outside containment the four safety-related division battery rooms are in the "northern" portion of the auxiliary building. Each divisional battery room is separated by a 24-inch thick concrete wall and is routed to its containment penetration through separate three-hour fire areas, each bordered by 24-inch thick concrete walls. The backup DAS panel is spatially separated from divisional battery rooms and the main control rooms by several reinforced concrete walls. The applicant states these two diverse *methods to actuate safety injection and ADS satisfy the Chapter 4 guidance of NEI 06-12 for providing the key safety function, safety injection.*

Key Safety Function – Decay Heat Removal

The applicant provided the following description of the normal, primary and alternative means for decay heat removal. The normal means of providing decay heat removal for the reactor core is through the main feedwater system. The primary means of providing decay heat removal is to use the startup feedwater system, which consists of two redundant trains, each capable of providing adequate flow from the condensate storage tank to remove decay heat from the steam generators. If normal alternating current power (ac-power) and direct current power (dc-power) are lost, the operators can manually start the standby diesel generators (DGs) and provide power to the standby feedwater pumps and their support systems. The alternate means of providing decay heat removal is via natural circulation through the passive residual heat removal (PRHR) heat exchanger, located in the IRWST. The inlet valve from the RCS to the PRHR heat exchanger is normally open and the outlet valves in the line from the heat exchanger to the RCS are normally closed. To initiate decay heat removal through the PRHR heat exchanger, operators would open the valves (normally closed; air operated; open on loss of air pressure or electrical power, or a safety injection signal) located in the PRHR outlet. The applicant states these two methods for decay heat removal satisfy the guidance of Chapter 4 of NEI 06-12 for providing decay heat removal.

Staff Evaluation of Reactor Core Cooling Mitigation

The staff evaluated the applicant's submittal on core cooling against the guidance of NEI 06-12 and DC/COL-ISG-016. NEI 06-12 directs COL applicants to employ strategies and processes similar to those employed at currently operating reactors to address core cooling, SFP cooling, and fission product barrier integrity issues caused by LOLA events. The following areas were evaluated by the staff.

The staff evaluated the applicant's list of key safety functions. The staff reviewed Section 7.0, Phase 3 Mitigative Strategies, in the applicant's May 29, 2009, submittal. Based on the AP1000 design, the staff finds the listed key safety functions to be appropriate, and the staff has not identified any other safety functions that need to be addressed by the applicant to help maintain or restore core cooling. The staff evaluated the process used by the applicant to identify strategies and procedures to mitigate a LOLA event, and determined that the process used by the applicant followed the guidance of DC/COL-ISG-016.

Guidance in NEI 06-12 specifies that an applicant identify target areas (i.e., areas of the plant where core damage or fission product releases could be a direct result of a LOLA event). In RAI 19-10 the staff asked the applicant to identify all target areas and provide a site map with the areas identified. In its response dated October 29, 2009, the applicant identified the Auxiliary Building and Shield Building for VEGP Units 3 and 4 as target areas for those units. The staff finds *the applicant followed the guidance in NEI 06-12, and considers this RAI resolved.*

The staff reviewed the primary and alternate means of meeting the key safety functions (RCS makeup, safety injection, and core decay heat removal) identified in the applicant's May 29, 2009, submittal.

In RAI 19-40 the staff asked the applicant to more fully discuss the key safety function RCS makeup. The staff asked the applicant to use NEI 06-12 and (a) more clearly identify the primary and alternate means to meet the key safety function including key support functions such as ac-power and dc-power, instrumentation and control, instrument air, and component cooling, (b) identify the success criteria used for determining the minimal set of equipment (e.g., from the VEGP Units 3 and 4 PRA) needed to maintain or restore the key safety function, and (c) discuss the separation of the primary and alternative means and their support functions, including which of the separation criteria in NEI 06-12 is met by the alternate means. When considering whether separation is adequate, the evaluation should consider a LOLA event that occurs either inside or outside the structure containing the primary means of meeting the key safety function. In its response dated November 13, 2009, the applicant stated the primary means was the CVS and the alternate means was the CMTs. The response discussed the support systems needed for the primary and alternate systems. The applicant stated the success criteria for RCS makeup are from Table 7.4-1, "Systems Required for Safe Shutdown," of the AP1000 DCD. The applicant stated that the primary and alternate means for providing RCS makeup meet the separation criteria in NEI 06-12. In RAI 19-99, which refers to RAI 19-40, the staff asked the applicant to more clearly address the degree of spatial separation, as defined by NEI 06-12, between the primary and alternate means of providing RCS makeup. In its response dated May 24, 2010, the applicant proposed to revise the wording in the MSD to clarify that the primary and alternate means of providing RCS makeup are CMT A and CMT B, respectively. The staff reviewed the applicant's description of the separation between the primary and alternate means, and agrees that it meets the guidance of NEI 06-12. The staff considers these RAIs resolved, pending the incorporation of this change in a future revision of the MSD. This is ***Confirmatory Item 19.A-33****.*

<u>*Resolution of Standard Content Confirmatory Item 19.A-33*</u>

Confirmatory Item 19.A-33 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-33 is now closed.

In RAI 19-41 the staff asked the applicant to more fully discuss the key safety function, safety injection. The staff asked the applicant to use NEI 06-12 and (a) more clearly identify the primary and alternate means to meet the key safety function including key support functions such as ac-power and dc-power, instrumentation and control, instrument air, and component cooling, (b) identify the success criteria used for determining the minimal set of equipment (e.g., from *the VEGP Units 3 and 4 PRA) needed to maintain or restore the key safety function, and (c) discuss the separation of the primary and alternative means and their support functions, including which of the separation criteria in NEI 06-12 is met by the alternate means. When considering whether separation is adequate, the evaluation should consider a LOLA event that occurs either inside or outside the structure containing the primary means of meeting the key safety function. In its response dated November 13, 2009, the applicant stated that the primary means for providing safety injection is by injecting water from the CMTs, accumulators, and the IRWST into the reactor vessel using the PMS and Class 1E direct current batteries. These components are located inside containment with exception of the PMS and the Class 1E batteries which are located in the non-radiological portion of the auxiliary building. Only fourth stage ADS squib valves require dc-power to actuate. The dc-power is normally supplied by the Class 1E safety-related batteries located in the auxiliary building. The applicant identified the alternate means to provide safety injection as the DAS, which can actuate a limited number of plant controls, including the fourth stage ADS squib valves. The success criteria for safety injection were derived from Table 7.4-1 of the AP1000 DCD. The applicant discussed the separation between the DAS and other systems and components needed for safety injection. The staff found this response unacceptable and asked for more information in RAI 19-100.*

In RAI 19-100, which refers to RAI 19-41, the staff asked the applicant to (a) discuss in more detail and clarity the pathways and equipment credited for the primary and alternative means that provide the key safety function safety injection; (b) discuss the specifics of the spatial separation among the primary and alternative means and their support systems including ac-power; (c) state whether or not the AP1000 is designed so that dc-power to equipment needed to actuate the primary and alternative means of safety injection meets the separation criteria of NEI 06-12 including equipment either inside or outside of containment; (d) state whether or not the CMTs, accumulators, and their piping credited for the primary means are spatially separated from the equipment credited for the alternative means (e.g., CMTs, accumulators, and their piping); and (e) document this information. In its response dated May 24, 2010, the applicant provided details on pathways for safety injection and spatial separation of the pathways and proposed to revise Section 7.18 of the MSD. The applicant also stated that no LOLA event is possible that could simultaneously breach the containment shell and cause a non-isolatable LOCA. The staff reviewed the information provided by the applicant. The staff considers that the spatial separation of the pathways meets the guidance of NEI 06-12, and considers these RAIs resolved, pending the incorporation of this change in a future revision of the MSD. This is ***Confirmatory Item 19.A-34****.*

<u>Resolution of Standard Content Confirmatory Item 19.A-34</u>

Confirmatory Item 19.A-34 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-34 is now closed.

In RAI 19-42 the staff asked the applicant to more fully discuss the key safety function, decay heat removal. The staff asked the applicant to use NEI 06-12 and (a) more clearly identify the primary and alternate means to meet the key safety function including key support functions such as ac-power and dc-power, instrumentation and control, instrument air, and component cooling, (b) identify the success criteria used for determining the minimal set of equipment (e.g., from the VEGP Units 3 and 4 PRA) needed to maintain or restore the key safety function, and (c) discuss the separation of the primary and alternative means and their support functions, including which of the separation criteria in NEI 06-12 is met by the alternate means. When considering whether separation is adequate, the evaluation should consider a LOLA event that occurs either inside or outside the structure containing the primary means of meeting the key safety function. In its response dated November 13, 2009, the applicant stated that the primary means of decay heat removal was from using one of the two redundant startup feedwater system trains using flow from the condensate storage tanks to deliver water to the steam generators. The startup feedwater pumps are located inside the turbine building. The applicant identified figures in the AP1000 DCD that detailed the flow paths. If offsite power is lost, the applicant stated the pumps can be loaded on the standby DGs. The applicant identified the alternate means as the PRHR heat exchanger located in the IRWST. This flow path needs no power to initiate decay heat removal. Assuming loss of ac-power and dc-power, the air-operated valves in the associated flow paths fail open and initiate natural recirculation. The success criteria were identified as coming from Table 7.4-1 of the AP1000 DCD. The applicant stated that the locations of the primary equipment are the turbine building and the auxiliary building, and the location of the alternate system is in the containment. The staff found this response acceptable, because the primary and alternate systems were clearly defined and their separation meets the guidance and criteria in NEI 06-12. The staff considers this RAI resolved.

In RAIs 19-67 and 19-102, the staff sought to have the applicant assure that certain ac-power and dc-power capabilities as well as capabilities for manual start of the standby DGs and certain equipment (e.g., the startup feedwater pumps) existed. The manual start capabilities are provided in the AP1000 design as a defense-in-depth measure similar to that required by Orders to operating reactors following the events of 9/11. In its response to RAI 19-67 dated November 13, 2009, the applicant stated that, based on the spatial separation guidance in NEI 06-12, adequate spatial separation is provided between sources of ac-power and dc-power such that no postulated LOLA event would disable the power for both the primary and alternate equipment credited to support key safety functions in Section 7 of the submittal. The applicant also stated there is adequate spatial separation between the standby DGs, the Class 1E batteries, and the dc-power supply for the secondary DAS panel. The applicant stated that, based on the guidance in NEI 06-12, there is adequate separation among the four divisions of safety-related dc batteries such that no single LOLA incident would disable all four safety-related trains of Class 1E batteries. In its response to RAI 19-102 dated May 24, 2010, the applicant stated the standby DGs included in the AP1000 design have an air start system. The applicant proposed *to modify the submittal to state the following: (1) that the standby DGs can be manually started without dc-power being available; (2) the standby DGs can power the startup feedwater pumps and their support systems; (3) the startup feedwater pumps and their support systems can be manually started without dc-power; and (4) operator instructions will be developed for performing a manual start of the standby diesel generator for responding to a LOLA event at the site. The applicant later revised its response to RAI 19-102 based on publically noticed closed meetings with the staff. The applicant challenged the regulatory basis for requiring the DGs to have a manual start function and the staff determined that there was no regulatory basis for requiring the DGs to have this function. In its revised response to RAI 19-102 dated October 8, 2010, the applicant stated that, because of the fundamental design differences and passive nature of the AP1000 as compared to the currently operating reactor fleet, the AP1000 does not rely upon the DGs for any safety functions. The applicant stated that the standard AP1000 design does not provide the capability to manually start or operate the diesel generators in the event of a loss of all ac and dc power. Instead, the AP1000 design provides independent means of providing decay heat removal which satisfies the separation criteria discussed in Section 4.2.3.4 of NEI 06-12. As stated in the response to RAI 19-67, adequate spatial separation is provided between the sources of ac-power and dc-power such that no postulated event would disable the power for both the primary and alternate equipment credited to support the key safety functions, including decay heat removal. The applicant stated that for Phase 1 strategies, the NEI 06-12 guidance specifies that the applicant should implement NRC's February 25, 2005, guidance, "Developing Mitigating Strategies/Guidance for Nuclear Power Plants to Respond to Loss of Large Areas of the Plant in Accordance with B.5.b of the February 25, 2002, Order." However, the applicant noted that the ability to perform decay heat removal is a Phase 3 strategy and that guidance for addressing Phase 3 strategies is discussed in Section 4.2.3 of NEI 06-12. The applicant stated it would modify its submittal to stipulate that procedures will be developed to start the standby diesel generators upon a loss of normal ac-power, assuming plant dc-power is available to power the necessary diesel generator controls and auxiliaries. These procedures will be completed prior to initial fuel load. The staff considers that this design meets the guidance of NEI 06-12, and considers these RAIs resolved, pending the incorporation of this change in a future revision of the MSD. This is* ***Confirmatory Item 19.A-35****.*

Resolution of Standard Content Confirmatory Item 19.A-35

Confirmatory Item 19.A-35 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-35 is now closed.

In RAIs 19-66 and 19-68 the staff asked the applicant to address how the plant will be brought to a safe shutdown condition and how key safety functions will be met following a LOLA event inside or outside containment. The applicant was to address primary and alternative means, their support systems, minimal *equipment, and the spatial separation between the means as defined in NEI 06-12. In its response dated November 13, 2009, the applicant addressed the primary and alternate pathways of providing core cooling and addressed other key safety functions defined for VEGP Units 3 and 4. It stated redundant equipment inside and outside containment is spatially separated. Equipment inside the containment is separated in reactor coolant loop compartments that are protected by modular walls that are greater than 18-inches thick. In addition, there are two vessel injections lines located 180 degrees apart at the reactor vessel. The applicant stated adequate spatial separation exists between the standby DGs, the Class 1E batteries, and the dc-power supply for the secondary DAS panel such that power is expected to be available to achieve safe shutdown of the unit. The staff finds that the arrangement described by the applicant meets the separation criteria in NEI 06-12. The staff considers these RAIs resolved.*

<u>Summary of Technical Information for Fission Product Release Reduction</u>

Although the AP1000 design is robust and includes many safety features, it is possible that a LOLA event could damage fuel in the SFP or the reactor core. The provisions of 10 CFR 50.54(hh)(2) require an applicant to develop and implement strategies and procedures for fission product release mitigation. The object is to have strategies and procedures that will limit fission product releases to the public should a LOLA event occur. The applicant identified a list of key safety functions that are to be protected should a LOLA event occur. The list identified by the applicant is similar to the generic list provided in NEI 06-12. The methodology used by the applicant is similar to the method used by currently operating reactor utilities (i.e., NEI 05-07, Revision 1) to identify fission product release mitigation strategies. After identifying the key safety functions and the normal, primary, and alternate means, the applicant identified the minimal set of equipment for both the primary and alternate means of satisfying each key safety function (including necessary support equipment such as power, cooling water, ventilation, etc.).

Next, the applicant identified the physical locations of equipment in the identified systems and support systems. The applicant compared the physical locations of the primary and alternate means to the separation criteria in NEI 06-12. The applicant stated that the separation of the primary and alternate means meet the criteria of NEI 06-12.

In its submittal, the applicant identified the following key safety functions for reducing fission product releases should a LOLA event occur:

Key Safety Function - Containment Cooling

The applicant identified containment cooling as a key safety function for fission product release mitigation. It identified that the PCS is involved in the normal, primary, and alternate means to provide containment cooling should a LOLA event occur. When providing containment cooling in its normal mode, the PCS delivers cooling water (stored in the PCCWST) to the exterior of the containment via gravity flow. This condenses steam in the containment to avoid exceeding *containment design limits and to compensate for RCS boil off following an accident. The PCS provides the heat transfer mechanism to provide for condensation, while the PXS serves to return condensate to the IRWST or the containment sump. The applicant stated that if the PCCWST is unavailable, design bases code calculations indicate that air-only cooling is sufficient to initially maintain containment cooling. The applicant identified the following primary and alternate means. The primary means of providing containment cooling involves pumping water to the PCS bucket at the top of the containment through one of two pipes 180 degrees apart on the exterior of the shield building. The alternate means involves pumping water to the PCS bucket using the other pipe on the shield building exterior. The following pumps are available to pump through the primary or alternate paths: the PCS makeup pumps, the motor-driven FPS pump, the diesel-driven FPS pump, or the portable pump via a connection to the PCS on the exterior of the auxiliary building. Sources of water include the fire water storage tanks (greater than 325,000 gallons each), the passive containment cooling ancillary water storage tank (780,000 gallons), and the cooling tower basin (greater than 5,000,000 gallons). The normal PCS gravity drain does not require ac-power or dc-power to activate the in-line valves, which fail open on loss of power. Neither the diesel-driven FPS pumps nor the portable pump require ac-power or dc-power to operate.*

Key Safety Function - Containment Isolation

The applicant stated that if a LOLA event were to occur that caused a LOCA, it would be necessary to isolate containment prior to actuating PXS, if PXS is to be fully effective. In the AP1000 design, a high percentage of containment penetrations are normally closed, and the few that are normally open either fail closed or use dc-power valves for isolation. The one exception is a fluid line that penetrates containment, where both isolation valves require dc-power to close, and there is no check valve in the line to prevent reverse flow. As designed for the AP1000 plant, the applicant states the containment isolation valves and their power supplies in this line meet the spatial separation criteria defined in NEI 06-12.

The applicant stated that each containment penetration has a redundant means of containment isolation. Where the redundant means of isolation is also an electric-operated valve, the redundant valve is assigned to a different electrical and instrumentation and control (I&C) division. Additionally, outside of containment the four electrical and I&C divisions are maintained in separate fire areas divided by three hour fire barriers consisting of 24-inch thick concrete walls. The applicant indicated the two diverse methods of providing containment isolation satisfy the guidance in Chapter 4 of NEI 06-12, and no alternative means are required.

In addition, the applicant discussed the use of sprays (sprays inside buildings and using sprays such as portable pumps with hoses and monitor nozzles to spray the exterior of buildings) to reduce fission product releases.

Staff Evaluation of Fission Product Release Reduction

The staff evaluated the applicant's strategies and procedures for the key safety function fission product release reduction using the criteria in NEI 06-12, as modified by DC/COL-ISG-016.

The staff reviewed Section 7.0, Phase 3 Mitigative Strategies, in the applicant's May 29, 2009, submittal. Based on the AP1000 design, the staff finds the listed key safety functions to be appropriate, and the staff has not identified any other safety functions that need to be addressed by the applicant to help maintain the capability to reduce fission product releases.

In RAI 19-43 the staff asked the applicant to identify the primary and alternate means to meet the key safety function, containment cooling, including support functions. The staff asked that the applicant identify the source of the success criteria used, and asked for a discussion of the separation of the primary and alternate means using the separation criteria of NEI 06-12. In its response dated November 13, 2009, the applicant stated that the primary means of providing containment cooling is via gravity drain from the PCCWST to the outside surface of the containment. The source of the success criteria was identified as Table 7.4-1, "Systems Required for Safe Shutdown," of the AP1000 DCD. The response noted that the PCS makeup pumps can pump water to the distribution bucket of the PCCWST, which is hung from the shield building roof and suspended just above the containment dome for optimum water delivery. The applicant identified alternative paths of providing water to the distribution bucket as two lines routed on opposite sides of the shield building. The lines can be fed by either of the fire protection pumps (one ac-driven and the other diesel-driven) or the portable pump. The staff considers the design to meet the guidance of DC/COL-ISG-016 and NEI 06-12. The staff considers this RAI resolved.

The staff reviewed the means of meeting the key safety function, containment isolation. In RAI 19-44, the staff asked the applicant to specifically identify the primary and alternate means of meeting the key safety function, identify from what source the applicant developed its success criteria, and discuss the spatial separation of equipment credited for the primary and alternative means of providing the key safety function, containment isolation. The staff reviewed the applicant's responses dated November 13, 2009, and February 5, 2010. The applicant identified the inboard and outboard containment isolation valves as the primary and alternate means, indicated that the success criteria were taken from Table 6.2.3-1, "Containment Mechanical Penetrations and Isolation Valves," of the AP1000 DCD, and discussed the physical separation of the inboard and outboard valves. The staff found these explanations acceptable, but in RAI 19-101 requested that the applicant reference where it is documented that the two I&C divisions controlling and powering the containment isolation valves are separated by at least two full height 24-inch thick concrete walls. In its response dated May 24, 2010, the applicant proposed to modify the submittal accordingly. The staff considers this RAI resolved, pending the incorporation of this change in a future revision of the MSD. This is ***Confirmatory Item 19.A-36****.*

<u>*Resolution of Standard Content Confirmatory Item 19.A-36*</u>

Confirmatory Item 19.A-36 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-36 is now closed.

The guidance in NEI 06-12 specifies that an applicant discuss the evaluation of existing dose projection models for their adequacy in projecting doses for a LOLA event. The staff requested the applicant to address this in the staff's RAI 19-62. The staff's RAI sought to get the applicant to choose from among an array of dose projection models one that would be appropriate for onsite responders following a LOLA event where there were fission product releases. In its response dated November 13, 2009, the applicant stated that due to the unbounded nature of a LOLA event, the severe accident dose rates in Chapter 49 of the AP1000 PRA were determined to be the most appropriate dose models for projecting doses to onsite event responders. The staff considers PRA dose projections acceptable for this purpose and considers this RAI resolved.

Staff Evaluation of Command and Control

While having equipment, guidance, and strategies for LOLA mitigation is important, loss of the control room and its personnel, or loss of command and control could negate the best of strategies, if such a loss is not factored into planning. Although the nuclear industry has well-developed command and control structures for design basis events, the extent and type of damage postulated for some beyond design basis security threats may create unique challenges. Normal command and control structures may be interrupted. The guidance in NEI 06-12 is designed to enhance command and control by pre-thinking strategies for dealing with command and control interruption. This includes guidelines for initial site operational response should a LOLA event occur. NEI 06-12 provides appropriate guidance on helping to assure command and control issues are capable of being performed following a LOLA event. Additional requirements are given by 10 CFR 50.54(hh)(1) and its attendant guidance, Regulatory Guide 1.214, "Response Strategies for Potential Aircraft Threats," July 2009, which describes approaches for conforming with the requirements that power reactor licensees develop, implement, and maintain procedures for responding to potential aircraft threats.

The applicant stated that command and control protocols will be in place that specify that the site maintains overall command authority for onsite firefighting actions so that firefighting priorities, as defined by operations personnel (i.e., a licensed operator), are communicated to the offsite incident commander. This is an important consideration, because operations should make the decision whether the first priority of fire fighters is the recovery of key safety functions.

The applicant has committed to develop site assembly areas for recovery operations personnel, identify helicopter landing zones, and develop enhanced communications capabilities to focus fire/response teams on necessary plant *operations. The applicant has committed to develop procedures to provide guidance to plant personnel on how to maintain core cooling in the wake of a LOLA event. This will include guidance for early responders (e.g., through Extensive Damage Mitigation Guidelines (EDMGs)) and high level guidance to the incident commander and ERO that will take over once additional operations support arrives on site or is mustered. In its response dated December 23, 2009, to RAIs 19-75 and 19-77, the applicant stated that it would modify the submittal to add a commitment that the EDMG will be written prior to initial fuel load to cover an event where the control room staff and resources are substantially affected. The command and control structure will be established using EDMG guidance and should state that the most senior operations person that survives the event becomes the onsite incident commander until relieved. As discussed in its response to RAI 19-76, dated October 29, 2009, the applicant committed that, prior to initial fuel load, the EDMGs will cover offsite notifications, emergency response organization callout, and damage assessment. The staff finds these responses follow the guidance of NEI 06-12 and considers these RAIs resolved, pending the incorporation of these changes in a future revision of the MSD. This is* ***Confirmatory Item 19.A-37****.*

Resolution of Standard Content Confirmatory Item 19.A-37

Confirmatory Item 19.A-37 is an applicant commitment to revise its MSD under Part 9 to its COL application to incorporate the described changes. The staff verified that the MSD under Part 9 of the VEGP COL application was appropriately revised. As a result, Confirmatory Item 19.A-37 is now closed.

Guidance in NEI 06-12 discusses dispersal of plant personnel to help assure that command and control capabilities will survive a LOLA event, if the plant is warned prior to the event. In the submittal and in response to various RAIs, the applicant stated that procedures will identify staging areas for operations and support personnel that maximize survivability from a LOLA event, if there is prior warning. The procedures provide guidance and direction for relocation of personnel based on various threats. Guidance documents will exist to obtain corporate resources, and helicopter landing zones are designated in procedures. The applicant stated that radios needed to support operational recovery teams are provided in appropriate locations and are separate from firefighting radios. SAMG-like guidance is available to operators and the ERO to implement the mitigative strategies. Thus, the staff finds this acceptable because it meets the guidance in NEI 06-12.

Evaluation of WLS RAIs

In addition to the site-specific RAIs, the NRC staff issued RAIs to the WLS applicant that mirrored the RAIs issued to the VEGP applicant. For other RAIs, the WLS applicant endorsed the VEGP applicant's responses. Specifically, the following RAIs issued to the WLS applicant correspond to RAIs issued to the VEGP applicant.

WLS RAI	WLS RAI Response Date	Corresponding VEGP RAI
19-16	3/31/2010	19-10
19-44	3/31/2010	19-40
19-45	3/31/2010	19-41
19-46	3/31/2010	19-42
19-47	3/31/2010	19-43
19-48	3/31/2010	19-44
19-62	3/31/2010	19-62
19-66	3/31/2010	19-66
19-67	3/31/2010	19-67
19-68	3/31/2010	19-68
19-74	3/31/2010	19-75
19-75	3/31/2010	19-76
19-76	3/31/2010	19-77
N/A	[a]	19-99
N/A	[a]	19-100
N/A	[a]	19-101
N/A	[b]	19-102

[a] Endorsed 11/4/2010
[b] Endorsed 4/25/2011

The NRC staff compared the responses provided by the WLS applicant to the responses provided by the VEGP applicant, and concluded that the responses are essentially identical. Therefore, the conclusions reached by the NRC staff regarding Phase 3 reactor core cooling mitigation, fission product release reduction, and command and control should a LOLA event occur are applicable to WLS.

Conclusion

Based on the staff's review of the information provided by the applicant under 10 CFR 52.80(d), the staff concludes that the applicant has adequately followed the guidance of DC/COL-ISG-016; NEI 06-12; and the February 25, 2005, guidance letter. The staff finds that the applicant provided sufficient information at the COL application stage, including commitments made in the MSD, to meet the requirements of 10 CFR 52.80(d) and to provide reasonable assurance that the requirements in 10 CFR 50.54(hh)(2) will be met prior to the initial fuel load of WLS Units 1 and 2, respectively.

Table 19-A-1. WLS RAI Responses that Correspond to VEGP RAI Responses

WLS RAI	WLS RAI Response Date	VEGP RAI
19-16[2]	3/31/2010	19-10
19-17[1]	3/31/2010	19-11
19-18[2]	3/31/2010	19-12
19-19[1]	3/31/2010	19-13
19-20[2]	3/31/2010	19-14
19-21[2]	3/31/2010	19-15
19-22[1]	3/31/2010	19-16
19-23[1]	3/31/2010	19-17
Endorsed	11/4/2010	19-18
19-24[1]	3/31/2010	19-19
19-25[1]	3/31/2010	19-20
19-26[1]	3/31/2010	19-21
19-27[1]	3/31/2010	19-22
19-28[1]	3/31/2010	19-23
19-29[1]	3/31/2010	19-24
19-30[1]	3/31/2010	19-25
19-31[1]	3/31/2010	19-26
19-32[1]	3/31/2010	19-27
19-33[1]	3/31/2010	19-28
19-34[2]	3/31/2010	19-29
19-35[1]	3/31/2010	19-30
19-36[1]	3/31/2010	19-31
19-37[1]	3/31/2010	19-32
19-38[1]	3/31/2010	19-33
19-39[1]	3/31/2010	19-34
19-40[1]	3/31/2010	19-35
19-41[1]	3/31/2010	19-37
19-42[1]	3/31/2010	19-38
19-43[1]	3/31/2010	19-39
19-44[1]	3/31/2010	19-40
19-45[1]	3/31/2010	19-41
19-46[1]	3/31/2010	19-42
19-47[1]	3/31/2010	19-43
19-48[1]	3/31/2010	19-44
19-49[2]	3/31/2010	19-45
N/A	N/A	19-46
19-50[2]	3/31/2010	19-47
19-51[2]	3/31/2010	19-48
19-52[2]	3/31/2010	19-49
19-53[2]	3/31/2010	19-50
N/A	N/A	19-51
19-54[2]	3/31/2010	19-52
19-55[2]	3/31/2010	19-53
19-56[2]	3/31/2010	19-54

Table 19-A-1. WLS RAI Responses that Correspond to VEGP RAI Responses

WLS RAI	WLS RAI Response Date	VEGP RAI
19-57[2]	3/31/2010	19-55
N/A	N/A	19-56
19-58[2]	3/31/2010	19-57
19-59[2]	3/31/2010	19-58
N/A	N/A	19-59
19-60[2]	3/31/2010	19-60
19-61[2]	3/31/2010	19-61
19-62[1]	3/31/2010	19-62
19-63[2]	3/31/2010	19-63
19-64[2]	3/31/2010	19-64
19-65[1]	3/31/2010	19-65
19-66[1]	3/31/2010	19-66
19-67[1]	3/31/2010	19-67
19-68[1]	3/31/2010	19-68
Endorsed	4/25/2011	19-69
19-70[2]	3/31/2010	19-70
19-71[2]	3/31/2010	19-71
19-72[1]	3/31/2010	19-72
19-73[1]	3/31/2010	19-74
19-74[2]	3/31/2010	19-75
19-75[2]	3/31/2010	19-76
19-76[2]	3/31/2010	19-77
19-77[1]	3/31/2010	19-78
19-78[2]	3/31/2010	19-79
19-79[2]	3/31/2010	-
[1] The WLS applicant identified its RAI response as consistent with VEGP RAI response.		
[2] The WLS applicant identified its RAI response as plant-specific.		

Table 19-A-2. VEGP RAIs that are Endorsed by WLS

VEGP RAI	VEGP Response Date	WLS Endorsement Date
19-81	05/28/2010	11/4/2010
19-82	05/05/2010	11/4/2010
19-83	05/05/2010	11/4/2010
19-84	05/05/2010	11/4/2010
19-85	08/13/2010	4/25/2011
19-86	05/05/2010	11/4/2010
19-87	05/05/2010	11/4/2010
19-88	05/05/2010	11/4/2010
19-89	05/05/2010	12/12/2011
19-90	05/05/2010	11/4/2010
19-91	05/28/2010	12/12/2011
19-92	05/05/2010	11/4/2010
19-93	05/28/2010	11/4/2010
19-94	05/05/2010	11/4/2010
19-95	05/24/2010	11/4/2010
19-96	05/24/2010	11/4/2010
19-98	05/24/2010	11/4/2010
19-99	05/24/2010	11/4/2010
19-100	06/04/2010	11/4/2010
19-101	05/24/2010	11/4/2010
19-102	05/24/2010	4/25/2011
19-103	05/24/2010	11/4/2010
Erratum 1	05/24/2010	11/4/2010
Erratum 2	05/24/2010	NA
Erratum 3	05/24/2010	11/4/2010
Erratum 4	05/24/2010	NA

20.0 REQUIREMENTS RESULTING FROM FUKUSHIMA NEAR-TERM TASK FORCE RECOMMENDATIONS

This chapter addresses the Fukushima Near-Term Task Force (NTTF) recommendations that are applicable to the William States Lee III Nuclear Station (WLS) Units 1 and 2 Combined License (COL). The applicable recommendations address four topics: reevaluations of the seismic and flood hazard (related to Recommendation 2.1), mitigation strategies for beyond-design-basis external events (related to Recommendation 4.2), spent fuel pool (SFP) instrumentation (related to Recommendation 7.1), and emergency preparedness staffing and communications (related to Recommendation 9.3).

Background

In response to the events at Fukushima resulting from the March 11, 2011, Great Tohoku earthquake and tsunami in Japan, the U.S. Nuclear Regulatory Commission (NRC) established the NTTF to conduct a systematic and methodical review of NRC processes and regulations to determine whether the agency should make additional improvements to its regulatory system and to make recommendations to the Commission for policy direction. In July 2011, the NTTF issued a 90-day report, SECY-11-0093, “Near Term Report and Recommendations for Agency Actions Following the Events in Japan,” identifying 12 recommendations. On September 9, 2011, in SECY-11-0124, “Recommended Actions to Be Taken Without Delay From NTTF Report,” the staff provided to the Commission for its consideration NTTF recommendations that can and, in the staff’s judgment, should be initiated, in part or in whole, without delay. In SECY-11-0124 the staff identified and concluded that the following subset of actions had the greatest potential for safety improvement in the near-term:

1. Recommendation 2.1: Seismic and Flood Hazard Reevaluations

2. Recommendation 2.3: Seismic and Flood Walkdowns

3. Recommendation 4.1: Station Blackout Regulatory Actions

4. Recommendation 4.2: Equipment covered under Title 10 of the *Code of Federal Regulations* (10 CFR) 50.54(hh)(2)

5. Recommendation 5.1: Reliable Hardened Vents for Mark I Containments

6. Recommendation 8: Strengthening and Integration of Emergency Operating Procedures, Severe Accidents Management Guidelines, and Extensive Damage Mitigation Guidelines

7. Recommendation 9.3: Emergency Preparedness Regulatory Actions (staffing and communications).

On October 3, 2011, in SECY-11-0137, “Prioritization of Recommended Actions to Be Taken in Response to Fukushima Lessons Learned,” the staff identified two actions in addition to the actions discussed in SECY-11-0124 which had the greatest potential for safety improvement in the near-term. The additional actions are:

1. Inclusion of Mark II containments in the staff's recommendation for reliable hardened vents associated with NTTF Recommendation 5.1

2. The implementation of SFP instrumentation proposed in Recommendation 7.1

The staff also prioritized the NTTF recommendations into Tier 1, Tier 2, and Tier 3, where the recommendations in Tier 1 represent those that the staff determined should be started without unnecessary delay, while recommendations in Tier 2 are those that could not be initiated in the near term, and recommendations in Tier 3 require further study to support regulatory action.

On February 17, 2012, in SECY-12-0025, "Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami," the staff provided the Commission with proposed orders and requests for information to be issued to all power reactor licensees and holders of construction permits.

On March 9, 2012, the Commission then approved issuance of the proposed orders with some modifications in the staff requirements memorandum (SRM) to SECY-12-0025. As set forth in the Orders in SRM-SECY-12-0025, additional requirements are needed to provide adequate protection to public health and safety or to significantly enhance the protection of public health and safety. In accordance with its statutory authority under Section 161 of the Atomic Energy Act of 1954, as amended (the Act), the Commission may impose these requirements.

On March 12, 2012, the NRC issued Orders EA-12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events" and EA-12-051, "Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation" to the appropriate licensees and permit holders.

The staff also issued the request for information pursuant to 50.54(f) regarding Recommendations 2.1, 2.3 and 9.3, as described in SECY-12-0025, to the appropriate licensees and permit holders in letters dated March 12, 2012.

The following Tier 1 recommendations in SECY-11-0137 as addressed in SECY-12-0025 were considered in determining those that are applicable to the WLS COL review:

1. Recommendation 2.1: Seismic and Flood Hazard Reevaluations
2. Recommendation 2.3: Seismic and Flood Walkdowns
3. Recommendation 4.1: Station Blackout Regulatory Actions
4. Recommendation 4.2: Equipment covered under 10 CFR 50.54(hh)(2)
5. Recommendation 5.1: Reliable Hardened Vents for Mark I and Mark II Containments
6. Recommendation 7.1: Spent Fuel Pool Instrumentation
7. Recommendation 8: Strengthening and Integration of Emergency Operating Procedures, Severe Accidents Management Guidelines, and Extensive Damage Mitigation Guidelines

8. Recommendation 9.3: Emergency Preparedness Regulatory Actions (staffing and communications)

Staff determined that the following three recommendations were applicable and should be addressed by the WLS COL applicant:

1. Recommendation 4.2: Equipment covered under 10 CFR 50.54(hh)(2) - Order licensees to provide reasonable protection for equipment currently provided pursuant to 10 CFR 50.54(hh)(2) from the effects of design-basis external events and to add equipment as needed to address multiunit events while other requirements are being revised and implemented.

2. Recommendation 7.1: Spent fuel pool instrumentation - Order licensees to provide reliable spent fuel pool level instrumentation.

3. Recommendation 9.3: Emergency preparedness regulatory actions (staffing and communications) - Order licensees to do the following until rulemaking is complete:

   - Determine and implement the required staff to fill all necessary positions for response to a multi-unit event.

   - Provide a means to power communications equipment needed to communicate onsite (e.g., radios for response teams and between facilities) and offsite (e.g., cellular telephones and satellite telephones) during a prolonged station blackout.

The staff determined that the remaining Tier 1 recommendations did not need to be further considered in the WLS COL review.

The applicant evaluated the seismic and flood hazards using the current guidance and methodologies. For the seismic hazard, consistent with guidance in Regulatory Guide 1.208, “A Performance-Based Approach to Define the Site- Specific Earthquake Ground Motion,” regarding the need to consider the latest information in the evaluation of seismic hazard, this included consideration of the NUREG–2115, “Central and Eastern United States Seismic Source Characterization for Nuclear Facilities,” (CEUS-SSC) model as described in FSER Chapter 2, Section 2.5.2. Further, FSER Section 3.7 addresses the staff’s evaluation of the seismic analyses of NI structures and adjacent seismic Category II structures based on the WLS Units 1 and 2 seismic hazard. For flood hazard, as evaluated in FSER Chapter 2, Sections 2.4.5 and 2.4.6, the applicant used Regulatory Guide 1.59, “Design Basis Floods for Nuclear Power Plants,” as supplemented by best current practices, as it relates to providing assurance that natural flooding phenomena that could potentially affect the site have been appropriately identified and characterized. Thus, the staff determined that the applicant has already addressed the seismic and flood hazard reevaluation portion of Recommendation 2.1. Therefore, there are no additional requirements left to be addressed in Recommendation 2.1 for seismic and flood hazard reevaluations applicable to the WLS Units 1 and 2 Site COL application.

Additionally, the staff determined that Recommendation 2.3 was not applicable to the WLS COL because the plant is not yet constructed, and Recommendation 5.1 was not applicable because it applied to boiling water reactor type plant designs with Mark I and Mark II Containments. Recommendations 4.1 and 8 did not need to be further considered because SECY-11-0137 and its associated SRM direct that regulatory action associated with them be initiated through rulemaking.

In SECY-12-0025, the staff stated that it would request all COL applicants to provide the information required by the orders and request for information letters through the review process. Accordingly, for the WLS COL application, the staff issued request for additional information (RAI) Letter No. 105, dated April 25, 2012, related to Implementation of Fukushima Near-Term Task Force Recommendations pertaining to seismic hazard reevaluation, mitigation strategies for beyond-design-basis external events, spent fuel pool instrumentation, and emergency preparedness based on Recommendations 2.1, 4.2, 7.1, and 9.3, as modified by SRM-SECY-12-0025. The following sections of this chapter present the staff's safety evaluation related to these areas.

20.1 Mitigation Strategies for Beyond-Design-Basis External Events (Based on Recommendation 4.2)

20.1.1 Introduction

NRC Commission Paper SECY-12-0025 states that the NRC staff will request all COL applicants to provide the information required by the orders and request for information letters described in SECY-12-0025, as applicable, through the review process. For mitigation strategies for beyond-design-basis external events, SECY-12-0025 outlined a three-phase approach for mitigating beyond-design-basis external events. The initial phase involves the use of installed equipment and resources to maintain or restore core cooling, containment, and SFP cooling without alternating current (ac) power. The transition phase involves providing sufficient, portable, onsite equipment and consumables to maintain or restore these functions until they can be accomplished with resources brought from offsite. The final phase involves obtaining sufficient offsite resources to sustain those functions indefinitely.

SECY-12-0025 notes that the AP1000 standard design (which is referenced for WLS) includes passive design features that provide core cooling, containment, and SFP cooling capabilities for 72 hours, without reliance on ac power. The AP1000 design also includes equipment to maintain required safety functions in the long term (beyond 72 hours to 7 days). As such, provisions related to the final phase must be addressed.

NRC Interim Staff Guidance (ISG) JLD-ISG-2012-01, Revision 0, "Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," endorses with clarifications, the methodologies described in the industry guidance document, Nuclear Energy Institute (NEI) 12-06, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," Revision 0. JLD-ISG-2012-01 describes an acceptable approach for developing mitigation strategies for beyond-design-basis external events at nuclear power plants based on the guidance in NEI 12-06.

20.1.2 Summary of Application

The WLS Units 1 and 2 Final Safety Analysis Report (FSAR) provides information on systems used to establish and sustain core cooling, containment, and SFP cooling capabilities for the WLS Units 1 and 2. For example, Section 6.3, “Passive Core Cooling System,” of the FSAR discusses the passive core cooling system (PXS), which provides emergency core cooling following postulated design-basis events, and incorporates by reference Section 6.3 of the AP1000 DCD Tier 2 with identified departures and supplements. FSAR Section 6.2, “Containment Systems,” and Section 9.1, “Fuel Storage and Handling,” address containment systems and fuel storage and handling systems, respectively, and incorporate by reference Section 6.2.2, “Passive Containment Cooling System,” and Section 9.1.3, “Spent Fuel Pool Cooling System,” of the AP1000 DCD Tier 2.

In SECY-12-0025, the NRC staff indicated its intent to review information provided by COL applicants to describe their mitigation strategies for beyond-design-basis external events. In light of SECY-12-0025, the staff issued RAI Letter No. 105, dated April 25, 2012, to request information regarding the WLS Units 1 and 2 mitigation strategies to sustain core cooling, containment, and SFP cooling capabilities functions indefinitely.

The applicant provided an initial response to the RAI in a letter dated June 11, 2012. In its initial response, the WLS Units 1 and 2 COL applicant proposed a license condition related to mitigation strategies for beyond-design-basis conditions resulting from an extended loss of ac power and loss of access to the normal heat sink (referred to below as an ELAP event). Subsequent to that response, the applicant provided the NRC staff with the general mitigation strategy that will be used by WLS, including the strategies for initial (0 to 72 hours) mitigation, in a letter dated May 7, 2015. The letter, which was a supplemental response to RAI Letter No. 105, provided the staff with a Westinghouse report (designated as APP-GW-GLR-171, “AP1000 Flex Integrated Plan,” for the publicly available version) that included a description of the mitigating strategies for beyond-design-basis external events that will be applied at WLS Units 1 and 2.

In Item 12, “Fukushima Response Actions,” of Part 10, “Proposed License Conditions (including ITAAC),” of the WLS COL application, the applicant proposed a license condition related to this subject.

20.1.3 Regulatory Basis

The requirements and guidance for mitigation strategies for beyond-design-basis external events are established or described in the following:

- Atomic Energy Act of 1954, as amended, § 161, authorizes the Commission to regulate the utilization of special nuclear material in a manner that is protective of public health and in accord with the common defense and security.

- 10 CFR 52.97(a)(1), which authorizes the Commission to issue a COL if it finds, among other things, that issuance of the license will not be inimical to the health and safety of the public. This regulation applies here because the Commission found in Order EA-12-049 that it is necessary for power reactor licensees to develop, implement and maintain guidance and strategies to restore or maintain core cooling, containment, and SFP cooling capabilities in the event of a beyond-design-basis external event in order to ensure adequate protection of the public health and safety.

- SRM-SECY-12-0025, "Staff Requirements – SECY-12-0025 – Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami," dated March 9, 2012, approves issuance of orders for beyond-design-basis external events, as necessary for ensuring continued adequate protection under the 10 CFR 50.109(a)(4)(ii) exception to the Backfit Rule.

- JLD-ISG-2012-01, Revision 0, "Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," issued August 29, 2012, endorses NEI 12-06, Revision 0, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide" (issued August 21, 2012), with exceptions/clarifications.

- Order EA-12-049, "Issuance of Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," dated March 12, 2012. Although Order EA-12-049 does not apply to WLS Units 1 and 2, the staff followed the current NRC and industry guidance for establishing mitigation strategies for beyond-design-basis external events at AP1000 reactors in evaluating the equipment used as part of the mitigation strategy for WLS Units 1 and 2.

20.1.4 Technical Evaluation

The NRC staff reviewed the information submitted by the WLS COL applicant regarding its proposed mitigation strategies for beyond-design-basis conditions resulting from an ELAP event. To assess whether the proposed mitigation strategies provided an acceptable approach, the staff applied JLD-ISG-2012-01, Revision 0, which endorses, with clarifications, the methodologies described in industry guidance document NEI 12-06, Revision 0. Appendix F, "Guidance for AP1000 Design," to NEI 12-06 outlines the process to be used by AP1000 COL licensees and applicants to define and implement the mitigation strategies for beyond-design-basis conditions resulting from an ELAP event.

In Section 7.0, "Guidance for AP1000 Design," of JLD-ISG-2012-01, the NRC staff states that the guidance in Appendix F of NEI 12-06 provides an acceptable means to meet the requirements of Order EA-12-049 or license conditions imposing similar requirements for the AP1000 reactor design. Appendix F to NEI 12-06 specifies that the underlying strategies for coping with ELAP events for AP1000 plants involve a three-phase approach as follows:

1. Initial coping through installed plant equipment without ac power or makeup to the ultimate heat sink. From 0 to 72 hours, the certified AP1000 design includes passive systems that provide core cooling, containment, and SFP cooling.

2. Following the 72-hour passive system coping time, support is necessary to continue passive system cooling. From 3 to 7 days, this support can be provided by installed plant ancillary equipment or by offsite equipment installed to connections provided in the AP1000 design.

3. To extend the passive system cooling time beyond 7 days to an indefinite time, offsite assistance is necessary, such as the delivery of diesel fuel oil. Appendix F includes provisions related to the qualification and use of equipment intended to mitigate an ELAP event.

As mentioned in Appendix F to NEI 12-06, APP-GW-GLR-171, referenced above, indicates that core cooling, containment, and SFP cooling is provided for the initial time period of 0 to 72 hours through installed, safety-related plant equipment that is part of the certified design. These systems do not rely on ac power or on access to any external water sources, because the containment vessel and the passive containment cooling system serve as the safety-related ultimate heat sink. The NRC staff reviewed and found acceptable the site-specific functional design, qualification, and inservice testing program descriptions for this safety-related equipment for WLS Units 1 and 2 as discussed in the applicable sections of this report.

Following the initial 72-hour coping period, APP-GW-GLR-171 indicates that support is necessary to continue passive system cooling, and this support can be provided by installed ancillary equipment or by offsite equipment interfacing with installed plant connections. For example, additional inventory for the passive containment cooling system (PCS) and SFP can be supplied from the onsite passive containment cooling ancillary water storage tank (PCCAWST) using the onsite PCS recirculation pumps, powered using the onsite ancillary diesel generators or offsite replacement generators. The installed ancillary equipment and stored cooling water are capable of supporting passive system cooling from 3 days after the event to 7 days after the event. Beyond this time period, the report indicates that offsite assistance and resources are needed. For indefinite coping after 7 days, an offsite pump (PCCAWST makeup pump) and appropriate connection materials to refill the PCCAWST from the closest water source will be provided. In the event that the PCS recirculation pumps are unavailable, a second self-powered, offsite pump (PCS/SFP makeup pump) and appropriate connection materials will be available.

APP-GW-GLR-171 also includes several additional provisions related to the qualification and use of commercially procured equipment that will be used 72 hours after an ELAP event:

- Programmatic controls for this equipment include quality attributes, equipment design, equipment storage, procedure guidance, maintenance, testing, training, staffing, and configuration control.

- The quality assurance (QA) provisions in AP1000 DCD Tier 2, Table 17-1, “Quality Assurance Program Requirements for Systems, Structures, and Components Important to Investment Protection,” will be applied to this AP1000 FLEX equipment.

- The graded approach to availability and testing as shown in AP1000 DCD Tier 2, Section 16.3, “Investment Protection,” will be applied to the FLEX equipment.

- The design and maintenance of the FLEX equipment will be in accordance with Section 11.2, “Equipment Design,” and Section 11.5, “Maintenance and Testing,” respectively, of NEI 12-06.

- AP1000 DCD Tier 2, Section 1.9.5.4, “Additional Licensing Issue – Post-72 Hour Support Actions,” describes procedures that address actions that would be necessary 72 hours subsequent to an ELAP event to maintain core, containment, and SFP cooling for an indefinite period of time.

The NRC staff reviewed the applicable sections of the WLS FSAR, along with their respective AP1000 DCD sections, the FSER for the AP1000 design certification, and other sections of this report to verify the above information. For example, Table 8.1-201, “Site-Specific Guidelines for Electric Power Systems,” in the WLS Units 1 and 2 FSAR indicates that station blackout is addressed as a design issue in the AP1000 DCD. The staff reviewed station blackout as part of its review of Chapter 8 of the AP1000 DCD Tier 2. Section 8.5.2.1, “Station Blackout,” of the AP1000 FSER states that the AP1000 safety-related passive systems automatically establish and maintain safe-shutdown conditions for the plant following design-basis events, including the loss of ac power sources, and the passive systems can maintain these safe-shutdown conditions after design-basis events for 72 hours, without operator action, following a loss of both onsite and offsite ac power sources. The staff reviewed the applicability of this FSER conclusion to WLS Units 1 and 2.

Section 8.3.2, “Direct Current Power and Uninterruptible Power Systems” of the AP1000 FSER, Supplement 2, states that Class 1E batteries will be sized adequately to perform their safety functions as designed and that ITAAC verifying that the batteries are adequately designed are identified in AP1000 DCD Tier 1, Table 2.6.3-3. APP-GW-GLR-171 discusses the connections for the onsite ancillary diesel generators and the offsite portable generators. Electrical isolation between safety related power systems and power sources utilized in Phase 3 is addressed in APP-GW-GLR-171, which states that voltage regulating transformers are the connection point for the offsite portable generators. Section 8.3.2, “Direct Current Systems” of this document discusses how the voltage regulating transformer in combination with fuses and/or breakers will interrupt the input or output (ac) current under faulted conditions to achieve electrical isolation. As part of the license condition, part (c), as set forth in Section 20.1.5 of this SER, the capacity of the offsite portable generators will be assessed by DEC to ensure they are capable of providing power to the necessary loads described in AP1000 DCD Tier 2 Table 8.3.1-4, “Post-72 hours nominal load requirements.” Section 9.5.3 of this document addresses plant lighting systems, specifically emergency lighting which provides illumination in areas where emergency operations are performed.

Emergency core cooling for the WLS Units 1 and 2 is accomplished using the AP1000 PXS, which is described in Section 6.3 of the AP1000 DCD Tier 2. The WLS Units 1 and 2 FSAR specifies that Section 6.3 of the AP1000 DCD Tier 2 was incorporated by reference with identified departures. The staff reviewed WLS Units 1 and 2 FSAR Section 6.3, and found that the departures have no impact on the capability of the PXS to establish and maintain safe-shutdown conditions for 72 hours following a loss of both onsite and offsite ac power sources. Therefore, core cooling for the initial phase (0-72 hours) of mitigation for WLS Units 1 and 2 will be accomplished by its safety-related PXS, per the WLS Units 1 and 2 licensing basis.

The mitigation of a station blackout, as required by 10 CFR 50.63, addresses the capability of a nuclear power plant to provide adequate core cooling during a loss of ac power. In addition to core cooling, the recommendations for mitigation strategies for beyond-design-basis external events also address containment function, and SFP cooling.

The control of containment pressure and temperature for WLS Units 1 and 2 is accomplished using the AP1000 PCS, which is described in Section 6.2.2, "Passive Containment Cooling System," of the AP1000 DCD Tier 2. In its review of the WLS Units 1 and 2 FSAR, the staff found, with the exception of a departure related to the containment leak rate test program, that Section 6.2.2 of the AP1000 DCD Tier 2 was incorporated by reference into the WLS Units 1 and 2 FSAR. In Section 6.2.2 of the AP1000 FSER, the staff stated the principal design basis for the PCS is to maintain the containment internal pressure below the design value for 3 days following a design-basis accident. The staff review, as documented in Section 6.2.1.1, "Containment Pressure and Temperature Response to High-Energy Line Breaks," of the AP1000 FSER, found that the PCS met its design objectives. Therefore, the containment function for the initial phase of (0-72 hours) mitigation for WLS Units 1 and 2 will be accomplished by its safety-related PCS per the WLS Units 1 and 2 licensing basis.

The SFP cooling function for the WLS Units 1 and 2 is accomplished by maintaining sufficient water inventory in the SFP to keep the fuel covered and, therefore, provide the necessary cooling in the event of an extended loss of SFP cooling due to the loss of ac power. In Section 9.1.3.2.3, "Increase in Number of Spent Fuel Storage Locations," in Supplement 2 of the AP1000 FSER, the staff concluded that the SFP will maintain water coverage above the spent fuel assemblies for at least 72 hours following a loss of nonsafety-related SFP cooling, using only safety-related makeup water. Therefore, initial phase mitigation is accomplished through passive means. However, as indicated in Note 9 in the DCD Tier 2 Table 9.1-4, "Station Blackout/Seismic Event Times," for the most limiting scenario (full core offload) the need for operator action 18 hours into the event is specified. In Attachment 1, "Sequence of Events Timeline," to the AP1000 FLEX integrated plan, this action has been identified and the appropriate procedure cited to assure the task is performed. Hence, SFP cooling for the initial phase (0-72 hours) of mitigation for WLS Units 1 and 2 will be accomplished by passive cooling of the SFP in accordance with the WLS Units 1 and 2 licensing basis.

The NRC staff has reviewed the mitigation strategies for beyond-design-basis external events for WLS Units 1 and 2 based on the information provided by the WLS COL applicant, including referenced mitigation guidance for beyond-design-basis external events applicable to AP1000 reactors. The staff finds that the WLS COL applicant has provided or referenced information to describe its mitigation strategies for beyond-design-basis external events in an acceptable manner. The staff recognizes that full implementation of the mitigation strategies for beyond-design-basis external events at AP1000 reactors cannot be established until after licensing (e.g., during procedure development). The staff prepared a license condition for implementation of the mitigation strategies for beyond-design-basis external events at WLS Units 1 and 2, based on the applicant's proposed license condition with specific enhancements to provide consistency with current NRC staff expectations. Completion of the activities associated with the license condition, including lessons learned from initial AP1000 implementation, can be verified through NRC inspection activities.

20.1.5 Post Combined License Activities

For the reasons discussed in the technical evaluation section above, the staff proposes to include the following license condition related to the mitigation strategies program:

- License Condition (20-1) – Mitigation Strategies for Beyond-Design-Basis External Events:

  a. The Licensee shall complete development of an overall integrated plan of strategies to mitigate a beyond-design-basis external event at least 1 year before the completion of the last ITAAC on the schedule required by 10 CFR 52.99(a).

  b. The overall integrated plan required by this condition must include guidance and strategies to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities. The overall integrated plan must include provisions to address all accident mitigation procedures and guidelines (including the guidance and strategies required by this section, emergency operating procedures, abnormal operating procedures, and extensive damage management guidelines).

  c. The guidance and strategies required by this condition must be capable of (i) mitigating a simultaneous loss of all alternating current (ac) power and loss of normal access to the normal heat sink and (ii) providing for adequate capacity to perform the functions upon which the guidance and strategies rely for all units on the WLS Units 1 and 2 site and in all modes at each unit on the site.

  d. Before initial fuel load, the Licensee shall fully implement the guidance and strategies required by this condition, including:

     1. Procedures;

     2. Training;

     3. Acquisition, staging, or installation of equipment and consumables relied upon in the strategies; and

     4. Configuration controls and provisions for maintenance and testing (including testing procedures and frequencies for preventative

maintenance) of the equipment upon which the strategies and guidance required by this condition rely.

e. The training required by condition d.2 must use a Systematic Approach to Training (SAT) to evaluate training for station personnel, and must be based upon plant equipment and procedures upon which the guidance and strategies required by this condition rely.

f. The Licensee shall maintain the guidance and strategies described in the application upon issuance of the license, and the integrated plan of strategies upon its completion as required by condition a. The Licensee may change the strategies and guidelines required by this Condition provided that the Licensee evaluates each such change to ensure that the provisions of conditions b and c continue to be satisfied and the Licensee documents the evaluation in an auditable form.

20.1.6 Conclusion

The NRC staff reviewed the mitigating strategies for WLS Units 1 and 2 to provide assurance of core cooling, containment, and SFP cooling capabilities in the event of a beyond-design-basis external event resulting in an ELAP event. The staff finds that the approach for mitigating beyond-design-basis external events to be used at WLS Units 1 and 2 is consistent with NRC Order EA-12-049 and both general and AP1000-specific NRC guidance (including NEI 12-06, Appendix F, as endorsed by the NRC staff). Therefore, the staff concludes that the mitigating strategies for beyond-design-basis external events described for WLS Units 1 and 2 are acceptable. The staff will impose a license condition as discussed in this SER section to verify the implementation of the mitigation strategies for beyond-design-basis external events at WLS Units 1 and 2 as described in the specified documentation.

20.2 Reliable Spent Fuel Pool Instrumentation (Based on Recommendation 7.1)

20.2.1 Introduction

During the events in Fukushima, responders were without reliable instrumentation to determine the water level in the SFP. This caused concerns that the pool may have boiled dry, resulting in fuel damage, and highlighted the need for reliable SFP instrumentation. The SFP level instrumentation at United States (U.S.) nuclear power plants is typically narrow range and, therefore, only capable of monitoring normal and slightly off-normal conditions. Although the likelihood of a catastrophic event affecting nuclear power plants and the associated SFPs in the U.S. remains very low, beyond-design-basis external events could challenge the ability of existing spent fuel pool instrumentation in providing emergency responders with reliable information on the condition of SFPs. Reliable and available indication is essential to ensure plant personnel can effectively prioritize emergency actions.

SECY-12-0025, Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami" states that the staff will request all combined license (COL) applicants to provide the information required by the orders and request for information letters described in SECY-12-0025, as applicable, through the review process. With regard to Recommendation 7.1 for reliable spent fuel pool instrumentation, SECY-12-0025 notes that the AP1000 standard design includes two permanently fixed safety related level instruments with the capability for a third instrument connection.

JLD-ISG-2012-03, Revision 0, "Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation," endorses with exceptions and clarifications the methodologies described in the industry guidance document, NEI 12-02, Revision 1, "Industry Guidance for Compliance with Nuclear Regulatory Commission (NRC) Order EA-12-051, To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," and provides an acceptable approach for satisfying the applicable requirements.

20.2.2 Summary of Application

The NRC issued RAI Letter No. 105 dated April 25, 2012, concerning spent fuel pool instrumentation. The applicant responded to the staff's RAI in letters dated June 11, 2015. As part of the RAI response, the applicant submitted a Westinghouse report, APP-SFS-M3R-004, "Response to NRC Orders EA-12-051 and EA-12-063 and Background Information for Future Licensees on AP1000 Spent Fuel Instrumentation." The RAI responses also proposed adding supplemental information to the FSAR and proposed a license condition.

Supplemental Information

- WLS Supplement (SUP) 9.1-1

The applicant provided supplemental information WLS SUP 9.1-1 addressing spent fuel pool instrumentation in FSAR Section 9.1.3.7.

License Condition

- Part 10, License Condition 12.B

The applicant proposed a license condition related to personnel training for reliable spent fuel pool level instrumentation to Part 10 of the COL application.

20.2.3 Regulatory Basis and Guidance

The requirements and guidance for reliable spent fuel pool instrumentation are established or described in the following:

- SRM-SECY-12-0025, "Staff Requirements – SECY-12-0025 – Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami," dated March 9, 2012, approves issuance of orders for reliable spent fuel pool instrumentation under an administrative exemption to the Backfit Rule and the issue finality requirements in 10 CFR 52.63 and 10 CFR Part 52, Appendix D, Paragraph VIII.

- Atomic Energy Act of 1954, as amended, (the Act), § 161, authorizes the Commission to regulate the utilization of special nuclear material in a manner that is protective of public health and in accord with the common defense and security.

- JLD-ISG-2012-03, Revision 0, "Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation," issued August 29, 2012, endorses NEI 12-02, Revision 1, "Industry Guidance for Compliance with NRC Order EA-12-051, To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," with exceptions and clarifications.

20.2.4 Technical Evaluation

In light of the SECY-12-0025, the staff issued RAI Letter No. 105 requesting additional information in relation to the lessons learned from Great Tohoku Earthquake and Tsunami. In RAI Letter No. 105, Question 1.5-1, third bullet, the staff requested the applicant to:

- Provide sufficient reliable instrumentation, able to withstand design-basis natural phenomena, to monitor key spent fuel pool parameters (i.e., water level, temperature, and area radiation levels) from the control room (detailed Recommendation 7.1 - Enclosure 6 of SECY-12-0025).

Out of these parameters, the most indicative of SFP conditions is the water level. The radiation monitors are used to confirm the integrity of the stored fuel, but cannot be used to determine how much time remains before the fuel integrity is compromised. The SFP water temperature can be used to monitor SFP water temperature from normal range up to boiling temperature. After the SFP water reaches the boiling point it will remain constant while the pool boils dry, therefore, water temperature cannot be used to determine how much time remains before the fuel integrity is compromised. SFP water level is the most useful parameter to indicate SFP condition. The water stored in the pool provides spent fuel cooling and radiation shielding for the operators on the SFP deck. Therefore, the SFP water level can be used to determine how much time remains before the fuel integrity is compromised.

In Commission Order EA-12-051, the Commission describes the key parameters used to determine that a level instrument is to be considered reliable. NEI 12-02, Appendix A4, "AP1000 Spent Fuel Pool Instrumentation Guidance," provides an AP1000-specific acceptable approach for satisfying the applicable requirements. In order to address the staff's RAI, the applicant submitted a series of letters that discussed how the Lee SFP level instrument is designed to be reliable, following the guidance provided in NEI 12-02, Appendix A4, and the applicant added supplemental information WLS SUP 9.1-1 to Section 9.1.3.7 of the FSAR.

Arrangement:

Commission Order EA-12-051, Attachment 2, Section 1.1 states that the spent fuel pool level instrument channels shall be arranged in a manner that provides reasonable protection of the level indication function against missiles that may result from damage to the structure over the spent fuel pool. This protection may be provided by locating the safety-related instruments to maintain instrument channel separation within the spent fuel pool area, and to utilize inherent shielding from missiles provided by existing recesses and corners in the spent fuel pool structure.

The applicant's response states that the AP1000 design has three safety-related SFP level instrument channels (AP1000 DCD Revision 19, Table 7.5-1 (Sheet 7 of 12)). All three channels and associated instrument tubing lines are located below the fuel handling area operating deck and the cask washdown pit as stated in the supplemental information WLS SUP 9.1-1 added to WLS Units 1 and 2 FSAR Section 9.1.3.7. This location provides level indication function protection from missiles that may result from damage to the structure over the spent fuel pool. In addition, the SFP level instruments associated with protection and safety monitoring system (PMS) Divisions A and C are physically separated from the SFP instrument associated with PMS Division B as stated in the supplemental information added to the WLS FSAR Section 9.1.3.7.

The staff evaluated the instrument description provided in the DCD and the proposed supplemental information added to WLS FSAR Section 9.1.3.7 and determined that the SFP level instrument will be arranged in a manner that provides reasonable protection against missiles, and therefore, the staff concludes that these features are in conformance with Commission Order EA-12-051, and the guidance provided by JLD-ISG-2012-03.

Qualification:

Commission Order EA-12-051, Attachment 2, Section 1.2 states that the level instrument channels shall be reliable at temperature, humidity, and radiation levels consistent with the spent fuel pool water at saturation conditions for an extended period.

The applicant's response states that the three safety-related SFP level instruments are seismically qualified and are located below the fuel handling area operating deck (AP1000 DCD Revision 19, Section 9.1.3.4.3.4 and Table 7.5-1 (Sheet 7 of 12)).[1] The environment in these areas is mild with respect to safety-related equipment qualification and affords access for post-accident actions. Even though they are not directly exposed to SFP boiling, the instruments are qualified to function at the conditions (temperature, humidity, and radiation) that could be seen where these instruments are located. This provides assurance that the SFP level transmitters exposed to these environmental conditions will remain available and functional for an extended period.

The staff reviewed the applicant's response and concludes that since the SFP level transmitters are not located on the pool area, they are not required to be designed to handle the pool area conditions. However, they must be designed to remain operational under the worst expected

[1] The RAI responses for this topic discuss a departure from the AP1000 DCD related to environmental zones for the level instruments. The departure is evaluated in FSER Section 3.11.4.

conditions for the area in which they are located. The AP1000 DCD does state that the instruments are designed to remain functional at the expected local conditions; therefore, the staff concludes that these features are in conformance with Commission Order EA-12-051, and the guidance provided by JLD-ISG-2012-03.

Power Sources:

Commission Order EA-12-051, Attachment 2, Section 1.3 states that the instrumentation channels shall provide for power connections from sources independent of the plant alternating current (ac) and direct current (dc) power distribution systems, such as portable generators or replaceable batteries. Power supply designs should provide for quick and accessible connection of sources independent of the plant ac and dc power distribution systems. Onsite generators used as an alternate power source and replaceable batteries used for instrument channel power shall have sufficient capacity to maintain the level indication function until offsite resource availability is reasonably assured.

The applicant’s response states that the AP1000 SFP level instruments are provided with Class 1E DC power supply for at least 72 hours of post-accident monitoring. One of these safety-related instruments is powered through PMS Division A which contains a 24-hour battery supply. The safety-related SFP level instrument PMS divisions are described in the supplemental information (WLS SUP 9.1-1) added to the WLS FSAR Section 9.1.3.7. A description of the AP1000 Class 1E DC and UPS system is contained in AP1000 DCD Revision 19, Section 8.3.2.1.1. Beyond the initial 72 hours, instrument power can be supplied by the use of onsite permanently installed ancillary diesel generators or offsite portable generators with quick and accessible connection points. Permanently installed onsite ancillary diesel generators are capable of providing power for Class 1E post-accident monitoring including SFP level instrumentation. This capability is described in Westinghouse AP1000 DCD Revision 19, Section 8.3.1.1.1. As described in Westinghouse AP1000 DCD Revision 19, Section 1.9.5.4, offsite portable generators are capable of being connected to distribution panels or to a safety-related connection.

As discussed in the applicant’s response and as described in the AP1000 DCD, the safety related power distribution system has the capability of using portable generators to power safety related distribution panels, which power the level instruments. These panels are Seismic Category I and designed to remain operational following a safe shutdown earthquake. Based on the system description, the staff concludes that these design features are in conformance with Commission Order EA-12-051, and the guidance provided by JLD-ISG-2012-03.

Accuracy:

Commission Order EA-12-051, Attachment 2, Section 1.4 states that the instrument shall maintain its designed accuracy following a power interruption or change in power source without recalibration.

The applicant’s response states that the measured range of the SFP level by the safety-related instruments is from the top of the SFP to the top of the fuel racks, the level instruments are calibrated at a reference temperature suitable for normal SFP operation and will read conservatively at elevated temperatures, including during boiling conditions. These instruments are calibrated on a regular basis and their accuracy is not affected by power interruptions. All these design features are described in the supplemental information (WLS SUP 9.1-1) added to WLS FSAR Section 9.1.3.7.

Based on the system description provided above, the staff concludes that these design features are in conformance with Commission Order EA-12-051, and the guidance provided by JLD-ISG-2012-03.

Display:

Commission Order EA-12-051, Attachment 2, Section 1.5 states that the display shall provide on-demand or continuous indication of spent fuel pool water level.

The applicant's response states that the safety-related SFP level sensors provide continuous indication of the SFP level to the main control room as well as the Remote Shutdown Workstation and are included in the Qualified Data Processing System PMS display as indicated in Westinghouse AP1000 DCD Revision 19, Table 7.5-1 (Sheet 7 of 12). Safety-related instrumentation gives an alarm in the main control room when the water level in the SFP reaches the low-low-level setpoint as stated in AP1000 DCD Revision 19, Section 9.1.3.7.D.

Based on the system description provided above, the staff concludes that these design features are in conformance with Commission Order EA-12-051, and the guidance provided by JLD-ISG-2012-03.

License Condition

Commission Order EA-12-051, Attachment 2, Section 2 states that the spent fuel pool instrumentation shall be maintained available and reliable through appropriate development and implementation of a training program. Personnel shall be trained in the use and the provision of alternate power to the safety-related level instrument channels.

The applicant's COLA Part 10 includes License Condition 12.B, which requires the development and implementation of a training program in accordance with the guidance contained in JLD-ISG-2012-03.

The applicant's proposed license condition states:

> B. RELIABLE SPENT FUEL POOL LEVEL INSTRUMENTATION
>
> Prior to initial fuel load, DEC shall fully implement the following requirements for spent fuel pool level indication using the guidance contained in JLD-ISG-2012-03, Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation, Revision 0.

- The spent fuel pool instrumentation shall be maintained available and reliable through the development and implementation of a training program. The training program shall include provisions to insure trained personnel can route the temporary power lines from the alternate power source to the appropriate connection points and connect the alternate power source to the safety-related level instrument channels.

The proposed license condition is consistent with the guidance provided in JLD-ISG-2012-03, and is intended to ensure that the operators will be properly trained in the adequate equipment maintenance procedures and the proper operational procedures in order to establish the necessary alternate power connections. Based on this, the staff concludes that the proposed license condition is acceptable because the development and implementation of a training program is consistent with Commission Order EA-12-051 and the guidance provided by JLD-ISG-2012-03.

20.2.5 Post Combined License Activities

For the reasons discussed in the technical evaluation section above, the staff proposes to include the following license condition related to development and implementation of a training program:

- License Condition (20-2) – Prior to initial fuel load, the Licensee shall address the following requirements using the guidance contained in JLD-ISG-2012-03, Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation, Revision 0:

  The spent fuel pool instrumentation shall be maintained available and reliable through the development and implementation of a training program. The training program shall include provisions to ensure trained personnel can route the temporary power lines from the alternate power source to the appropriate connection points, and connect the alternate power source to the safety-related level instrument channels.

20.2.6 Conclusion

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to SFP instrument reliability, and there is no outstanding information expected to be addressed in the WLS COL FSAR.

The staff evaluated the applicant's and the AP1000 design description of the SFP water level instrument and determined that the instruments are in accordance with the guidance provided in JLD-ISG-2012-03. Therefore, the staff concludes that the applicant's SFP level instruments are considered reliable, able to withstand design-basis natural phenomena and monitor key SFP level parameters as described in Commission Order EA-12-051. In addition, the staff concludes that the information presented in the WLS COL FSAR is acceptable because it conforms to the guidance provided in JLD-ISG-2012-03. The staff based its conclusions on the following:

- WLS SUP 9.1-1 is acceptable because, when combined with the information in Table 7.5-1 and Sections 8.3.1.1.1 and 9.1.3.7.D of the AP1000 DCD, it includes provisions for SFP instrumentation arrangement, qualification, power sources, accuracy and display that are consistent with the requirements described in SECY-12-0025 and Commission Order EA-12-051.

- The proposed license condition is acceptable because it provides that, prior to fuel load, the licensee will have in place procedures for the proper maintenance of the level instruments and for the connection and use of an alternate power source in order to power the level instruments.

20.3 <u>Emergency Preparedness (Based on Recommendation 9.3)</u>

20.3.1 Introduction

The accident at Fukushima reinforced the need for effective emergency preparedness, the objective of which is to ensure the capability exists for a licensee (or COL applicant) to implement measures that mitigate the consequences of a radiological emergency and provide for protective actions of the public. The accident at Fukushima highlighted the need to determine and implement the required staff to fill all necessary positions of the emergency organization responding to a multi-unit event with impeded access to the site. Additionally, there is a need to ensure that the communication equipment relied on has adequate power to coordinate the response to an event during an extended loss of ac power.

20.3.2 Summary of Application

In Revision 11 of the WLS COL application, Part 10, the applicant proposed a license condition related to emergency preparedness communications and staffing. The staff's discussion is located in the Technical Evaluation section below.

20.3.3 Regulatory Basis

The requirements and guidance for emergency preparedness for beyond-design-basis external events are established or described in the following:

- 10 CFR 50.47(b)(1) states, in part: ". . . each principal response organization has staff to respond and to augment its initial response on a continuous basis."

- 10 CFR 50.47(b)(2) states, in part: ". . . adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, timely augmentation of response capabilities is available, . . ."

- 10 CFR 50.47(b)(6) states that provisions exist for prompt communications among principal response organizations to emergency personnel and to the public.

- 10 CFR Part 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," Section IV. E. 9. states that adequate provisions shall be made and described for emergency facilities and equipment, including "at least one onsite and one offsite communications system; each system shall have a backup power source."

- SECY-12-0025 states, in part, that the staff will also request all COL applicants to provide the information required by the orders and request for information letters described in this paper, as applicable, through the review process.

- NEI 12-01, "Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities," Revision 0 – By NRC letter from David Skeen, Director, Japan Lessons-Learned Directorate, to NEI, Susan Perkins-Grew, Director, Emergency Preparedness, dated May 15, 2012, NRC finds the guidance in NEI 12-01 to be an acceptable method for licensees to employ when responding to the 10 CFR 50.54(f) letters regarding NTTF Recommendation 9.3.

- 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," Section B, Onsite Emergency Organization, states in part:

  > 5. Each licensee shall specify . . . functional areas of emergency activity. These assignments shall cover the emergency functions in Table B-1 entitled, "Minimum Staffing Requirements for Nuclear Power Plant Emergencies." The minimum on-shift staffing shall be as indicated in Table B-1. The licensee must be able to augment on-shift capabilities within a short period after declaration of an emergency. This capability shall be as indicated in Table B-1.

- NUREG-0696, "Functional Criteria for Emergency Response Facilities," issued February 1981, offers guidance on how to meet the requirements of Appendix E to 10 CFR Part 50 and describes the onsite and offsite communications requirements for the licensee's emergency response facilities.

20.3.4 Technical Evaluation

The NRC issued RAI Letter No. 105 dated April 25, 2012 to the applicant, concerning implementation of the Fukushima NTTF Recommendation 9.3 in the combined license application for WLS. In response, the applicant proposed a license condition in the WLS COL application, to address the 10 CFR 50.54(f) request for information letters sent to existing licensees – including COL applicants - regarding communications and staffing for NTTF Recommendation 9.3. This license condition was subsequently revised in the license application. As part of its proposed license condition, the applicant committed to perform assessments for NTTF Recommendation 9.3 using NEI 12-01, Revision 0. By letter from the NRC to NEI dated May 15, 2012, the NRC stated that the guidance in NEI 12-01, Revision 0, provides an acceptable method for licensees to employ when responding to the 10 CFR 50.54(f) letters regarding NTTF Recommendation 9.3. The applicant proposed the license condition on communications and staffing in License Condition 12, Section C to Part 10 of the WLS COL application. The staff reviewed the applicant's proposed license condition and revised it to reflect the NRC's expectation when addressing NTTF Recommendation 9.3 as stated below in Section 20.4.5 of this SER. The NRC staff has revised the timeframe of the completion of this license condition to be consistent with the schedules provided in 10 CFR 52.99(a) and 10 CFR 52.103(a).

20.3.5 Post Combined License Activities

The license condition language in this section has been clarified from previously considered language. In a letter dated March 22, 2016 (ADAMS Accession No. ML16084A099), the applicant did not identify any concerns with the clarified license condition language. The changes do not affect the staff's above analysis of the conditions, and therefore, for the reasons discussed in the technical evaluation section above, the staff finds the following license conditions acceptable:

- License Condition (20-3) – No later than eighteen (18) months before the latest date set forth in the schedule submitted in accordance with 10 CFR § 52.99(a) for completing the inspections, tests, and analyses in the ITAAC, the licensee shall have performed an assessment of the on-site and augmented staffing capability for response to a multi-unit event. The staffing assessment shall be performed in accordance with NEI 12-01, "Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities," Revision 0.

  No later than one hundred eighty (180) days before the date scheduled for initial fuel load, as set forth in the notification submitted in accordance with 10 CFR § 52.103(a), the licensee shall revise the Emergency Plan to include the following:

  (a) Incorporation of corrective actions identified in the staffing assessment required by this license condition; and
  (b) Identification of how the augmented staff will be notified, given degraded communications capabilities.

- License Condition (20-4) – No later than eighteen (18) months before the latest date set forth in the schedule submitted in accordance with 10 CFR § 52.99(a) for completing the inspections, tests, and analyses in the ITAAC, the licensee shall have performed an assessment of on-site and off-site communications systems and equipment relied upon during an emergency event to ensure communications capabilities can be maintained during an extended loss of alternating current power. The communications capability assessment shall be performed in accordance with NEI 12-01, "Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities," Revision 0.

No later than one hundred eighty (180) days before the date scheduled for initial fuel load set forth in the notification submitted in accordance with 10 CFR § 52.103(a), the licensee shall have completed implementation of corrective actions identified in the communications capability assessment, including revisions to the Emergency Plan.

20.3.6 Conclusion

Based on the staff's review, the staff finds that the license condition, as revised by the staff above, is acceptable because it conforms to the guidance provided in SECY-12-0025 and NEI 12-01 regarding communications and staffing to address NTTF Recommendation 9.3, in NUREG-0654/FEMA-REP-1, and in NUREG-0696, and meets the applicable requirements in 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50.

21.0 DESIGN CHANGES PROPOSED IN ACCORDANCE WITH ISG-11

This safety evaluation report (SER) chapter contains the staff's evaluations of five requests from the Duke Energy Carolinas (DEC) William States Lee III Nuclear Station (WLS), Units 1 and 2 combined license (COL) applicant to depart from the AP1000 certified design referenced in the COL application. The applicant made the requests subsequent to determining that the departures in its COL application involved changes to the application that did not meet the criteria for post-COL deferral identified in Interim Staff Guidance DC/COL-ISG-011, "Finalizing Licensing-Basis Information." The five requests include six departures from the AP1000 certified design. Because each of the requests contains changes to the AP1000 Tier 1 information or technical specifications (TS), exemptions are required, in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, Appendix D, Section VIII, in order for the staff to find the departures acceptable. The applicant included exemption requests in its application, and the staff review of each request also appears in this chapter as part of each technical evaluation. The requests address the following five aspects of the AP1000 certified design:

- Passive core cooling system containment condensate return
- Main control room (MCR) dose
- MCR Heatup
- Hydrogen Vent Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)
- Neutron Flux Logic Operating Bypass

The staff evaluated each of the departures for impact on the WLS plant-specific probabilistic risk assessment (PRA). None of them have any impact on the quantification of core damage frequency or large release frequency. Only one (the departure relating to the passive core cooling system containment condensate return) resulted in a revision to any PRA-based insight. As discussed in Section 21.1.4 of this SER, this clarification did not alter any staff finding related to AP1000 design certification. The staff finds that the cumulative risk impact of these design changes and departures is acceptable.

For the staff's evaluations of the applicant's five exemption requests to depart from the AP1000 certified design, the staff applied the design centered review approach discussed in Section 1.2.3 of this SER. Under this approach, the staff performed a single review where multiple COL applicants submitted identical information. In this case, the reference COL is the Levy Nuclear Plant (LNP) Units 1 and 2, and the WLS COL is a subsequent COL.

21.1 <u>Passive Core Cooling System Containment Condensate Return</u>

21.1.1 Introduction

General Design Criteria (GDC) 34 of Appendix A to 10 CFR Part 50, requires that nuclear power plant designs have a system capable of removing residual heat, such that the decay heat does not exceed design limits for the fuel and pressure boundary. Inherent in this requirement is the need to bring the plant to a safe, stable condition following an anticipated transient. The AP1000 design accomplishes this function via the passive core cooling system (PXS). The PXS is designed to perform the following safety-related functions:

- emergency core decay heat removal
- reactor coolant system (RCS) emergency makeup and boration
- safety injection
- containment sump pH control

In order to support long term decay heat removal in a closed loop configuration, the AP1000 passive core cooling system must achieve a sufficient condensate return rate such that inventory in the in-containment refueling water storage tank (IRWST) is maintained in order to retain the heat transfer capability of the passive residual heat removal (PRHR) heat exchanger (HX). Water is steamed from the IRWST during transients that require the PRHR HX to remove decay heat from the RCS. The steam that reaches the containment shell condenses and returns to the IRWST through a gutter system. WLS DEP 3.2-1, a departure from the AP1000 design control document (DCD) requested by the applicant and reviewed below, proposes design changes to increase the fraction of condensate return to the IRWST and quantifies the condensate losses associated with the pressurization of the containment atmosphere, condensation on heat sinks within the containment, and from dripping or splashing from structures and components attached to the containment shell. WLS DEP 6.3-1, another departure reviewed below, makes further changes to the final safety analysis report (FSAR) supporting the design change proposed in WLS DEP 3.2-1.

21.1.2 Summary of Application

DEC incorporated in WLS COL application, dated April 11, 2016 (Agencywide Documents and Access Management System (ADAMS) Accession No. ML16124A854), the same information that Duke Energy Florida (DEF) incorporated into the LNP COL application related to the voluntary submittal of an exemption request and design change description for departure from the AP1000 DCD to address containment condensate cooling design. The information was originally submitted in endorsement and exemption request letters dated February 17, 2016 (ADAMS Accession Nos. ML16053A430, ML16053A431, and ML16050A173) and March 24, 2016 (ADAMS Accession No. ML16088A022).

Tier 1 and Tier 2 Departures

The applicant proposed the following Tier 1 and Tier 2 departures from the AP1000 DCD:

- WLS DEP 3.2-1 and WLS DEP 6.3-1

In WLS DEP 3.2-1, the applicant included a departure from Tier 1 and Tier 2 DCD information related to design changes of the containment condensate return system used to direct water that has condensed on the containment shell to the IRWST during accident scenarios. The Tier 2 departure includes additional information in FSAR Chapters 3, 5, 6, 7, 14, 15, and 19 as well as the TS and corresponding Bases appearing in Part 4 of the COL application. In addition, the applicant requested an exemption from the incorporation by reference of AP1000 DCD Tier 1 information, specifically Tier 1 Subsection 2.2.3, Tables 2.2.3-1 and 2.2.3-2. The exemption request proposes to revise the list of components in these tables to include additional components of the containment condensate return cooling system of the PXS.

In WLS DEP 6.3-1, the applicant included changes to FSAR Chapters 5, 6, 7, 9, 15, and 19 to address a departure related to quantifying the duration that the PRHR HX can maintain safe shutdown conditions, changing the description of the duration from indefinite to at least 14 days.

This exemption request involves a departure from Tier 1 Section 2.2.3, Tables 2.2.3-1 and 2.2.3-2, with Tier 2 involved departures. Therefore, these departures require NRC approval and are evaluated below.

21.1.3 Regulatory Basis

The changes proposed in WLS DEP 3.2-1 and WLS DEP 6.3.1 are required to meet the following GDC, which applies to the AP1000 DCD:

> 10 CFR Part 50, Appendix A, GDC 34, “Residual heat removal,” as it applies to the capability of the PRHR HX to perform safety related safe shutdown cooling of the RCS. Additionally, WLS DEP 3.2-1 and WLS DEP 6.3.1 are required to meet GDC 44, “Cooling Water,” as it applies to the ability of the containment systems to transfer heat from the PRHR HX to the ultimate heat sink via the passive containment cooling system.

21.1.4 Technical Evaluation

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and used this review in evaluating subsequent COL applications. To ensure that the staff’s findings on standard content that were documented in the SER for the reference COL application (LNP Units 1 and 2) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the LNP COL FSAR, Revision 9 to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from requests for additional information (RAIs).
- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.
- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting.

Tier 1 and Tier 2 Departures

- WLS DEP 3.2-1 and WLS DEP 6.3-1

The following portion of this technical evaluation section is reproduced from Section 21.1.4 of the LNP COL application FSER:

- *LNP DEP 3.2-1 and LNP DEP 6.3-1*

LNP DEP 3.2-1 proposes to change the PXS to increase the fraction of condensate returning to the IRWST when there is steam in the containment building. This change creates intermediate gutters at the top and bottom of the polar crane girder and at the containment shell intermediate ring stiffener. It blocks drain holes that were in these structures and adds dams where needed to collect condensate. It adds downspouts from these gutters to the IRWST. It also modifies the gutter drip lip so that condensate is not lost between the containment wall and the gutter. Condensate that is "lost" does not return to the IRWST, and instead drips off of the shell into various containment holdup volumes, such as the loop compartments or reactor vessel cavity.

LNP DEP 6.3-1 proposes additional changes to the FSAR in conjunction with the design changes described in LNP DEP 3.2-1 to clarify the duration of operation of the PRHR HX and separate the description of the safety functions from the non-safety design function of the PXS.

The staff reviewed a request for an exemption submitted by the applicant. The request proposed changes to Tier 1 Tables 2.2.3-1 and 2.2.3-2 and generic TS Surveillance Requirement (SR) 3.5.4.7 in the AP1000 DCD. Additionally, the staff reviewed the Tier 2 changes for potential effects on safety functions of the PXS and the associated Chapter 15 safety analyses, the safe-shutdown temperature evaluation in Chapter 19E, the seismic classification in Chapter 3, and the TS and Bases in Chapter 16. The regulatory evaluation of the exemption request appears in Subsection A, below, and the technical evaluation of the exemption request and departure appears in Subsection B, below.

A. Regulatory Evaluation of Exemption Request

A.1 Summary of Exemption

The applicant requested an exemption from the provisions of 10 CFR Part 52, Appendix D, Section III.B, "Design Certification Rule for the AP1000 Design, Scope and Contents," that require the applicant referencing a certified design to incorporate by reference Tier 1 information. Specifically, the applicant proposed to revise Tier 1 Tables 2.2.3-1 and 2.2.3-2 by adding components to the condensate return design to enable the PXS to more effectively perform its design functions and revised TS SR 3.5.4.7 to address downspout screens.[1]

A.2 Regulations

[1] While the applicant describes the requested exemption as being from Section III.B of 10 CFR Part 52, Appendix D, the entirety of the exemption pertains to proposed departures from Tier 1 information and generic TS in the generic DCD. In the remainder of this evaluation, the NRC will refer to the exemption as an exemption from Tier 1 information and generic TS to match the language of Sections VIII.A.4 and VIII.C.4 of 10 CFR Part 52, Appendix D, which specifically govern the granting of exemptions from Tier 1 information and generic TS.

- *10 CFR Part 52, Appendix D, Section VIII.A.4 states that exemptions from Tier 1 information are governed by the requirements of 10 CFR 52.63(b) and 10 CFR 52.98(f). It also states that the Commission may deny such a request if the design change causes a significant reduction in plant safety otherwise provided by the design. This subsection of Appendix D also provides that a design change requiring a Tier 1 change shall not result in a significant decrease in the level of safety otherwise provided by the design.*

- *10 CFR Part 52, Appendix D, Section VIII.C.4 states that an applicant may request an exemption from the generic TS or other operational requirements. The Commission may grant such a request only if it determines that the exemption will comply with the requirements of 10 CFR 52.7.*

- *10 CFR 52.63(b)(1) allows an applicant or licensee to request NRC approval for an exemption from one or more elements of the certification information. The Commission may only grant such a request if it complies with the requirements of 10 CFR 52.7 which in turn points to the requirements listed in 10 CFR 50.12 for specific exemptions, and if the special circumstances present outweigh the potential decrease in safety due to reduced standardization. Therefore, any exemption from the Tier 1 information certified by Appendix D to 10 CFR Part 52 must meet the requirements of 10 CFR 50.12, 52.7, and 52.63(b)(1).*

A.3 Evaluation of Exemption

As stated in Section VIII.A.4 of Appendix D to 10 CFR Part 52, an exemption from Tier 1 information is governed by the requirements of 10 CFR 52.63(b)(1) and 52.98(f). Additionally, the Commission will deny an exemption request if it finds that the requested change to Tier 1 information will result in a significant decrease in safety. Pursuant to 10 CFR 52.63(b)(1), the Commission may, upon application by an applicant or licensee referencing a certified design, grant exemptions from one or more elements of the certification information, so long as the criteria given in 10 CFR 50.12 are met and the special circumstances as defined by 10 CFR 50.12 outweigh any potential decrease in safety due to reduced standardization.
As stated in Section VIII.C.4 of Appendix D to 10 CFR Part 52, the Commission may grant an exemption from generic TS of the DCD only if it determines that the exemption will comply with the requirements of 10 CFR 52.7. As stated above, Section 52.7 points to 10 CFR 50.12 for specific exemptions.

Applicable criteria for when the Commission may grant the requested specific exemption are provided in 10 CFR 50.12(a)(1) and (a)(2). Section 50.12(a)(1) provides that the requested exemption must be authorized by law, not present an undue risk to the public health and safety, and be consistent with the common defense and security. The provisions of 10 CFR 50.12(a)(2) list six special circumstances for which an exemption may be granted. It is necessary for one of these special circumstances to be present in order for NRC to consider granting *an exemption request. The applicant stated that the requested exemption meets the special circumstances of 10 CFR 50.12(a)(2)(ii). That subsection defines special circumstances as when "[a]pplication of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule." The staff's analysis of each of these findings is presented below.*

A.3.1 Authorized by Law

This exemption would allow the applicant to implement approved changes to Tier 1 Tables 2.2.3-1 and 2.2.3-2 and generic TS SR 3.5.4.7. This is a permanent exemption limited in scope to particular Tier 1 information and generic TS, and subsequent changes to this information or any other Tier 1 information or generic TS would be subject to full compliance with the change processes specified in Sections VIII.A.4 and VIII.C.4 of Appendix D to 10 CFR Part 52. As stated above, 10 CFR 52.63(b)(1) allows the NRC to grant exemptions from one or more elements of the certification information, namely, as discussed in this exemption evaluation, the requirements of Tier 1. Moreover, Section VIII.C.4 allows the NRC to grant exemptions from generic TS if the exemption meets the requirements of 10 CFR 52.7 and 50.12. The NRC staff has determined that granting of the applicant's proposed exemption will not result in a violation of the Atomic Energy Act of 1954, as amended, or the Commission's regulations. Therefore, as required by 10 CFR 50.12(a)(1), the exemption is authorized by law.

A.3.2 No Undue Risk to Public Health and Safety

The underlying purpose of AP1000 Tier 1 Tables 2.2.3-1 and 2.2.3-2 and generic TS SR 3.5.4.7 is to ensure that the plant will be constructed and operated with a safe and reliable condensate return system in the event of an accident.

Additions to the condensate return portion of the passive core cooling system improve the reliability and effectiveness of the condensate return system; these additions to the system, therefore, support the system's intended design functions. The plant-specific Tier 1 DCD and TS will continue to reflect the approved licensing basis for the applicant and will maintain a level of detail consistent with that which is provided elsewhere in Tier 1 of the plant-specific DCD. The affected design description in the plant-specific Tier 1 DCD provides the detail to support the performance of the associated ITAAC. The proposed changes to Tier 1 information and generic TS are evaluated and found to be acceptable in Section 6.3 of this safety evaluation. Therefore, the staff finds the exemption presents no undue risk to public health and safety as required by 10 CFR 50.12(a)(1).

A.3.3 Consistent with Common Defense and Security

The proposed exemption would allow the applicant to implement modifications to the Tier 1 information and generic TS requested in the applicant's submittal. This is a permanent exemption limited in scope to particular Tier 1 information and a specific TS. Subsequent changes to this information or any other Tier 1 information or generic TS would be subject to full compliance with the change *processes specified in Sections VIII.A.4 and VIII.C.4 of Appendix D to 10 CFR Part 52. This change is not related to security issues. Therefore, as required by 10 CFR 50.12(a)(1), the staff finds that the exemption is consistent with the common defense and security.*

A.3.4 Special Circumstances

Special circumstances, in accordance with 10 CFR 50.12(a)(2)(ii), are present whenever application of the regulation in the particular circumstances would not serve the underlying purposes of the rule or is not necessary to achieve the underlying purpose of the rule. The underlying purpose of the specific Tier 1 Tables 2.2.3-1 and 2.2.3-2 and TS SR 3.5.4.7 being modified in the exemption request is to identify and conduct surveillances of the components that will be added to the design of the condensate return portion of the passive core cooling system. The additional components and new surveillance requirements for those components are needed so that the passive core cooling system can perform its intended function, that is, to bring the reactor coolant system to safe shutdown conditions during certain non-loss-of-coolant-accident events.

Application of the requirements in Tier 1 Tables 2.2.3-1 and 2.2.3-2 and generic TS SR 3.5.4.7 is not necessary to achieve the underlying purpose of those portions of the rule. The proposed additions to the condensate return portion of the passive core cooling system support the system's intended design functions, as does the addition of a generic TS to conduct surveillances of those additional components. The system and tables listing its components and surveillances, as modified in the requested exemption, will continue to perform their intended functions and will, therefore, meet the underlying purposes of the rule. Accordingly, because application of the requirements in Tier 1 Tables 2.2.3-1 and 2.2.3-2 and the generic TS SR 3.5.4.7 is not necessary to achieve the underlying purpose of the rule, special circumstances are present. Therefore, the staff finds that special circumstances exist as required by 10 CFR 50.12(a)(2)(ii) for the granting of an exemption from the Tier 1 information and generic TS described above.

A.3.5 Special Circumstances Outweigh Reduced Standardization

This exemption, if granted, would allow the applicant to change certain Tier 1 information incorporated by reference from the AP1000 DCD into the LNP COL application. An exemption from Tier 1 information may only be granted if the special circumstances of the exemption request, required to be present under 10 CFR 52.7 and 10 CFR 50.12, outweigh any reduction in standardization. The proposed exemption would modify the condensate return portion of the passive core cooling system to improve the reliability and effectiveness of the condensate return system. The proposed additions to the system support the system's intended design functions and the key design functions of the passive core cooling system will be maintained.[2]

[2] Based on the nature of the proposed changes to the generic Tier 1 information in Tables 2.2.3-1 and 2.2.3-2 and TS SR 3.5.4.7, both of which maintain and support the design functions of the

As described below in the technical evaluation, the changes to the condensate return system (1) ensure the capability of the PRHR HX to maintain the RCS in a safe, stable condition, as described in DCD Chapter 19E, "Shutdown Temperature Evaluation," and (2) demonstrate the existing non-loss-of-coolant accident (LOCA) analyses in Chapter 15 that credit the PRHR HX remain valid. Consequently, while there is a small possibility that standardization may be slightly reduced by the granting the exemption from the specified Tier 1 requirements, the proposed exemption modifying the condensate return portion of the passive core cooling system will improve the reliability and effectiveness of the condensate return system, to better allow the system to perform its intended function. For this reason, the staff determined that even if other AP1000 licensees and applicants do not request similar departures, the special circumstances supporting this exemption outweigh the potential decrease in safety due to reduced standardization of the AP1000 design, as required by 10 CFR 52.63(b)(1).

A.3.6 No Significant Reduction in Safety

The proposed exemption would modify the passive core cooling system from the design presented in the original application. As described below in the technical evaluation, the changes to the condensate return system (1) ensure the capability of the PRHR HX to maintain the RCS in a safe, stable condition, as described in DCD Chapter 19E, "Shutdown Temperature Evaluation," and (2) demonstrate the existing non-LOCA analyses in Chapter 15 that credit the PRHR HX remain valid. The proposed changes to the PXS design will increase the reliability of the system, maintain its key design functions, and will not adversely affect its function. Therefore, the staff finds that granting the exemption would not result in a significant decrease in the level of safety otherwise provided by the design, as required by 10 CFR Part 52, Appendix D, Section VIII.A.4.

A.4 Conclusion

The staff has determined that pursuant to Section VIII.A.4 of Appendix D to 10 CFR Part 52, the exemption: (1) is authorized by law, (2) presents no undue risk to the public health and safety, (3) is consistent with the common defense and security, (4) has special circumstances that outweigh the potential decrease in safety due to reduced standardization, and (5) does not significantly reduce the level of safety at the licensee's facility. The staff has also determined, pursuant to Section VIII.C.4 of Appendix D to 10 CFR Part 52 , that the generic TS portion of the exemption request: (1) is authorized by law, (2) presents no undue risk to the public health and safety, (3) is consistent with the common defense and security, (4) demonstrates the existence of special circumstances. Therefore, the staff grants the applicant an exemption from the requirements of Tier 1 Tables 2.2.3-1 and 2.2.3-2 and generic TS SR 3.5.4.7 of the generic DCD associated with the LNP Units 1 and 2.

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passive core cooling system, other AP1000 licensees and applicants may request the same exemption, preserving the intended level of standardization.

B. Technical Evaluation of Exemption Request and Departure

B.1 Passive Core Cooling System, Accident Analysis, and Shutdown Temperature Evaluation

Letter NPD-NRC-2014-005, submitted by the applicant and dated February 7, 2014, requested the previously described departures from 10 CFR Part 52, Appendix D, Section III.B. A revised submittal, letter NPD-NRC-2015-015, dated May 5, 2015, included two supporting reports as Enclosures 2 and 3: APP-GW-GLR-161, Revision 2 (proprietary) and APP-GW-GLR-607, Revision 2 (non-proprietary), respectively, both titled "Changes to Passive Core Cooling System Condensate Return." These reports describe the change and the basis for the change. In addition, APP-GW-GLR-161 and APP-GW-GLR-607 references three calculations and a test report further described below. Enclosure 6 provides the applicant's request for exemption related to this topic. Enclosures 7 and 8 present, respectively, changes to AP1000 DCD Revision 19 and the LNP COLA information that will be included in a future revision to the COLA. Letter NPD-NRC-2014-005 and its enclosures are the subject of the following review by the staff.

The applicant indicated that the changes described in LNP DEP 3.2-1 are necessary to (1) ensure the capability of the PRHR HX to maintain the RCS in a safe, stable condition, as described in DCD Chapter 19E, "Shutdown Temperature Evaluation," and (2) to demonstrate the existing non-LOCA analyses in Chapter 15 that credit the PRHR HX remain valid. The safe shutdown temperature evaluation, presented in DCD Chapter 19E Revision 19, assumes a constant condensate return fraction (the fraction of the water boiled off from the IRWST that will condense on the containment shell and return to the IRWST). Water that does not return to the IRWST can be referred to as condensate losses. The NRC staff understands that the applicant's analyses showed there are a number of mechanisms for condensate losses that vary with time including: steam to pressurize the containment atmosphere, condensation on passive heat sinks within the containment, and condensate splashing from the containment vessel and its attachments that does not reach to the PXS gutter system. The NRC staff's review of this departure request indicates some of these losses, such as the steam to pressurize the atmosphere, initially account for the majority of the condensation losses but decrease as the transient progresses, while other losses, such as the splashing from the attachments to the shell, are relatively time-independent and only a function of the amount of condensation on the shell. Condensate return is one of the primary factors influencing the performance of the PRHR HX.

Section 5.0, "Design Changes," of APP-GW-GLR-607 and APP-GW-GLR-161 detail the changes proposed by the applicant for increasing the condensate return rate. Subsection 1 describes the PXS downspout piping network added at the polar crane girder and stiffener, the routing for which is shown in the revised Figure 6.3-1 of the FSAR. Four collection points are located on both the upper portion and the lower flange of the polar crane girder and the stiffener ring that are routed to common lines that empty into two collection points already existing on either side of the IRWST. These downspouts, collection points and connecting *piping serve to capture condensate that previously would have been lost, and are sized such that any one line can accommodate the full flow anticipated during a transient to prevent a single failure from impacting the return flow to the IRWST. Subsection 2 describes the screens added to the downspouts and new guttering that is similar to screens existing on the IRWST gutter. These screens are designed to keep larger debris from blocking piping while still allowing condensate flow. The seismic qualifications of the downspouts and screens are further discussed later in this section. Subsection 3 explains how fabrication holes are blocked in the polar crane girder and the stiffener. Subsection 4 details the dam added to the polar crane girder to alleviate flow interactions between the containment shell and polar crane girder that contributed to losses. Furthermore, changes to the gutter drip lip and gutter routing were made to reduce losses from the gutter-wall interaction as much as possible. The effect of these changes on the transient analysis is described in detail below.*

The design changes, which are intended to reduce the condensate losses, prompted review of the analyses associated with transients that rely on condensate return. The effectiveness of the condensate return to the IRWST is captured in a series of proprietary calculations supporting the submittal, which were audited by the staff (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML14219A200 and ML15187A248) and are described in Section A.2 of APP-GW-GL-161 and APP-GW-GLR-607. The containment response is analyzed in calculation APP-PXS-M3C-071, "Containment Response Analysis for the Long Term PRHR Operation," via modifying the NRC-approved AP1000 WGOTHIC model used for containment peak pressure calculation that is part of the licensing basis, and provides transient containment pressure, temperature, and condensate holdup volumes input to the other calculations. Condensate losses implemented in WGOTHIC are obtained from a second calculation, APP-PXS-M3C-072, "Condensate Return to IRWST for Long Term PRHR Operation," which uses the parameters from WGOTHIC in concert with test results to provide a bounding condensate loss fraction from the containment shell. The test data used to calculate the losses are summarized in Section 4 of APP-GW-GL-161 and APP-GW-GLR-607 and described in detail in report TR-SEE-III-12-01, "AP1000 Condensate Return Test Report." A further calculation, APP-SSAR-GSC-536, "AP1000 Safe Shutdown Temperature Evaluation," incorporated the containment parameters and condensate behavior from the WGOTHIC analysis into LOFTRAN to calculate the behavior of the RCS and PRHR heat exchanger. This calculation was performed both for a 72-hour design basis case to verify that the assertions in Chapter 6 of the FSAR remain valid for all FSAR Chapter 15 events reliant on the PRHR, and for the 36-hour cooldown case depicted in Chapter 19 of the FSAR. A further calculation, APP-SSAR-GSC-009, "AP1000 Plant Safe Shutdown Duration Evaluation," justifies the duration of extended operation to 14 days using a LOFTRAN analysis. Further discussion of the analyses is located below in the "Evaluation of Containment Response," "Safety Design Bases," and "Non-Safety Design Bases" subsections of this SER section.

B.1.1 Evaluation of Containment Response

Although the staff audited the calculations referenced in the February 7, 2014 submittal by the applicant (ADAMS Accession Nos. ML14219A200 and ML15187A248), the submittal did not contain sufficient information for the staff to make a safety finding based on the docketed information, and thus the staff issued RAI 7439 in a letter dated March 6, 2014, asking the applicant to summarize the containment response calculation and its relationship with the other calculations. In its response dated May 5, 2014, the applicant provided a summary to address the impact of the cited calculation on the changes in LNP DEP 3.2-1. The staff requested in RAI 7439, Question 6.03-1, that the applicant provide additional detail on the results described in "Containment Response Analysis for the Long Term PRHR Operation" (ADAMS Accession Nos. ML14077A609 and ML14126A702), which describes the WGOTHIC model used to calculate the containment pressure and temperature as well as the steaming rate from the IRWST to the containment atmosphere, heat sinks and the containment shell, to address the technical merits of the changes in LNP DEP 3.2-1. The staff reviewed this response and finds it acceptable, as it provides an accurate summary of the analysis explaining how the containment response calculation relates to other calculations, inputs, and key results with sufficient information for the staff to make its finding.

Operation of the PRHR HX is affected by the amount of condensate returned to the IRWST. Therefore, in order to bound all events that credit the PRHR HX, the staff considered events requiring operation of the PRHR HX. The applicant identified the loss of normal feedwater coincident with a loss of alternating current (ac) power to the plant auxiliaries as the most limiting transient. The discussion below analyzes this scenario, and the justification for the loss of ac power as the most limiting transient is provided below in the "Safety Design Basis" subsection of this SER.

Using WGOTHIC, the applicant modeled the containment behavior during a transient involving the actuation of the PRHR by modifying the containment model used for the peak pressure calculation such that it conservatively captured the phenomena that would challenge the performance of the PRHR HX. This was accomplished by modifying the existing peak pressure calculation model in the following ways: increasing the area of the passive heat sinks as modeled by applying a multiplying factor, creating a volume to capture the condensate losses on the shell, adding a flow path to account for containment leakage, changing the IRWST (including a structure simulating PRHR heat exchanger using boundary conditions from LOFTRAN) to better represent the conditions during a non-LOCA transient, and adding a heat structure in the cavity to represent the vessel, among other minor changes. The net effect of these changes is to minimize the condensation rate on the containment inner shell, maximize the amount of steam and condensate that does not return to the IRWST—such as on passive heat sinks in containment and in the containment atmosphere—and maximize the amount of heat input to the IRWST, all of which are conservatisms for the non-LOCA transients that challenge the PRHR HX.

The addition of the heat structure to represent the reactor vessel in the reactor cavity, although used appropriately to capture a physical phenomenon present in the problem, is not the most conservative modeling choice with respect to the *calculation of condensate return. Most condensate that is lost from the containment shell eventually reaches the reactor cavity. This water fills the cavity to the point that it reaches the vessel and begins steaming. The vessel is surrounded by metallic insulation material designed to admit water through gaps and release the resultant steam through larger gaps between the insulation and the vessel. Although steaming from the reactor vessel cavity has competing effects on the system performance, as it both cools the reactor vessel and results in additional mixing below the operating deck, it does result in a larger net condensate return fraction to the IRWST. The applicant explored mechanisms that stimulate mixing within containment, but the precise extent of the mixing beneath the operating deck is not fully defined. The applicant states that additional mixing below the operating deck results in more condensate holdup on passive heat sinks, but also that in the long term steaming from the reactor vessel results in additional inventory return to the IRWST.*

The analysis in WGOTHIC accounts for the heat removal from the reactor vessel by subtracting it from heat that would be removed by the PRHR HX so that the energy balance is maintained. Temperature data from LOFTRAN is extracted and input into one boundary of the WGOTHIC vessel, while the other boundary exposed to the control volume uses a boiling correlation. The amount of heat removed by the boiling from the vessel is stored and subtracted from the PRHR HX heat input. Due to the nature of the modeling of the heat structure in the cavity in WGOTHIC, the entirety of the structure participates in heat transfer to the fluid in the reactor cavity. To mitigate against the effects of this, the applicant subtracted the volume in the cavity underneath the vessel and added it to the reactor coolant drain tank room so as to increase the holdup volume that must fill prior to condensate reaching the reactor vessel. This still results in additional boiling from the condensate that reaches the reactor vessel, as a larger area available (at least until the water would have reached the top of the bottom head) results in higher heat transfer. Conversely, in the very long term, the WGOTHIC model does not consider additional area that would participate as the water in the cavity rises above the lower head of the reactor vessel. In "Containment Response Analysis for the Long Term PRHR Operation," the applicant documents a sensitivity study that explores the effect on IRWST level of no condensate return resulting from reactor vessel steaming. The analysis shows that IRWST level is reduced by as much as 7 inches in the 72-hour period following the transient as a result of not accounting for reactor vessel steaming. This reduction in IRWST inventory does not appreciably impact system performance during the first 72 hours and would not challenge the operability of the system until much later in the transient. The staff performed a confirmatory analysis on the effect of the lower condensate return rate using LOFTRAN, which showed the lack of steaming from the reactor vessel would have less impact than was calculated by the applicant in their sensitivity study. In addition, the staff confirmatory calculation in MELCOR documented below tracks level along the reactor vessel heat structure and uses a conservatively high holdup volume such that steaming from the cavity is not established until almost one day into the transient. The applicant's design basis calculation bounds the confirmatory analysis performed by the staff. As a result, the staff finds the treatment of steaming from the vessel bottom head acceptable for this analysis.

The applicant made additional changes as compared to the approved W GOTHIC model used for peak pressure analyses in the most recent revisions of the calculations referenced in the May 5, 2015, submittal. The elevation of a modeled volume was changed, (resulting in changes to flow paths not representative of pipes but rather a function of the modeling divisions) in the analysis to prevent condensate build up in the control volume from inhibiting air flow between the control volumes to prevent non-physical behavior and better represent real conditions. The condensate return fraction was further modified to be a flat value representative of the loss rate determined by testing at the highest flow rate (discussed further below) plus a margin of 0.7 percent. In addition, the heat structures representing the PRHR HX and reactor vessel receive temperature conditions from iterative runs of the LOFTRAN model discussed later in the "Safety Design Basis" section of this report, rather than bounding values.

In the applicant's supporting analysis, condensation on most of the heat sinks is directly analyzed in WGOTHIC, while condensation holdup on surfaces such as the operating deck floor and other equipment was incorporated into a horizontal film holdup volume assumed proportional to the cross sectional area of containment multiplied by a factor with no provided justification. Therefore, in RAI 7439, Question 6.03-3, the staff requested that the applicant justify the multiplication factor used and the treatment of the horizontal film in the WGOTHIC model. In a response dated June 12, 2014, the applicant determined that the earlier treatment of film may not have been conservative. Thus, the applicant performed a sensitivity study to determine the effect of a different approach. The approach detailed in the response changed the representative area to a value incorporating the total surface area of the heat sinks modeled within containment in WGOTHIC, which are a conservative representation of the total passive heat sink area inside containment, incorporating the fixed components. For direct condensation in WGOTHIC, the applicant further increased this value to bound the total passive heat sink area within containment. Though this value does not directly represent the film holdup area as some heat sinks like the core makeup tanks (CMTs), polar crane girder and stiffener are excluded, the use of total surface area rather than horizontal surface area incorporates margin such that this treatment is conservative.

In addition, the applicant used a different approach to determine film thickness for condensation on surfaces utilizing a maximum contact angle for wetting in the design basis analyses and a more realistic contact angle for the "conservative, non-bounding" analyses to determine the thickness of the film. Although these changes increase the film holdup by a factor of more than three, there is a negligible effect on the performance of the PRHR HX during the first 72 hours. Initially following a non-LOCA transient, the significantly lower condensate return rates for the first few hours and lack of steaming from the reactor vessel cause the impact of additional holdup resulting from the more conservative film holdup calculation to be lessened and the level in the IRWST to be relatively unchanged. As condensate return increases to its long term value, and steaming from the reactor vessel begins to have a measurable impact on the transient, the submittal shows a minor reduction in the time before the RCS begins to reheat, well after the safety-related 72-hour period. The PRHR is required to remove decay heat following a design basis event for a minimum of 72 hours, in accordance with the *revised FSAR Section 6.3.1.1.1, “Emergency Core Decay Heat Removal” in LNP DEP 6.3-1. The staff verified that this calculation was incorporated into “Containment Response Analysis for the Long Term PRHR Operation” calculation in a subsequent audit (ADAMS Accession No. ML15187A248).*

The amount of condensation held up on surfaces within containment is also an important parameter during containment floodup following a LOCA or automatic depressurization system (ADS) actuation. Because the AP1000 relies on gravity for the driving force for recirculation in the long-term following an accident, the height of water in containment must be sufficient to force flow through the direct vessel injection lines for an opening in the RCS above the floodup level. The NRC staff’s confirmatory analysis applying the revised film holdup to the floodup calculation shows a negligible impact on the containment water level following a LOCA or ADS actuation. Thus, the staff finds the treatment of film holdup on surfaces within containment acceptable because it conservatively accounts for condensation on surfaces using conditions for maximum condensate losses, and does not adversely affect current bounding analyses for other transients.

Containment response heavily depends on the initial conditions assumed for the transient of interest. Containment pressure and temperature, IRWST temperature, and the ambient outside temperature (equal to passive containment cooling system (PCS) water temperature) all have an impact. Pressure response can be divided into two phases for this transient, an initial spike up in pressure as the IRWST boils off, followed by a slow levelling off to a peak and decay as passive cooling occurs. Confirmatory analysis performed by the staff using MELCOR for design basis conditions follows a similar trend as the analysis performed by the applicant documented in “Containment Response Analysis for the Long Term PRHR Operation” (ADAMS Accession No. ML14219A200), although the pressure calculated by the applicant bounds the pressure in MELCOR at all points within an hour after steaming begins for the design basis. For best estimate conditions, the staff’s confirmatory analysis shows a peak pressure of approximately 2 pounds per square inch greater than the applicant’s WGOTHIC analysis, while design basis conditions result in confirmatory analysis yielding a pressure approximately 5 pounds per square inch less than the conservative value calculated by the applicant in WGOTHIC; these events, like all events involving PRHR actuation, do not challenge the design pressure. More importantly for this transient, the applicant’s pressure used for the design basis analysis results in a higher saturation pressure for water in containment, which results in additional holdup in the containment atmosphere and higher IRWST temperatures and, therefore, reduced heat transfer through the PRHR. As such, the applicant’s modeled pressure response in containment is conservative because it uses bounding inputs into an approved methodology and yields a more conservative value than staff models of the same conditions.

In each analysis performed by the applicant, calculations were performed for design basis conditions for Chapter 15 and “non-bounding, conservative” conditions for Chapter 19. Design basis conditions should represent the conservatively bounding set of values for any given transient, and the design basis values for the maximum temperature inside containment is 120 degrees Fahrenheit (°F) (48.9 degrees Celsius (°C)) and outside *containment is 115 °F (46.1 °C). The analysis submitted used an in-containment initial temperature of 85 °F (29 °C) (capturing all the heat sinks as well as the IRWST) and an environment temperature of 115 °F (46.1 °C). In RAI 7439, Question 6.03-4, the staff requested the applicant justify the assumption of 85 °F (29 °C) for the initial temperature of containment for the design-basis accident (DBA) analysis. In the response dated July 1, 2014, the applicant explained that the effect of the temperature of the heat sinks outweighed the effect of the IRWST temperature. That is, a lower heat sink temperature results in more condensation on heat sinks and, therefore, more losses when compared with the effect of a change in the initial enthalpy in the IRWST, which affects the time to begin boiling. The NRC staff reviewed analysis supporting this assertion (ADAMS Accession No. ML14219A200), and although the effect is slight, lower heat sink temperatures result in a lower IRWST level as the transient progresses.*

The choice of 85 °F (29 °C) for in-containment initial temperature was based on the use of an exterior temperature of 115 °F (46.1 °C), the TS maximum for ambient air temperatures for the environment outside containment. The applicant performed a study for a plant located at a site where meteorological data indicates ambient temperatures could reach 115 °F (46.1 °C) and calculated in-containment temperatures for an operating facility with containment coolers running to show that containment temperatures (and therefore the temperatures of the heat sinks and the IRWST) would not reach below 88 °F (31 °C) for an ambient temperature of 115 °F (46.1 °C). The influence of exterior temperatures is more dramatic on PRHR HX performance: while lower temperatures inside containment would result in additional condensation on heat sinks, higher ambient temperatures result in higher initial PCS water temperatures, which result in less heat removal from containment during a transient and thus higher containment pressures and temperatures. The staff agrees that 85 °F (29 °C) for the in containment temperature presents an acceptably conservative value for a transient given a bounding environmental temperature of 115 °F (46.1 °C), due to the large thermal inertia of the heat sinks within containment and the sizable heat load for the operating plant under the steady state conditions leading up to the transient, in addition to the applicant's justification based on ambient temperatures.

Section 6.3.2.1.1 of the revised FSAR, "Emergency Core Decay Heat Removal at High Pressure and Temperature Conditions," in LNP DEP 6.3-1, addresses the impact of the revised analysis due to the design changes. The revised FSAR discusses the integrated system, including emphasis on the condensate return features, and explicitly describes the mechanics of in-containment condensation as the heat transfer mechanism. In addition, the FSAR now highlights that "[c]ondensation that is not returned to the in-containment refueling water storage tank drains to the containment sump." This is in accordance with the staff's understanding of the system as discussed in this subsection, and is acceptable because most water that does not return to the IRWST fills holdup volumes, which must fill to a certain level before overflowing and eventually reaching the lowest point in containment and filling the reactor coolant drain tank room and reactor cavity.

Section 6.3.2.1.1 also explains the impact of the condensate return rate on the duration of operation of the PRHR HX, and explains that if ac power is not *recovered, the PRHR HX can continue to perform for a period of time beyond 72 hours. The plant also retains the ability to transition to open loop cooling via the automatic depressurization system if inventory in the IRWST is insufficient. This agrees with the staff analysis of the performance of the system and is an acceptable change to the FSAR, discussed further in the following section, "Safety Design Basis."*

The changes made to Figures 6.3-1 and 6.3-2 in the FSAR appropriately capture the design changes as modeled in the analyses described in the submittal and are acceptable. The components in these figures added to Tier 1 are discussed in the "Classification of Structures, Components, and Systems" subsection below.

The applicant stated that the modifications referenced above to the WGOTHIC model, such as those incorporating condensate return to the IRWST, have no effect on the peak containment pressure calculation. Peak containment pressure is reached well before condensate return has a measurable impact on the transient, and any benefits from condensate return at later times are not credited. The addition of downspouts at the polar crane and stiffener have no impact on the current peak pressure analysis because the model already assumes that condensate reaching the polar crane and stiffener makes its way to the reactor coolant drain tank room, which overflows to the reactor cavity region. The assumptions used in these analyses for initial conditions for temperature, humidity, and heat sink area limiting the amount of condensate return are less bounding for the case of peak containment pressure and, therefore, would not be applicable to the peak pressure calculation. The staff finds the peak pressure analysis in the licensing basis is unaffected by the changes implemented in the current analyses.

For the analyses supporting LNP DEP 3.2-1, the treatment of the PCS water coverage of the outside of the containment shell is consistent with that used in the peak pressure calculation model previously approved by the staff. That is, an assumed film coverage below the weir of 90 percent (for design basis conditions) at nominal flow rates, decreasing as the level in the PCS water storage tank drops during the 72-hour period (discussed in Section 6.2.1 of NUREG-1273 and Table 6.2.2-1 of the AP1000 DCD). Thus, that treatment is conservative for this analysis, as minimizing shell coverage maximizes the energy within containment, which maximizes the containment pressure and saturation temperature.

The calculation, "Containment Response Analysis for the Long Term PRHR Operation," receives inputs from the "Condensate Return to IRWST for Long Term PRHR Operation" calculation (ADAMS Accession No. ML14219A200), which calculates the effective condensate losses on the inside surface of the containment shell. The NRC staff requested in RAI 7439, Question 6.03-2 that the applicant submit additional detail on the results described in "Condensate Return to IRWST for Long Term PRHR Operation," which describes the methodology used to calculate losses over the containment shell, including the tests used to determine losses over attachments to the shell. This request was to address deficiencies in the submittal related to insufficient justification of the applicability of the development of the condensate loss model. The applicant summarized the calculation in a response dated June 12, 2014. The NRC staff *reviewed the response and found it acceptable because it provides a summary with sufficient information on the calculation for the staff to make its finding.*

Tests for losses over attachments to the shell were performed at lower temperatures than the prototypic conditions on the containment shell during a non-LOCA transient, which could peak in excess of 220 °F (104 °C). Therefore, in RAI 7439, Question 6.03-5, the staff requested the applicant justify the extrapolation from the losses for tested values of condensate losses over attachments to the wall to the values used in the analysis at containment pressure and temperature. In its response to the RAI dated June 27, 2014, the applicant explained that although the losses over wall attachments are extrapolated, the extrapolation is overly conservative and prior research indicates that film thickness should decrease at the same Reynolds number at higher temperatures and thus decrease the condensate losses. In addition, the applicant performed sensitivity studies on the effect of increasing the losses on the performance of the PRHR HX. Those sensitivities indicate that even for a case when losses over attachments are increased by a factor of 1.4 to 1.75, there is a negligible effect on the performance of the system in the first 72 hours and only a minor (approximately 5 percent) reduction in the long term capability of the system. The NRC staff remains unconvinced as to the validity of the applicant's temperature scaling argument, especially given the relative variance in the test results. However, on the basis of the large degree of conservatism inherent in the extrapolation and the fact that a further 40 percent increase in losses over wall attachments results in an insignificant impact to the system performance, the staff finds the treatment of film losses over attachments to the containment shell acceptable.

The analysis described above using WGOTHIC passes a set of inputs to analyses in LOFTRAN (discussed below). The applicant extracts a table including time, condensate return flow, condensate temperature, IRWST steaming rate, containment pressure, and CMT compartment temperature. The data for condensate return flow and condensate temperature are combined to create a recirculation ratio (the fraction of boil off from the IRWST returning as condensate). The recirculation ratio and containment pressure are then used in the LOFTRAN analysis; in the case of the LOFTRAN run using design basis conditions, the recirculation ratio is further reduced and the pressure is increased from the values calculated in WGOTHIC for additional conservatism.

On the bases that the modifications to the gutter system are appropriately incorporated into the analyses for events that actuate the PRHR, that the data from tests used to determine the losses on the containment shell conservatively bound realistic losses, and that condensate loss mechanisms have been quantified and captured in the analysis, the staff finds the treatment of containment conditions in calculations supporting LNP DEP 3.2-1 and LNP DEP 6.3-1 acceptable. Therefore, the staff finds the proposed LNP DEP 3.2-1 FSAR revisions related to containment response noted above to be acceptable pending the staff's confirmation that the proposed FSAR revisions are incorporated in the LNP Units 1 and 2 COL application. The staff is tracking these revisions as ***LNP Confirmatory Item 21.1-1.***

Resolution of LNP Confirmatory Item 21.1-1

LNP Confirmatory Item 21.1-1 is a commitment by the applicant to revise the LNP COL FSAR to provide additional information related to containment response as indicated in the letter dated January 14, 2016. The staff confirmed that the LNP COL FSAR has been appropriately revised. As a result, LNP Confirmatory Item 21.1-1 is now closed.

B.1.2 Safety Design Bases

The PXS performs the following safety-related functions:

1. *Emergency decay heat removal*
2. *Emergency reactor makeup/boration*
3. *Safety injection*
4. *Containment pH control*

The following subsections evaluate the impact of LNP DEP 3.2-1 and LNP DEP 6.3-1 on each safety function of the PXS.

B.1.2.1 Emergency Decay Heat Removal

LNP DEP 3.2-1 impacts the condensate return rate to the IRWST and thus impacts the emergency decay heat removal function of the PRHR HX. Under LNP DEP 3.2-1 and LNP DEP 6.3-1, the revised FSAR Section 6.3 states that for non-LOCA events in which a loss of core decay heat removal capability via the steam generators (SGs) occurs, the PRHR HX is designed to perform the following functions:

1. *Remove core decay heat following a design basis event.*
2. *Maintain acceptable reactor coolant system conditions for a minimum of 72 hours following a non-LOCA event. Applicable post-accident evaluation criteria are specified in Chapter 15.*
3. *Sufficiently reduce RCS temperature and pressure during an SG tube rupture (SGTR) event to terminate breakflow, without overfilling the SG.*

Emergency decay heat removal functions 1 and 3 are design criteria that have been evaluated in DCD Chapter 15, Revision 19 for the events identified in Table 21.1-1 and reviewed in NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design." Previous staff review of DCD Chapter 15 events did not consider the possibility of PRHR HX tube uncovery. Therefore, calculations could be terminated once the acceptance criteria for the design basis events were initially met. LNP DEP 3.2-1 revealed that the PRHR HX can provide cooling for a finite period of time before performance degrades and transition to open-loop cooling, via ADS actuation, is required to maintain the reactor in a safe, stable shutdown condition. LNP DEP 3.2-1 states that the water level in the IRWST remains above the uppermost points of the PRHR HX for the duration of all DCD Chapter 15 analyses and, therefore, there is no impact to the calculated heat transfer through the heat exchanger. This *caused the staff to question the mission time for the PRHR HX and the termination criteria for DCD Chapter 15 analyses for events that credit the PRHR HX (Table 21.1-1).*

Table 21.1-1. Chapter 15 Events that Credit the PRHR HX for Decay Heat Removal

DCD Section	*Scenario*	*Calculation Duration*
15.2.6	*Loss of AC Power to Plant Auxiliaries*	*6.2 hrs*
15.2.7	*Loss of Normal Feedwater Flow*	*5.4 hrs*
15.2.8	*Feedwater System Pipe Break*	*3.1 hrs*
15.5.1	*Inadvertent Operation of CMTs During Power Operation*	*8.6 hrs*
15.5.2	*CVCS Malfunction that Increases RCS Inventory*	*5.6 hrs*
15.6.3	*Steam Generator Tube Rupture*	*6.7 hrs*

Section 4.3.3.5 of the Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document (URD) and Section 2.3.2 of the staff's corresponding safety evaluation (NUREG-1242, "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document, Evolutionary Plant Designs," Volume 3) both state that a design expectation for the passive decay heat removal system is to have sufficient water capacity in the passive decay heat water pools to permit 72 hours of operation after SCRAM without the need for refill. The 72-hour capacity of the passive residual heat removal system was approved by the Commission in their responses to SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," and SECY-95-132, "Policy and Technical Issues Associated with Regulatory Treatment of Non-Safety Systems in Passive Plant Designs (SECY-94-084)." Based upon the Commission position expressed in SECY-94-084 and SECY-95-132, the licensing guidance in the URD, NUREG-1242, "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document, Evolutionary Plant Designs," and the Regulatory Treatment of Non-Safety Systems as discussed in Section 19.3 of the Standard Review Plan, in order for the PRHR HX to meet the requirements of GDC 34 and GDC 44, the IRWST should have sufficient capacity to permit a minimum of 72 hours of operation after SCRAM following an accident without the need for refill. In RAI-7475, Question 6.03-10, the staff requested clarification of the mission time for the PRHR HX. In a response dated June 27, 2014 (ADAMS Accession No. ML14182A106), the applicant stated that the PRHR HX operates to bring the RCS to an acceptable, stable condition and maintain this condition for at least 72 hours after a non-LOCA event to allow ample time for decision-making and initiation of recovery actions. During this 72-hour time period, applicable Chapter 15 design basis safety evaluation criteria are met. The 72-hour operational requirement for the PRHR HX following a non-LOCA event is consistent with the Commission position for compliance with GDC 34 and GDC 44.

DCD Chapter 15 analyses that credit the PRHR HX, shown in Table 21.1-1, terminate before the 72-hour operational requirement of the PRHR HX. This caused the staff to question the possibility of PRHR HX tube uncovery during the 72-hour time period, and the resulting impact to Chapter 15 analyses. In RAI

7440, Question 15.02.06-2, the staff requested the applicant to (1) identify the bounding Chapter 15 event in terms of PRHR HX performance, and (2) extend the calculation for the bounding event out to 72 hours in order to demonstrate the 72-hour operational requirement of the PRHR HX.

In their response dated June 27, 2014 (ADAMS Accession No. ML14182A106), to the first part of RAI 7440, Question 15.02.06-2, the applicant identified the Loss of AC Power to Plant Auxiliaries (LOAC) as the limiting event in terms of PRHR HX performance. The applicant explained that the LOAC event combines a relatively late reactor trip with a significant loss of secondary side inventory in both steam generators, and a loss of forced reactor coolant flow. It therefore, represents the largest mismatch between primary side energy and secondary side/PRHR HX heat removal capability. The applicant's response to RAI 7440, Question 15.02.06-2 included a sensitivity study, performed with the MAAP4.0.7 code, to evaluate the impact of different events on PRHR HX performance. The results demonstrated that the plant response to different events begins to converge after approximately 8 hours into the event with the LOAC event producing slightly bounding heat loads on the PRHR HX over the 72-hour calculation time. The NRC staff performed confirmatory calculations as part of the review, which include a sensitivity study to investigate the impact of the initiating event. The result of the staff's sensitivity study is consistent with the applicant's response to RAI 7440, Question 15.02.06-2. Based upon considerations discussed in this paragraph, the staff finds the selection of LOAC as the limiting event in terms of PRHR HX performance to be acceptable.

In their response to the second part of RAI 7440, Question 15.02.06-2, the applicant performed a 72-hour calculation of the LOAC event. The analysis utilized the LOFTRAN code to model the response of the reactor coolant system. In evaluating the applicant's response, the staff evaluated the analytical procedure (i.e., use of LOFTRAN) and the results of the calculation. In the NRC staff's safety evaluation for the AP1000 DCD, NUREG-1793, the staff concluded that the applicant's use of LOFTRAN as described in WCAP-15644 (ADAMS Accession No. ML040890663) is acceptable for licensing calculations of the AP1000 subject to the following limitation:

- *LOFTRAN is approved to analyze the transients listed in Table 21-2 of NUREG-1793. Use of the code for other analytical purposes will require additional justification.*

Previous licensing calculations that utilized LOFTRAN extended less than 10 hours and did not experience uncovery of the PRHR HX tubes. Thus, the staff investigated the applicability of the code to the analyses referenced in the departure. Modeling of tube uncovery in LOFTRAN uses a collapsed liquid level within the IRWST, where surface area of the PRHR HX above the collapsed liquid level is not credited for heat removal. The surface area below the liquid level is calculated as described in WCAP-14235 (ADAMS Accession No. 9709290174) and approved in the staff's safety evaluation of the AP1000 DCD in NUREG-1793. During pool boiling, the secondary side heat transfer is modeled using a modified Rosenhow correlation. This modified Rosenhow correlation was developed from experimental data obtained from the AP600 PRHR HX test *program described in WCAP-13573 (ADAMS Accession No. 9705280203). The AP600 PRHR HX test program included a series of tests where PRHR HX tubes were uncovered to different levels (75 percent, 50 percent, and 25 percent) which demonstrated insignificant heat transfer for the uncovered tubes and heat transfer consistent with nucleate boiling for the covered tubes. Details of the staff review of the PRHR HX test program are available in Section 21.5.3 of NUREG-1512, "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design." Of specific concern were the flow distribution and behavior in the tubes and two-phase flow behavior in the IRWST, especially within the tube bundle. High heat transfer rates could cause violent boiling on the outer surface of the tube, resulting in vapor blanketing of some portion of the heat exchanger surface and drastic reduction in heat transfer. Westinghouse analyzed the PRHR HX performance and concluded that it is unlikely that vapor blanketing could occur, and that if it did occur, such behavior would be limited to a very short length near the inlet of the tube bundle, leaving sufficient heat transfer area to meet its design performance requirements. Based upon the Westinghouse analysis and that vapor blanketing was not observed at any of the integral test facilities (OSU/APEX, SPES-2, or ROSA/LSTF), the staff concluded in NUREG-1512 that Westinghouse resolved the concern of vapor blanketing. The potential for the vapor generated by the lower tubes to impede the heat transfer of the upper (covered) tubes is reduced as the PRHR HX begins to uncover. Based upon considerations discussed in this paragraph, the staff finds the previous resolution of the vapor blanketing issue to remain valid for the case of tube uncovery and the heat transfer modeling of the PRHR HX to be acceptable.*

In order to understand the limits of the analysis, the staff explored additional input considerations. In RAI 7475, Question 6.03-10, the staff requested the tube plugging assumption used for DBA analyses. In the response, dated June 27, 2014, the applicant stated that a design change was implemented to reduce the allowable number of plugged tubes for the PRHR-HX from the number of tubes making up 8 percent of the heat transfer area to the number of tubes making up 5 percent of the heat transfer area. However, the original 8 percent assumption is utilized for the DBA analysis presented in the response to RAI 7440, Question 15.02.06-2. Existing Chapter 15 analyses assume 8 percent tube plugging in the PRHR-HX (in terms of heat transfer area) for scenarios where minimizing heat removal is bounding and 0 percent tube plugging in the PRHR-HX where maximizing heat removal is bounding (e.g., steam line break). Boundary conditions for the containment response (i.e., containment pressure and condensate return ratio) were input as functions of time and have been evaluated above in subsection "Evaluation of Containment Response" of this SER. During an audit, the NRC staff identified that the initial power utilized in the 72 hour analysis accounted for a 1 percent uncertainty. Section 15.0.3.2 of the AP1000 DCD, Revision 19, states that a 1 percent uncertainty is supported by the main feedwater flow measurement instrumentation, but that a bounding value of 2 percent is used in the analysis. The Levy COL FSAR contains COL Information Item STD COL 15.0-1, which identifies the plant operating instrumentation which when properly calibrated will support 1 percent uncertainty in the core power based on flow measurement uncertainty. Additionally, the NRC staff performed a sensitivity study investigating the impact of the reduced core power uncertainty on the 72-hour LOAC event. The results of this study *demonstrated that the reduction in core power uncertainty has an insignificant impact on the RCS response and Chapter 15 acceptance criteria.*

The analysis of the LOAC event submitted by the applicant demonstrates that during the 72-hour period the top horizontal portion of the PRHR HX becomes uncovered. However, the PRHR HX capacity remains sufficient to prevent RCS heatup for a time period greater than 72 hours. The submitted analysis demonstrates that once the Chapter 15 acceptance criteria are satisfied, at approximately 6.2 hours, they remain satisfied for a time period exceeding 72 hours. The NRC staff performed confirmatory calculations as part of the review, which include a 72-hour analysis of the LOAC event. The staff's confirmatory calculation for the LOAC event is consistent with the applicant's submitted analysis. Based upon the identification of the LOAC event being the bounding event in terms of PRHR HX operation, the acceptable modeling of the LOAC event, and the result demonstrating the 72-hour operational requirement for the PRHR HX, the staff finds the submitted analysis of the 72-hour LOAC event acceptable.

In a letter dated January 14, 2016 (ADAMS Accession No. ML16020A250), the applicant updated their submittal, which included the consideration of ambient heat losses from the RCS during Chapter 15 non-LOCA events. Previous analyses had assumed the RCS to be adiabatic, which would result in the highest required heat removal from the PRHR HX; due to ambient heat losses from the RCS, from the pressurizer in particular, and in the absence of positive pressure control associated with pressurizer heaters, the applicant was concerned that pressure in the RCS could be reduced to the point that subcooled margin is lost. A loss of subcooling was thought to have the potential to inhibit the performance of the PRHR HX. Additional analyses were conducted by the applicant to investigate the impact of ambient heat loss from the RCS. A description of these analyses is provided in APP-GW-GLR-607, Revision 4 "Changes to Passive Core Cooling System Condensate Return," which is included as an enclosure to the letter of January 14, 2016. The NRC staff audited the supporting calculations (documented in the audit report, ADAMS Accession No. ML16034A034). The audit resulted in a supplemental RAI response, provided in letter dated January 14, 2016 (ADAMS Accession No. ML16020A105), to establish the basis for the ambient heat losses associated with the pressurizer. The RAI response included (1) a description of the ambient heat loss flow paths from the pressurizer and their treatment in transient analyses, and (2) a FSAR update to Section 5.4.5.2.1 to include the average maximum heat transfer rate specification for the metallic reflective insulation installed on the external surfaces of the RCS. The NRC staff found the RAI response identified the applicable heat loss mechanisms from the pressurizer during a DBA. NRC reviewed the details of the heat loss calculation during their audit of the supporting calculations and observed that additional conservatism was included in pressurizer heat loss calculations. Additionally, the NRC staff performed confirmatory calculations for the heat losses from the pressurizer which resulted in values that were consistent with the applicant's analyses. The conservative modeling of the heat losses from the pressurizer is further supported by data from applicable literature identified in the NRC staff's audit report. Based upon the information discussed above, the NRC staff finds the *treatment of ambient heat losses in the analysis of DBAs to be suitably conservative. The applicant performed a DBA analysis that considers ambient heat losses, performed with LOFTRAN, showing that the RCS remains subcooled for a time period exceeding 72 hours. Therefore, the only impact on the DBA analysis was a lower temperature in the RCS due to the increased heat removal. The NRC staff performed confirmatory calculations as part of this review and obtained results that were consistent with the applicant's analysis. Based on the information in this paragraph, the NRC staff finds that ambient heat losses do not adversely impact DBA analyses for the AP1000.*

The staff performed confirmatory calculations, which included the Chapter 15 LOAC event, to assist in evaluating the impacts of LNP DEP 3.2-1 to Chapter 15. The calculations caused the staff to question whether containment backpressure effects on PRHR HX performance were accounted for in Chapter 15. During the staff audit of the applicant's documents related to LNP DEP 3.2-1 and LNP DEP 6.3-1 (ADAMS Accession No. ML14219A200), the staff verified that in Revision 19 of the DCD, Chapter 15 analyses that credit the PRHR HX for decay heat removal do not account for containment backpressure effects on the PRHR HX. Not accounting for containment backpressure on PRHR HX performance introduces a slightly non-conservative boundary condition that affects PRHR HX performance late in the transient. However, the staff verified that this effect does not alter the conclusions of Chapter 15 analyses and thus produces no consequential impact.

The change from indefinite operation of the PRHR HX to the 72-hour operational requirement, and subsequent analysis demonstrating the 72-hour operational requirement, are reflected in the applicant's proposed changes under FSAR Sections 5.4, 6.3, 7.4, and Table 19.59-18 in letter dated June 27, 2014. In the proposed FSAR changes noted above, indefinite operation is changed to extended operation at several locations. For consistency among the proposed changes, the staff is interpreting extended operation to be at least 72 hours. Based upon the considerations discussed within this subsection, the staff finds the proposed FSAR revisions noted above to be acceptable pending the staff's confirmation that the proposed revisions are incorporated in the LNP Units 1 and 2 COL application. The staff is tracking these revisions as ***LNP Confirmatory Item 21.1-1****.*

<u>Resolution of LNP Confirmatory Item 21.1-1</u>

LNP Confirmatory Item 21.1-1 is a commitment by the applicant to revise the LNP COL FSAR to provide additional information related to ambient heat losses as indicated in the letter dated January 14, 2016. The staff confirmed that the LNP COL FSAR has been appropriately revised. As a result, LNP Confirmatory Item 21.1-1 is now closed.

Indefinite is still used in the revised FSAR (in Sections 6.3.1.1.4, 6.3.3.3.3, 6.3.3.4.3, and 7.4) when considering the entirety of the passive core cooling system; that is, when ADS is actuated and the system transitions to open-loop cooling with gravity driven injection. At that point, the system is nominally limited *by normal containment leakage. This treatment remains unchanged from the system as reviewed by the staff in Revision 19 of the DCD.*

B.1.2.2 Emergency Makeup and Boration

Emergency makeup and boration for non-LOCA events are functions performed by the CMTs and are not impacted by LNP DEP 3.2-1.

B.1.2.3 Safety Injection

LNP DEP 3.2-1 is evaluated to ensure ADS actuation and transition to open loop cooling is retained as a defense-in-depth means of providing emergency core cooling during non-LOCA events. The evaluation includes investigating the impact of IRWST level on the performance of the ADS spargers, the impact of LNP DEP 3.2-1 on the containment floodup level, and the availability of the ADS, IRWST injection, and containment recirculation valves during an extended station blackout.

In the event that operator action is taken to prolong closed loop mode of PXS operation for an extended period of time, the level in the IRWST can drop below the ADS spargers, causing the staff to question whether ADS actuation can be inhibited by a low IRWST level. In RAI 7440, Question 15.02.06-1, the staff requested information regarding the minimum IRWST level required for ADS actuation. In a letter dated June 19, 2014, the applicant stated that no minimum IRWST level is required for ADS actuation because:

1. *ADS spargers do not limit the containment pressure increase for the bounding mass and energy release. The associated mass and energy release attributed to ADS actuation is bounded by the large break LOCA accident or a large main steam line break inside containment.*
2. *IRWST vents are more than sufficient to vent the amount of steam released if ADS Stages 1-3 are actuated after the spargers are uncovered. The IRWST vents are sized to vent steam relief from ADS stages 1-3 at high system pressures following several hours of PRHR HX operation during which the IRWST has reached saturation pressure.*
3. *During a long-term non-LOCA event, during which the IRWST level has fallen below the elevation of the ADS spargers, RCS pressure at the time of ADS actuation will be relatively low.*
4. *Steam relief from uncovered ADS spargers actually improves ADS Stages 1-3 performance due to the lower backpressure provided by the IRWST water. Limitations are imposed on the maximum sparger submergence depth to limit sparger discharge backpressure.*
5. *No damage is done to spargers, IRWST, or surrounding structures.*

The NRC staff identifies the reasons as valid, but requested further justification for the argument that no damage is done to the ADS spargers, IRWST, or surrounding structures. In a supplemental letter dated July 24, 2014, the applicant stated that the ADS spargers are designed to withstand spurious actuation of ADS Stages 1-3 at normal operating conditions. Spurious actuation of ADS Stages 1-3 is bounding in terms of stress on the spargers because it *results in bounding mass flows and temperatures experienced by the spargers. Additionally, with the IRWST water level below the spargers, the hydrodynamic loads associated with the initial discharge of air (trapped in the ADS valve discharge lines) or of the subsequent discharge of steam into the water are eliminated. Forces encountered by the IRWST and surrounding structures due to ADS actuation would not be large because the spargers contain a large number of small jets that would interact and dissipate over a relatively short distance. Based upon the considerations mentioned above and the equipment classification of the associated structures and components, the staff finds that ADS actuation is not inhibited by low IRWST level.*

The NRC staff reviewed the potential changes to containment holdup during floodup following a LOCA or ADS actuation as a result of the changes in LNP DEP 3.2-1. The NRC staff audited the "Containment Floodup Level" calculation (ADAMS Accession No. ML14219A200), and found that steam in the containment atmosphere and film on surfaces was accounted for. Applying the calculation for film condensing on surfaces used in RAI 7439, Question 6.03-3, results in a higher holdup than calculated in the supporting analysis in the form of film, which would reduce the containment level following depressurization of the RCS by less than 2 inches. Given the conservatisms inherent in the film holdup analysis in RAI 7439, Question 6.03-3, the staff finds no significant impact to containment floodup level as a result of LNP DEP 3.2-1.

An additional consideration is the availability of the ADS, IRWST injection, and containment recirculation valves during an extended station blackout event. The operator action to establish open loop cooling, if required, may occur at a time that exceeds the operating times for the ADS, IRWST injection, and containment recirculation valves specified in Table 3.11-1 of the FSAR. As part of the staff review of submittals from Southern Nuclear Operating Company (SNC) in response to "Order to Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events, Order EA-12-049," issued on March 12, 2012, for Vogtle Electric Generating Plant Units 3 and 4, which is licensed based on the same AP1000 certified design as the LNP Units 1 and 2 applicant, the NRC staff issued RAI 7741 and RAI 7756 to SNC seeking further justification that the AP1000 can transition to open loop cooling during an extended station blackout. SNC's response in letters dated December 4, 2014 (ADAMS Accession No. ML14338A658), and February 26, 2015 (ADAMS Accession No. ML15057A590), provided justification regarding (1) equipment qualification of the ADS, IRWST injection, and containment recirculation valves, and (2) diverse actuation capability for the squib valves.

SNC demonstrated the equipment qualification envelope for the ADS, IRWST injection, and containment recirculation valves is bounding for an event that utilized the PRHR HX long term. This was done by performing a best estimate calculation for the containment response to an event that utilized the PRHR HX over a 30-day duration. The pressure profile for the qualification envelope was shown to bound the results of the containment response calculation. The temperature profile from the containment response calculation was converted into an equivalent time at 150 °F (65.6 °C) using the Arrhenius method. This equivalent time is bounded by the qualification time specified for the ADS, *IRWST injection, and containment recirculation valves. The Arrhenius methodology has been previously reviewed and approved by the NRC staff for modeling the temperature effects in a post-LOCA environment (ADAMS Accession No. ML003701987). Based on the discussion in this paragraph, the NRC staff finds the equipment qualification envelope for the ADS, IRWST injection, and containment recirculation valves bounds the expected containment environment during an extended station blackout for at least 30 days.*

Additionally, SNC discussed the diverse capability for establishing open loop cooling. The primary means of establishing open loop cooling utilizes the Class 1E dc and uninterruptible power supply system (IDS). SNC's response included an analysis of the capacity of the IDS batteries. This analysis considered temperature de-rating of the batteries and self-discharge over a month and showed that sufficient margin is available for the batteries to perform their intended function during an extended station blackout. Should the battery supplies become completely exhausted, the ADS Stage 4, IRWST injection, and containment recirculation valves can be actuated via a diverse actuation system power independent device located at the secondary diverse actuation system station. Based upon the considerations in this paragraph, the NRC staff finds reasonable assurance that open loop cooling can be actuated during an extended station blackout event.

In a letter dated July 16, 2015 (ADAMS Accession No. ML15201A129), the applicant endorsed the RAI responses of SNC discussed above. Based upon the considerations of the environmental qualification of the ADS, IRWST injection, and containment recirculation valves, the containment floodup level, and the diverse actuation for establishing open loop cooling, the NRC staff finds that the safety injection function of the PXS is not impacted by LNP DEP 3.2-1.

B.1.2.4 Containment pH Control

Control of the pH in the containment sump post-accident is achieved through the use of pH adjustment baskets containing granulated trisodium phosphate (TSP) and is not impacted by LNP DEP 3.2-1.

B.1.2.5 Safe Shutdown

Short term safe shutdown conditions, defined in Section 7.4 of the DCD, include:

- *Maintaining the reactor in a subcritical condition*
- *Maintaining RCS average temperature less than or equal to no load temperature*
- *Retaining adequate coolant inventory*
- *Providing adequate core cooling*

Establishing short term safe shutdown conditions after an event has been demonstrated through DCD Chapter 15 analyses and reviewed by the staff in NUREG-1793. Through the evaluation of the PXS safety functions, the staff finds that short term safe shutdown is not impacted by LNP DEP 3.2-1.

Long term safe shutdown conditions, defined in Section 7.4 of the DCD, are the same as the short term conditions except that the RCS average temperature shall be less than 420 °F. The design requirement of entering a long term safe shutdown condition within 36 hours (i.e., reaching an average RCS temperature less than 420 °F in 36 hours) following an event is established in the URD and SECY-94-084. In Section 6.3 of the DCD, Revision 19, cooling the RCS to 420 °F in 36 hours is identified as part of the design basis for the PRHR HX. The ability of the PRHR HX to satisfy this design requirement is demonstrated in the shutdown temperature evaluation provided in DCD Section 19E.4.10.2.

The shutdown temperature evaluation utilizes the same model and evaluates the same event as discussed in subsection "Emergency Decay Heat Removal" of this SER. The analysis in Section 19E.4.10.2 differs in that several model inputs (e.g., containment response pressure, condensate return rate, initial power, and core decay heat) utilize more realistic values. Sections 6.3.3 and 7.4.1.1 of the revised FSAR refer to this analysis as "non-bounding, conservative." In order to better understand the sources of conservatism in the calculation, the NRC staff issued RAI 7475, Question 6.03-11. The response, provided in letter from the applicant dated June 27, 2014, identified conservatism inherent in the condensate return rate and several modeling choices that were taken to increase the heat load on the PRHR HX and limit the heat removal capability of the PRHR HX. The use of nominal and best-estimate values for reactor power and decay heat remains consistent with the shutdown temperature evaluation supporting the design certification as verified by the staff during an audit of the original calculation (ADAMS Accession No. ML14219A200). The results of the updated analysis demonstrate the RCS average temperature decreases below 420 °F within 36 hours. The staff performed confirmatory calculations as part of the review, which include a shutdown temperature evaluation. The result of the staff's confirmatory calculation for the shutdown temperature evaluation is consistent with the applicant's submittal. Based upon the considerations within this subsection, and the results of the bounding calculation discussed in subsection "Emergency Decay Heat Removal" of this SER, the staff finds the plant is consistent with SECY-94-084. The updated analysis is reflected in the applicant's proposed changes to FSAR Section 19E described in a letter from the applicant dated May 5, 2015.

In Revision 19 of the AP1000 DCD, the cooldown requirement of reaching an RCS temperature of 420 °F in 36 hours is the only performance criteria listed in Section 6.3.1.1.1 that is not demonstrated by a Chapter 15 analysis. In reading the original DCD, it would be possible to incorrectly conclude that this performance requirement was demonstrated by a Chapter 15 analysis. The applicant's proposed changes under FSAR Sections 6.3.1.1 in letters dated June 27, 2014, and July 24, 2014, clarify how this design requirement is demonstrated. Based upon considerations within this subsection, the staff finds the proposed FSAR revisions in Sections 6.3.1.1 and 19E, noted above, to be acceptable.

B.1.3 Non-Safety Design Basis

In the proposed FSAR revision under Section 6.3.1.2 the applicant states that the PRHR HX, in conjunction with the IRWST and the condensate return features of the PXS, has the capability to maintain the reactor coolant system in the specified, long-term shutdown condition for 14 days in a closed loop mode of operation. The 14-day operation is also reflected in the applicant's proposed changes under FSAR Section 19E. The basis for this duration is provided by extending the duration of the non-bounding conservative LOFTRAN calculation that was discussed in subsection "Safe Shutdown" of this SER. The staff verified the results of the analysis in an audit (see ADAMS Accession No. ML15187A248). In an update to the departure provided in a letter dated January 14, 2016 (ADAMS Accession No. ML16020A250), the applicant identified calculations incorporating ambient heat losses performed using RELAP 5, a transient analysis code, as LOFTRAN was not suited for demonstrating two-phase flow through the RCS. The RELAP calculations showed a loss of subcooling in the RCS occurring after 72 hours, but prior to 14 days. The calculations showed that the PRHR HX was capable of performing its function out to 14 days even with the loss of subcooling. The applicant provided test results from the APEX facility to demonstrate the ability of the PRHR HX to perform its function with a saturated RCS. The staff verified the results of the calculation and test results in an audit (ADAMS Accession No. ML16034A034). Operation of the PXS for 14 days in closed loop mode is not required to satisfy Commission regulations. The operational requirements of the PRHR HX have been evaluated in subsection "Safety Design Basis" of this SER. The staff finds the changes made to the operational duration and safety classification of the PRHR HX in LNP DEP 6.3-1 acceptable.

B.1.4 Post-72-Hour Actions

In DCD Section 6.3.4, it is stated that the only post-72-hour action required is a potential need for containment inventory makeup. This caused the staff to question the post-72-hour actions in the event that closed loop mode of PXS operation is extended following a non-LOCA event. In RAI-7440, Question 15.02.06-3, the staff requested clarification on post-72-hour actions following non-LOCA events. In a response dated June 19, 2014 (ADAMS Accession No. ML14171A453), the applicant stated that containment makeup would be necessary if containment leakage reduces the containment flood-up level, but there is no requirement to provide makeup to the IRWST to maintain PRHR HX operability. The primary post-72-hour actions are to provide water makeup to continue passive containment cooling and spent fuel cooling and, in the event that operators extend the closed loop mode of PXS operation, to provide power to the post-accident monitoring cabinets when transition to open loop cooling is required. In RAI 7440, Question 15.06.01, the NRC staff sought clarification on the criteria for operators to actuate ADS and transition to open loop cooling. The applicant's response provided in letter dated January 15, 2016 (ADAMS Accession No. ML16021A188), stated four criteria associated with reliable indication of core cooling which included (1) power availability to IDS divisions B and C, (2) hot leg and CMT level, (3) core exit thermocouple temperature, and (4) RCS pressure. The NRC staff finds this answer acceptable because it *requires operators to check for diverse and reliable indication of adequate core cooling. The impact of post-72-hour actions has been reviewed by the staff in subsection "Safety Design Bases" of this SER.*

B.2 Classification of Structures, Components, and Systems

Section 6.0, "Impacts to the Licensing Basis," of APP-GW-GLR-607 and APP-GW-GLR-161, Revision 2 describes the changes impacted to the COL application and provides the additional piping and components to the PXS. Subsection "Tier 1," states that "The added components of the PXS are integral to providing safety-related core decay heat removal during non-LOCA events. Therefore, it is appropriate to apply inspections, test, analyses and acceptance criteria to the added PXS components to provide reasonable assurance that the facility has been constructed and will be operated in conformity with the applicable design criteria, codes and standards." It further states that "As required by general design criterion 2 of Appendix A to 10 CFR Part 50, the PXS is designed to withstand the effects of natural phenomena and normal and accident conditions without loss of capability to perform its safety functions." The PXS containment recirculation downspout screens are identified as follows:

PXS-MY-Y81	*PXS-MY-Y85*
PXS-MY-Y82	*PXS-MY-Y86*
PXS-MY-Y83	*PXS-MY-Y87*
PXS-MY-Y84	*PXS-MY-Y88*

These component numbers will be added to the LNP Units 1 and 2 FSAR to supplement Table 2.2.3-1 of the AP1000 DCD, Revision 19, Tier 1. Mark-ups to Table 2.2.3-1 of the AP1000 DCD, Revision 19, Tier 1 and Table 3.2-3 of the AP1000 DCD, Revision 19, Tier 2, provided in Appendix B of APP-GW-GLR-607 and APP-GW-GLR-161, state that these eight additional downspout screens are not American Society of Mechanical Engineers (ASME) Code Section III components and the principal construction code is manufacturer standard.

In Section 6.0 of APP-GW-GLR-607 and APP-GW-GLR-161, under the subheadings "Tier 2," "Chapter 3: Impacted," the applicant states that, "The new PXS downspout screens are AP1000 Safety Class C and seismic Category I components. These components meet the quality assurance requirements of 10 CFR 50, Appendix B. Additionally, the screens must be demonstrated to have no functional damage following a seismic ground motion exceeding the one-third of the safe shutdown earthquake ground motion before resuming operations in accordance with 10 CFR Part 50, Appendix S." Under the subheading "Tier 1," the applicant further states that ITAAC design requirements will be met for these eight added downspout screens.

On the basis of the safety and seismic classifications of these eight added downspout screens, their quality assurance requirements, and the fact that SRP 3.2.1, "System Quality Group Classification," and Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," do not provide specific guidance for the code of construction for non-ASME, non-pressure retaining *components that belong to Quality Group C, the staff agrees that the use of manufacturer standards for the design of these downspout screens and the classification of AP1000 Safety Class C and seismic Category I is acceptable. Therefore, the staff finds the proposed FSAR revisions concerning these eight added downspout screens to be acceptable.*

Section 6.0 of APP-GW-GLR-607 and APP-GW-GLR-161, Subsection "Tier 1," states that "As required by general design criterion 4 of Appendix A to 10 CFR Part 50, the PXS containment downspout piping would be safety-related and required to withstand normal and seismic design basis loads without losing functional capability." The following PXS containment downspout piping are the proposed piping to be added to the LNP Units 1 and 2 FSAR to supplement Table 2.2.3-2 of AP1000 DCD, Revision 19, Tier 1:

PXS-L301A	*PXS-L306A*	*PXS-L301B*	*PXS-L306B*
PXS-L302A	*PXS-L307A*	*PXS-L302B*	*PXS-L307B*
PXS-L303A	*PXS-L308A*	*PXS-L303B*	*PXS-L308B*
PXS-L304A	*PXS-L309A*	*PXS-L304B*	*PXS-L309B*
PXS-L305A	*PXS-L310A*	*PXS-L305B*	*PXS-L310B*

Section 5.0, "Design Changes," Subsection "Polar Crane Girder and Internal Stiffener Modifications," Sub-subsection "1) PXS Downspout Piping," of APP-GW-GLR-607 and APP-GW-GLR-161 states that these added downspout piping are classified as AP1000 Safety Class C, seismic Category I. Mark-up of Table 2.2.3-2 to AP1000 DCD, Revision 19, Tier 1, provided in Appendix B of APP-GW-GLR-607 and APP-GW-GLR-161, further states that these added downspout piping are ASME Code Section III piping. According to the AP1000 DCD, Revision 19, Tier 2, Section 3.2.2, "AP1000 Classification System," Subsection 3.2.2.5, "Equipment Class C," Class C structures, systems and components are designed to codes and standards consistent with the guidelines for NRC Quality Group C. In addition, 10 CFR 50, Appendix B and ASME Code, Section III, Class 3 apply to pressure retaining components.

Section 6.0 of APP-GW-GLR-607 and APP-GW-GLR-161, Subsection "Tier 1," states that ITAAC design commitments will be met for these added downspout piping. In addition, Table 2.2.3-4 of the AP1000 DCD, Revision 19, Tier 1, provides ITAAC that 1) ensure the piping identified in Table 2.2.3-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements; 2) pressure boundary welds in piping identified in Table 2.2.3-2 as ASME Code Section III meet ASME Code Section III requirements; and 3) piping identified in Table 2.2.3-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.

On the bases that these downspout piping are designed to ASME Code Section III, Class 3 and the quality assurance requirements of 10 CFR 50, Appendix B, and that the ITAAC related to piping listed in Table 2.2.3-4 of the AP1000 DCD, Revision 19, Tier 1 apply, the staff finds the classification of this added downspout piping acceptable. Therefore, the staff finds the proposed FSAR revisions noted above to be acceptable.

B.3 Technical Specifications

In a letter dated February 7, 2014, the applicant submitted an exemption request titled "Supplement 3 to Submittal of Exemption Request and Design Change Description for Departure from AP1000 DCD Revision19 to Address Containment Condensate Return Cooling Design," for LNP Units 1 and 2. As a result of the condensate return testing conducted at the Waltz Mill Test Facility, modifications to the polar crane girder, internal stiffener, and IRWST gutter designs were made. In addition, extensions of the gutter were added above the upper personnel airlock and upper equipment hatch. A downspout system was also added to capture condensation at the polar crane girder and stiffener locations. These modifications result in minor editorial changes in a few sections of the TS and Bases (Chapter 16) in the COL application.

In a letter dated November 17, 2014, and titled "Supplement 5 to Submittal of Exemption Request and Design Change Description for Departure from AP1000 DCD Revision19 to Address Containment Condensate Return Cooling Design," the applicant provided further details on the condensate return issue including other editorial modifications to the TS and Bases.

These changes are necessary to ensure that the TS and Bases accurately reflect the updated design and are described below.

LCO Section of B3.3.3 (Postaccident Monitoring (PAM) Instrumentation)

On page B3.3.3-4, in the last line of the first paragraph in Section 11, "In-Containment Refueling Water Storage Tank (IRWST) Water Level," the text "…via a gutter." is updated to "…via a gutter and downspouts."

Background Section of B3.5.4 (Passive Residual Heat Removal Heat Exchanger (PRHR HX) – Operating)

On page B3.5.4-1, in the first and second lines of the third paragraph of the Background section, the text "…PRHR HX operation, a gutter is provided…" is updated to "…PRHR HX operation, downspouts and a gutter are provided…"

Also in that paragraph, the text in the fourth and fifth line is updated from "…collected by the gutter is directed…" to "...collected by the downspouts or gutter is directed…"

TS and SR Sections for B3.5.4.7

On page 3.5.4-3 of the TS, the text in SR 3.5.4.7 is updated from "…gutter is…" to "…gutter and downspout screens are…"

On page B3.5.4-7, the text in the first and second lines of the only paragraph in SR 3.5.4.7 is updated from "…IRWST gutters to verify…" to "…IRWST gutters and downspout screens to verify…"

Also in that paragraph, the text in the fourth and fifth lines is updated from "…the gutters could become restricted." to "…the gutter or downspout screens could become restricted."

The staff finds the proposed changes in both Supplement 3 and 5 acceptable because the changes make the TS and Bases consistent with the revised design. Therefore, the staff finds the proposed revisions noted above to be acceptable.

B.4 Risk Results and Insights

The proposed departure did not entail any change to the models used for plant-specific PRA. However, FSAR Table 19.59-202, "AP1000 PRA-Based Insights" item 1.e. was clarified to reflect how long the PRHR HX, IRWST, PCS, and condensate return features can now be relied on for core cooling.

The plant-specific PRA results and insights have been updated to account for this design change and departure. This is consistent with 10 CFR 52.79(d)(1) and is, therefore, acceptable to the staff.

21.1.5 Post Combined License Activities

There are no post-COL activities related to this section.

21.1.6 Conclusion

The NRC staff reviewed the WLS application and the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to the design change of the passive core cooling system, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the regulatory requirements and guidance discussed in Section 21.1.3 of this SER. The staff based its conclusion on the following:

- WLS DEP 6.3-1 and WLS DEP 3.2-1 are acceptable because the described changes permit the applicant to meet the licensing basis within the bounds of the updated licensing document.

21.2 <u>Main Control Room Dose Departure</u>

21.2.1 Introduction

At a meeting with the staff on July 23, 2014 (ADAMS Accession Nos. ML14220A110, ML14220A111, and ML14220A113), Westinghouse Electric Company, vendor for the AP1000 design, presented some self-identified discrepancies in underlying calculations supporting the AP1000 DCD, Revision 19, DBA MCR habitability dose analyses. Westinghouse identified the need to update the DBA analyses in order to show compliance with the control room habitability regulatory requirements in 10 CFR Part 50, Appendix A, GDC 19, "Control Room," because: (1) the analyses did not account for the MCR emergency ventilation system (VES) filter direct dose in the control room, (2) the nuclear island nonradioactive ventilation system (VBS) radiation monitor setpoints for control room ventilation system actuation did not account for all DBA release scenarios, and (3) the analyses that estimated the MCR dose contribution from direct radiation and skyshine used methodology that are not up-to-date. Subsequently, the staff issued RAI Letter No. 121, dated September 24, 2014 (ADAMS Accession No. ML14259A094), RAI 7661, to the LNP Units 1 and 2 COL applicant requesting them to address this information from the AP1000 design vendor.

21.2.2 Summary of Application

DEC incorporated in WLS COL application, dated April 11, 2016 (ADAMS Accession No. ML16124A854), the same information that DEF incorporated into LNP COL application related to the voluntary submittal of an exemption request and design change description for departure from the AP1000 DCD to address main control room dose. The information was originally submitted in endorsement and exemption request letter dated February 12, 2016 (ADAMS Accession No. ML16049A411).

Tier 1 and Tier 2 Departure

The applicant proposed the following Tier 1 and Tier 2 departure (DEP) from the AP1000 DCD, Revision 19:

- WLS DEP 6.4-1

In WLS DEP 6.4-1, the applicant included a departure from the AP1000 DCD, Tier 1 and Tier 2 information to reflect revised DBA dose analyses and design changes. As described in the letters referenced above, the proposed Tier 2 departure includes changes to FSAR Chapters 1, 3, 6, 7, 9, 11, 12, 14, and 15 in the WLS Units 1 and 2 COL application, as well as TS and TS Bases appearing in Part 4 of the COL application. In addition, the applicant requested an exemption from the incorporation by reference of AP1000 DCD Tier 1 information, specifically Tier 1 Section 2.7.1, to change the VES actuation signal name from “high-high” to “High-2” and to revise Tier 1 Section 2.2.5 and Tables 2.2.5-1 and 2.2.5-5 to add information on ITAAC for added shielding below the VES filter.

For the WLS DEP 6.4-1 revisions to FSAR Chapter 15 discussed above, the DBA dose analysis calculations that supported the DCD text are effectively replaced in full by site-specific DBA dose calculations that support departure WLS DEP 6.4-1. All seven of the DBA dose analyses documented in AP1000 DCD Chapter 15 are affected by at least one change to the analysis proposed in WLS DEP 6.4-1. The revisions to the DBA dose analyses affect both the MCR and offsite dose results.

This exemption request involves departures from Tier 1 Subsection 2.7.1 and the generic TS with other Tier 2 involved departures. Therefore, these departures require NRC approval and are evaluated below.

21.2.3 Regulatory Basis

The staff reviewed the departures related to the evaluation of control room habitability systems in accordance with NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” (SRP), Section 6.4, “Control Room Habitability

System." This guidance includes acceptance criteria that have been found acceptable by the staff for meeting the following control room habitability systems requirement:

- GDC 19, regarding providing a control room from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions

The staff used a dose criterion of 0.05 Sievert (Sv) (5 roentgen equivalent man (rem)) total effective dose equivalent (TEDE) for evaluating the control room radiological consequences resulting from DBAs, pursuant to GDC 19 of Appendix A to 10 CFR Part 50.

Because the proposed revisions to the DBA dose analyses affected the offsite dose results, the staff also evaluated the radiological consequences of DBAs against the dose criteria specified in 10 CFR 52.79(a)(1)(vi), of 0.25 Sv (25 rem) TEDE at the exclusion area boundary (EAB) for any 2-hour period, following the onset of the postulated fission product release, and 0.25 Sv (25 rem) TEDE at the outer boundary of the low population zone (LPZ) for the duration of exposure to the release cloud.

The staff used applicable guidance in SRP Section 6.4, "Control Room Habitability System," SRP Section 15.0.3, "Design Basis Accident Radiological Consequences Analyses for Advanced Light Water Reactors," and Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," in its review of the revised AP1000 DBA radiological consequence analyses.

21.2.4 Technical Evaluation

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff's findings on standard content that were documented in the SER for the reference COL application (LNP Units 1 and 2) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the LNP COL FSAR, Revision 9 to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.

- The staff verified that the site-specific differences were not relevant.

<u>Evaluation of Site Specific Content Related to Standard Content</u>

The pertinent site-specific information that affects the DBA dose analyses supporting WLS DEP 6.4-1 is the site characteristic short-term (accident) atmospheric dispersion factor (χ/Q) values. In LNP SER Section 21.2, the staff found that the revised DBA dose analyses were appropriately incorporated by reference in the LNP FSAR because the LNP site characteristic accident χ/Q values are less than the site parameter accident χ/Q values used in the revised DBA dose analyses in LNP DEP 6.4-1, which are the same values as used in the AP1000 DCD.

The WLS Units 1 and 2 site characteristic accident χ/Q values are different than the LNP site characteristic accident χ/Q values. However, the WLS site characteristic accident χ/Q values are unchanged by WLS DEP 6.4-1, and for each of the DBAs, the WLS site specific χ/Q values for each time averaging period are less than the comparable design reference χ/Q values used both in the AP1000 DCD and the revised DBA dose analyses provided in WLS DEP 6.4-1. Because the staff finds that the revised DBA dose analyses are appropriately incorporated by reference by comparison of the site characteristic accident χ/Qs to the values used in the revised DBA dose analyses, any site-specific differences in the values are not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting.

Tier 1 and Tier 2 Departure

- WLS DEP 6.4-1

The following portion of this technical evaluation section is reproduced from Section 21.2.4 of the LNP COL application FSER.

> - *LNP DEP 6.4-1*
>
> *LNP DEP 6.4-1 proposes to (1) revise the design description of the VBS to reflect the correct name of the actuation signal (high-high to High-2) for isolating the MCR penetrations, (2) reduce the allowable secondary coolant iodine activity to meet GDC 19 requirements for the main steam line break accident, and (3) address a number of other DCD changes based on issues that were identified through the design finalization process that challenge the ability of the AP1000 certified design to satisfy GDC 19.*
>
> *LNP DEP 6.4-1 also provides site-specific adoption of generic revisions to the AP1000 DBA dose analyses, including calculation of the MCR dose, and proposes a design change to add radiation shielding to the VES filter. Changes are made to each of the DBA dose analyses evaluated in Chapter 15 of the AP1000 DCD as referenced in the LNP Units 1 and 2 FSAR. Staff review of the specific changes will be discussed below in the technical evaluation of the departure.*
>
> *In addition, the staff reviewed a request for an exemption submitted by the applicant. The request proposed changes to Tier 1 Sections 2.2.5 and 2.7.1, Tier 1 Tables 2.2.5-1 and 2.2.5-4, and generic TS limiting condition for operation (LCO) 3.7.4 and surveillance requirement (SR) 3.7.4.1 and the related TS Bases in the AP1000 DCD. The regulatory evaluation of the exemption request appears in Subsection A, below, and the technical evaluation of the exemption request and departure appears in Subsection B, below.*
>
> *A. Regulatory Evaluation of Exemption Request*
>
> *A.1 Summary of Exemption*

The applicant requested an exemption from the provisions of 10 CFR Part 52, Appendix D, Section III.B, "Design Certification Rule for the AP1000 Design, Scope and Contents," that require the applicant referencing a certified design to incorporate by reference Tier 1 information.[3] Specifically, the applicant proposed to revise Tier 1 Section 2.2.5 and Tables 2.2.5-1 and 2.2.5-5 to add information on ITAAC related to the radiation shielding below the VES filter. Also, the applicant proposed to revise Tier 1 Section 2.7.1 to reflect a change to the name of the actuation signal for isolating the MCR penetrations and initiating the VES from "high-high" to "High-2". In addition, the applicant proposed a departure from the AP1000 generic TS, specifically TS LCO 3.7.4 and TS SR 3.7.4.1 to lower the allowable value for secondary coolant iodine activity concentration from 0.1 µCi/gm dose equivalent iodine-131 (DEI-131) to 0.01 µCi/gm DEI-131.

A.2 Regulations

- *10 CFR Part 52, Appendix D, Section VIII.A.4 states that exemptions from Tier 1 information are governed by the requirements of 10 CFR 52.63(b) and 10 CFR 52.98(f). It also states that the Commission may deny such a request if the design change causes a significant reduction in plant safety otherwise provided by the design. This subsection of Appendix D also provides that a design change requiring a Tier 1 change shall not result in a significant decrease in the level of safety otherwise provided by the design.*

- *10 CFR Part 52, Appendix D, Section VIII.C.4 states that an applicant may request an exemption from the generic TS or other operational requirements. The Commission may grant such a request only if it determines that the exemption will comply with the requirements of 10 CFR 52.7.*

- *10 CFR 52.63(b)(1) allows an applicant or licensee to request NRC approval for an exemption from one or more elements of the certification information. The Commission may only grant such a request if it complies with the requirements of 10 CFR 52.7 which in turn points to the requirements listed in 10 CFR 50.12 for specific exemptions, and if the special circumstances present outweigh the potential decrease in safety due to reduced standardization. Therefore, any exemption from the Tier 1 information certified by Appendix D to 10 CFR Part 52 must meet the requirements of 10 CFR 50.12, 52.7, and 52.63(b)(1).*

A.3 Evaluation of Exemption

[3] While the applicant describes the requested exemption as being from Section III.B of 10 CFR Part 52, Appendix D, the entirety of the exemption pertains to proposed departures from Tier 1 information and generic TS in the generic DCD. In the remainder of this evaluation, the NRC will refer to the exemption as an exemption from Tier 1 information and generic TS to match the language of Sections VIII.A.4 and VIII.C.4 of 10 CFR Part 52, Appendix D, which specifically govern the granting of exemptions from Tier 1 information and generic TS.

As stated in Section VIII.A.4 of Appendix D to 10 CFR Part 52, an exemption from Tier 1 information is governed by the requirements of 10 CFR 52.63(b)(1) and 52.98(f). Additionally, the Commission will deny an exemption request if it finds that the requested change to Tier 1 information will result in a significant decrease in safety. As required by 10 CFR 52.63(b)(1), the Commission may, upon application by an applicant or licensee referencing a certified design, grant exemptions from one or more elements of the certification information, so long as the criteria given in 10 CFR 50.12 are met and the special circumstances as defined by 10 CFR 50.12 outweigh any potential decrease in safety due to reduced standardization.

As stated in Section VIII.C.4 of Appendix D to 10 CFR Part 52, the Commission may grant an exemption from generic TS of the DCD only if it determines that the exemption will comply with the requirements of 10 CFR 52.7. As stated above, Section 52.7 points to 10 CFR 50.12 for specific exemptions.

Applicable criteria for when the Commission may grant the requested specific exemption are provided in 10 CFR 50.12(a)(1) and (a)(2). Section 50.12(a)(1) provides that the requested exemption must be authorized by law, not present an undue risk to the public health and safety, and be consistent with the common defense and security. The provisions of 10 CFR 50.12(a)(2) list six special circumstances for which an exemption may be granted. It is necessary for one of these special circumstances to be present in order for NRC to consider granting an exemption request. The applicant stated that the requested exemption meets the special circumstances of 10 CFR 50.12(a)(2)(ii). That subsection defines special circumstances as when "[a]pplication of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule." The staff's analysis of each of these findings is presented below.

A.3.1 Authorized by Law

This exemption would allow the applicant to implement approved changes to Tier 1 Sections 2.2.5 and 2.7.1, Tier 1 Tables 2.2.5-1 and 2.2.5-5 and generic TS LCO 3.7.4 and SR 3.7.4.1. This is a permanent exemption limited in scope to particular Tier 1 information and generic TS, and subsequent changes to this information or any other Tier 1 information or generic TS would be subject to full compliance with the change processes specified in Sections VIII.A.4 and VIII.C.4 of Appendix D to 10 CFR Part 52. As stated above, 10 CFR 52.63(b)(1) allows the NRC to grant exemptions from one or more elements of the certification information, namely, as discussed in this exemption evaluation, the requirements of Tier 1. Moreover, Section VIII.C.4 allows the NRC to grant exemptions from generic TS if the exemption meets the requirements of 10 CFR 52.7 and 50.12. The staff has determined that granting of the applicant's proposed exemption will not result in a violation of the Atomic Energy Act of 1954, as amended, or the NRC's regulations. Therefore, as required by 10 CFR 50.12(a)(1), the exemption is authorized by law.

A.3.2 No Undue Risk to Public Health and Safety

The underlying purpose of AP1000 Tier 1 Sections 2.2.5, 2.7.1, Tier 1 Tables 2.2.5-1 and 2.2.5-5 and generic TS LCO 3.7.4 and SR 3.7.4.1 is to ensure that the plant will be constructed and operated with appropriate protection of the public health and safety and provide radiation protection to workers in the event of an accident, including radiation shielding and limitation of radioactive material that could be released to the environment.

Addition of radiation shielding below the VES filter improves worker protection from the effects of radiation and ensures that the control room operators can occupy the control room in order to take actions to maintain the plant in a safe condition during accident conditions; this change, therefore, supports the system's intended design functions. Reducing the allowable iodine activity concentration in the secondary coolant limits the amount of radioactive material that is available for release to the environment during accidents and, therefore, reduces the potential dose to the public from accidents to meet the offsite dose criteria for the plant siting and safety assessment. Changing the name of the VES actuation signal for isolating the MCR penetrations in Tier 1, Section 2.7.1, ensures consistency with Tier 2 design information and does not change the function of the actuation signal.

The plant-specific Tier 1 DCD and TS will continue to meet regulatory requirements for protecting public health and safety and will maintain a level of detail consistent with that which is currently provided elsewhere in Tier 1 of the plant-specific DCD. The affected design description in the plant-specific Tier 1 DCD will continue to provide the detail necessary to support the performance of the associated ITAAC. The proposed changes to Tier 1 information and generic TS are evaluated and found to be acceptable in Section 21.2.B of this safety evaluation. Therefore, the staff finds the exemption presents no undue risk to public health and safety as required by 10 CFR 50.12(a)(1).

A.3.3 Consistent with Common Defense and Security

The proposed exemption would allow the applicant to implement modifications to the Tier 1 information and generic TS requested in the applicant's submittal. This is a permanent exemption limited in scope to particular Tier 1 information and a specific TS. Subsequent changes to this information or any other Tier 1 information or generic TS would be subject to full compliance with the change processes specified in Sections VIII.A.4 and VIII.C.4 of Appendix D to 10 CFR Part 52. This change is not related to security issues. Therefore, as required by 10 CFR 50.12(a)(1), the staff finds that the exemption is consistent with the common defense and security.

A.3.4 Special Circumstances

Special circumstances, in accordance with 10 CFR 50.12(a)(2)(ii), are present whenever application of the regulation in the particular circumstances would not serve the underlying purposes of the rule or is not necessary to achieve the underlying purpose of the rule. The underlying purpose of the specific Tier 1 Tables 2.2.5-1 and 2.2.5-5 and TS LCO 3.7.4 and SR 3.7.4.1 being modified in *the exemption request is to identify and conduct surveillances of the components that will be added to the design of the VES and also the control of radioactive material in the secondary coolant. The additional components and new surveillance requirements for those components are needed so that the MCR can perform its intended functions, that is, to (1) provide a control room from which actions can be taken to operate the nuclear power unit safely under normal conditions, (2) maintain the nuclear power unit in a safe condition under accident conditions, with adequate radiation protection, and (3) permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposure in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident, in accordance with GDC 19. The proposed change to the VES actuation signal name in Tier 1 Section 2.7.1 does not affect the design function of the VBS to isolate the MCR penetrations and ensures consistency with Tier 2 design information.*

Using the "high-high" name for the VES actuation signal in Tier 1, Section 2.7.1, and application of the requirements in Tier 1, Tables 2.2.5-1 and 2.2.5-5 (related to the VBS and VES design description and ITAAC) and generic TS LCO 3.7.4 and SR 3.7.4.1 (related to the specific activity limit in the secondary coolant), as was previously approved for the AP1000 design certification, is not necessary to achieve the underlying purpose of those portions of the rule, given that the departures proposed by the applicant improve consistency with Tier 2 design information and improve the function of systems designed to limit doses to workers and the public . The proposed additions to the VES filter shielding supports the MCR's intended design functions, as does the addition of ITAAC for those additional components. Likewise, the changes to the allowable iodine activity concentration in the secondary coolant supports the MCR's intended design function and compliance with the siting and safety assessment offsite dose requirements. Reducing the TS limit for DEI-131 improves accident consequence margins for DBAs involving secondary coolant release. These changes do not affect the ability of any structures, systems, or components to perform their functions or impair safety and, therefore, meet the underlying purposes of the rule. Accordingly, because application of the requirements in Tier 1 Tables 2.2.5-1 and 2.2.5-5 and the generic TS LCO 3.7.4 and SR 3.7.4.1 is not necessary to achieve the underlying purpose of the rule, special circumstances are present. Therefore, the staff finds that special circumstances required by 10 CFR 50.12(a)(2)(ii) for the granting of an exemption from the Tier 1 information and generic TS described above are present.

A.3.5 Special Circumstances Outweigh Reduced Standardization

This exemption, if granted, would allow the applicant to change certain Tier 1 information incorporated by reference from the AP1000 DCD into the LNP COL application. An exemption from Tier 1 information may only be granted if the special circumstances of the exemption request, required to be present under 10 CFR 52.7 and 10 CFR 50.12, outweigh any reduction in standardization. The proposed exemption would add shielding under the VES filter and change the name of the VES actuation signal that isolates the MCR. The proposed changes *to the VES filter shielding and VES actuation signal name support and maintain the MCR's intended design functions.[4]*

As described below in the technical evaluation, the changes to the VES filter shielding and the name of the VES actuation signal ensure the capability of the safety related VES to maintain habitability in the control room during accidents, as described in DCD Chapter 6.4 "Control Room Habitability Systems," and meet the dose limit requirements of GDC 19. Consequently, although there is a small possibility that standardization may be slightly reduced by the granting the exemption from the specified Tier 1 requirements, the proposed exemption adding shielding to the VES filter will improve the reliability and effectiveness of the MCR and associated heating, ventilation, and air conditioning (HVAC) systems, to better allow the MCR and the VES to perform their intended functions with respect to radiological habitability. For this reason, the staff determined that even if other AP1000 licensees and applicants do not request similar departures, the special circumstances supporting this exemption outweigh the potential decrease in safety due to reduced standardization of the AP1000 design, as required by 10 CFR 52.63(b)(1).

A.3.6 No Significant Reduction in Safety

The proposed exemption would add shielding under the VES filter and change the name of the VES actuation signal. As described below in the technical evaluation, these changes (1) ensure the design functions for the VES and the MCR are maintained, (2) ensure consistency with Tier 2 design descriptions, and (3) ensure that the requirements of GDC 19 are met for all DBAs. The proposed changes to the VES filter shielding design will maintain the MCR's key design functions and will not impair the function of the VES or the MCR. The proposed change to the VES actuation signal name does not affect the function of the VBS or VES, and, therefore, does not affect the function of the MCR. Because the proposed changes will ensure that the design functions for the VES and MCR are maintained and that the requirements of GDC 19 are met for all DBAs, there is no reduction in safety. Therefore, the staff finds that granting the exemption would not result in a significant decrease in the level of safety otherwise provided by the design, as required by 10 CFR Part 52, Appendix D, Section VIII.A.4.

A.4 Conclusion

The staff has determined that pursuant to Section VIII.A.4 of Appendix D to 10 CFR Part 52, the exemption: (1) is authorized by law, (2) presents no undue risk to the public health and safety, (3) is consistent with the common defense and security, (4) has special circumstances that outweigh the potential decrease in safety due to reduced standardization, and (5) does not significantly reduce the level of safety at the applicant's facility. The staff has also determined, pursuant to Section VIII.C.4 of Appendix D to 10 CFR Part 52 , that the generic TS portion of the exemption request: (1) is authorized by law, (2) presents no

[4] Based on the nature of the proposed changes to the plant-specific Tier 1 information in Sections 2.2.5 and 2.7.1, other AP1000 licensees and applicants may request the same exemption, preserving the intended level of standardization.

undue risk to the public health and safety, (3) is consistent with the common defense and security, and (4) demonstrates the existence of special circumstances. Therefore, the staff grants the applicant an exemption from the requirements of Tier 1 Sections 2.2.5 and 2.7.1, Tables 2.2.5-1 and 2.2.5-5 and generic TS LCO 3.7.4 and generic TS SR 3.7.4.

B. Technical Evaluation of Exemption Request and Departure

As summarized above in Section 21.2.2 of this safety evaluation, the applicant proposed LNP DEP 6.4-1 to depart from the AP1000 DCD. The applicant's departure is based on new DBA radiological consequence analyses instead of the generic site analyses that AP1000 DCD Chapter 15 is based on. The remainder of the analysis assumptions, inputs, and methodologies are the same as given in AP1000 DCD that the staff previously evaluated and found acceptable in NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," Initial Report, Section 15.3.

In addition to review of the departure information submitted by letter and incorporated into the FSAR and Parts 2, 4, 7, 9, and 10 of the COL application, the staff performed an audit of the applicant's proprietary calculation packages and had the opportunity during public meetings to discuss the contents of both the submittals and the audited calculations (ADAMS Accession No. ML15231A003). During the audit, the staff verified that the changes to the DBA dose analyses presented in LNP DEP 6.4-1 and reflected in the provided markups of DCD were included in the supporting DBA dose analysis proprietary calculation packages and that the calculations did not contain additional changes not reflected in LNP DEP 6.4-1. The staff's review of the proposed design changes and revisions to the DBA radiological consequences analyses, including calculation of the MCR dose, is discussed below in this section.

DBAs analyzed for radiological consequences and the corresponding AP1000 DCD sections where the radiological consequences analyses for those DBAs are discussed are given below.

DCD Section	*Design Basis Accident*
15.1.5.4	*Main Steam Line Break (MSLB)*
15.3.3.3	*Reactor Coolant Pump Shaft Seizure (Locked Rotor, LRA)*
15.4.8.3	*Control Rod Ejection Accident (REA)*
15.6.2	*Small Line Break*
15.6.3.3	*Steam Generator Tube Rupture (SGTR)*
15.6.5.3	*Loss of Coolant Accident (LOCA)*
15.7.4.3	*Fuel Handling Accident (FHA)*

B.1 MCR direct dose analysis revisions

At a public meeting with the staff on July 23, 2014, Westinghouse Electric Company presented information about some self-identified discrepancies in underlying calculations supporting the AP1000 DCD DBA MCR habitability dose analyses. Westinghouse identified the need to update the analyses in order to show compliance with GDC 19 because the analyses did not account for the MCR VES filter direct dose in the control room, and the MCR dose contribution from direct radiation and skyshine calculations used a methodology that was not up-to-date. Following this meeting, on September 24, 2014, the staff issued RAI Letter No. 121, RAI 7661 (ADAMS Accession No. ML14259A106). Section 1c of Question 06.04-2 of this RAI specifically asked for additional information regarding intended revisions to the MCR direct radiation and skyshine dose calculations.

At a public meeting held on February 26, 2015, the applicant for the LNP Units 1 and 2 COL presented information on the approaches to address three departures from the AP1000 DCD: estimated dose to MCR operators, MCR heatup, and hydrogen vent location ITAAC (ADAMS Accession No. ML15056A091). The purpose of the meeting was to discuss ways for resolving the issues identified in the July 2014 meeting, including RAI 7661, and to discuss the path for conducting the relevant staff reviews. In this meeting, the applicant indicated that it was changing the methods for calculating direct radiation and skyshine doses to MCR operators from those used in AP1000 DCD.

Information contained in Tier 2 Sections 6.4, 9.4.1, and 11.5, of the AP1000 DCD Tier 2 describes how the two ventilation systems operate during normal and accident conditions. In summary, the VBS system, provides heating, cooling, and air exchange during normal operation. The fans, controls, and air conditioning equipment receive power from non-safety-related alternating current sources. Radiation monitors are located in the outside air inlets to the VBS system. When the safety-related radiation monitors detect a release of radioactive material, non-safety-related signals activate controls to realign non-safety-related dampers that direct airflow through charcoal and high-efficiency particulate air (HEPA) filters. These actions help reduce the amount of activity added to the MCR air and act to reduce the amount of activity already present. If inlet radioactivity levels continue to rise, a safety-related signal (High-2) from the radiation monitors actuates safety-related controls that isolate the MCR from the VBS system and actuate the safety-related VES ventilation system. The VES system uses high-pressure air from compressed air bottles to supply make-up air to the MCR. The air flows through an eductor that recirculates air in the MCR through safety-related HEPA and charcoal filters. The operation of the safety-related radiation monitors, VBS dampers, and VES actuation on a High-2 signal serve to maintain MCR operator doses less than the dose criterion of GDC 19 during accidents.

The applicant's VBS analysis supporting LNP DEP 6.4-1 assumed that the VES system did not actuate when the safety-related High-2 signal actuated. The applicant's supporting calculation for the total dose resulting from exclusive use *of the VBS system without transitioning to the VES system is conservative and unnecessary for the staff to reach a safety finding.*

On February 24, 2015, the staff began auditing MCR-dose-related calculation packages. The packages reviewed indicated that the direct dose contribution for some portions of the MCR dose analysis were performed using the Monte Carlo N-Particle (MCNP) radiation-transport code, Version 5, developed by Los Alamos National Laboratory. The calculation packages initially reviewed by the staff did not contain listings of the MCNP input or output files used for these calculations. Information provided in the calculation packages indicated that in one area of the plant located adjacent to the MCR, the design used a flexible radiation shielding material to reduce post-loss-of-coolant accident (LOCA) dose rates from Zone IX to Zone VIII. Radiation Zones are defined in AP1000 DCD, Tier 2 Chapter 12, "Radiation Protection," Section 12.3 "Radiation Protection Design Features," of the AP1000 DCD (ADAMS Accession No. ML11171A354), Figure 12.3-2 (Sheet 1 of 16,) "Radiation Zones, Post-Accident Legend." Zone VIII is defined as greater than 100 rem/hr (1 Sv/hr) and less than or equal to 500 rem/hr (5 Sv/hr), and Zone IX as greater than 500 rem/hr (5 Sv/hr). Other portions of the calculation packages indicated that no shielding material is included in penetration models between the Shield Building wall opening and piping or electrical cabling passing through penetrations.

The June 5, 2015, response to RAI 7661 contained in Enclosure 1 to NPD-NRC-2015-014 (ADAMS Accession No. ML15161A042), stated that site-specific revisions for direct radiation and skyshine dose would be included in the LNP COL application. These revisions would include updated direct radiation and skyshine dose calculations to account for MCR penetrations shielding differences between the AP1000 and AP600 designs. In the AP1000 DCD, dose contributions from adjacent structure direct and skyshine radiation included in the MCR operator dose results for LOCA are based upon AP600 post-accident dose calculations and assume the presence of shielding that was not included in the AP1000 design. In LNP DEP 6.4-1, the applicant revised the post-accident radiological dose calculations to use updated AP1000 detailed design inputs and analyses for skyshine and direct radiation.

The information gathered by the staff during audits and the applicant's June 5, 2015, response to RAI 7661 led the staff to issue RAI Letter No. 130, RAI 8028, on August 7, 2015. RAI 8028 contained Questions 12.03-2 through 12.03-9, seeking additional information and clarification regarding the methods, models, and assumptions used to determine the direct and skyshine dose to the MCR operators. The applicant provided the initial response to this RAI in NPD-NRC-2015-042, dated November 2, 2015.

The calculation packages reviewed by the staff indicated that all penetrations greater than 6 inches in diameter were included in the applicant's MCNP model. The calculation packages further stated that contributions from penetrations less than 6 inches in diameter were not included in the MCNP model, but their contribution to the MCR dose was analyzed. The analysis of the contribution to MCR dose from penetrations less than 6 inches in diameter was not included in the set of initial documents reviewed by the staff.

It was not clear to the staff how the AP1000 design ensured that the contribution of direct radiation streaming through penetrations in the MCR envelope shield walls would result in MCR operator doses less than the requirements of GDC 19. In RAI 8028 Question 12.03-2, the staff asked the applicant to: (1) identify penetrations to the MCR shielding boundary, (2) identify the radiation protection design features credited for attenuating streaming radiation into the MCR, and (3) describe the direct radiation dose contribution to the MCR operators from MCR shielding penetrations. The applicant's response stated that Westinghouse had evaluated the control room layout and designed openings to identify penetrations with significant implications for radiation streaming. These penetrations were included in the MCNP model. The applicant excluded smaller penetrations from the model because ". . . previous analyses and informal work (using the Rockwell equations) showing streaming contributions through small penetrations is expected to be insignificant." "Reactor Shielding Design Manual," Editor Theodore Rockwell III, McGraw-Hill Book Company, Inc., 1956, available as TID-7004, Chapter 8, "Effects of Irregularities in Shields," Section 3, "Gammas," describes the referenced Rockwell equations. Using the referenced Rockwell equations, some penetration sizes representative of those portrayed in the RAI response, and the dose rates referred to in AP1000 DCD, Tier 2 Section 12.3, Figure 12.3-2, the staff performed some scoping calculations to ascertain the potential impact from penetrations on MCR operator dose. Because the Rockwell equations are not directly applicable to the radiation and shielding environment surrounding the MCR shielding envelope, the staff also performed an MCNP-based scoping analysis representing a penetration into the MCR at a right angle to the incident radiation. The analysis performed by the staff indicated that a potential existed for exceeding the requirements of GDC 19 to some MCR operators due to radiation streaming through penetrations under the conditions analyzed in the DCD.

From the audit reviews conducted, it was not clear to the staff how the AP1000 design used flexible shielding material to prevent radiation streaming through penetrations into areas located adjacent to the MCR envelope. The staff was concerned because the environmental conditions of some of the locations where this material was located could exceed the design characteristics of the shielding material. It was not clear to the staff to what extent the AP1000 MCR shielding design relied on the use of a flexible shielding material to maintain MCR operator doses less than the requirements of GDC 19. In RAI 8028 Questions 12.03-3 and 12.03-4, the staff asked the applicant to: (1) describe where radiation protection design features such as penetration sealants are credited for attenuating direct radiation entering the MCR, and (2) identify those locations where environmental conditions could limit the serviceability of radiation protection design features such as penetration sealants that are credited for attenuating direct radiation entering the MCR. The applicant's response dated November 2, 2015, acknowledged that there were inconsistencies in the calculation packages regarding crediting the use of flexible shielding material for the MCR dose calculations. The response stated that the MCR dose provided in Enclosure 1 to NPD-NRC-2015-014 and currently certified post-accident radiation zone results do not require penetration sealant materials to be credited, and that the associated dose calculation packages were being revised to clarify *this position. Because flexible shielding material is not credited in the MCR post-accident dose analysis used to demonstrate compliance with GDC 19, the staff finds this response acceptable.*

NPD-NRC-2015-027 Enclosure 3, Figure 9.4.1-1 (Sheet 5 of 7), "Nuclear Island Non-Radioactive Ventilation System," shows the particulate, iodine, and noble gas airborne radiation monitor sample points upstream of the isolation valves V186 and V187. AP1000 DCD, Tier 2 Figure 7.2-1, Sheet 13 of 21, "Functional Diagram Containment and Other Protection," shows that the MCR radiation monitors are de-energized and the MCR isolation is actuated on either a High-2 radiation signal or a low battery charger input voltage for greater than 10 minutes. DCD Tier 2 Tables 8.3.2-1 through 8.3.2-4, describing 250V dc Class 1E divisional battery nominal load requirements, do not show any MCR airborne activity radiation monitors or MCR area radiation monitors, nor does it indicate any provisions for power to supply portable airborne activity monitoring equipment. Therefore, in RAI 8028 Question 12.03-7, the staff asked how the applicant would perform the surveys required by 10 CFR 20.1501 needed to ensure that the MCR filtration system was maintaining MCR dose less than the requirements of GDC 19 during post-accident conditions. The applicant's response stated that results of manual surveys are not credited as part of the AP1000 design. Such actions and the scope for the surveys mentioned in this question would likely fall within an Emergency Planning and Response Program. In addition, the applicant stated that grab samples could be taken using battery-operated equipment or a supply of ac power from a battery-backed control room outlet could be temporarily diverted to sampling equipment to obtain a grab sample of the MCR atmosphere. Because of the limited duration of sampling and the minimal heat load provided by this type of equipment, such activities are expected to have an insignificant impact on temperatures in the MCR. The samples would be analyzed in laboratory space located outside of the MCR envelope. Because this response meets the requirements of 10 CFR 20.1501 for performing surveys, the staff finds this response acceptable.

During the audit reviews, the staff identified a number of individually minor differences between information contained within design basis documents, such as the density of concrete specified in DCD, discussions provided in calculation packages and the MCNP input/output files used to calculate MCR dose. Also, AP1000 DCD Tier 1 Table 3.3-1 "Definition of Wall Thicknesses for Nuclear Island Buildings, Turbine Building, and Annex Building," Footnote 2, states that the wall thicknesses have a tolerance of plus or minus 1 inch. The staff determined that the MCNP input/output files (proprietary) provided by the applicant used to calculate MCR dose calculations specified the nominal wall thicknesses instead of the minimum allowable wall thicknesses (ADAMS Accession Nos. ML15132A101 and ML15148A574). Using Grove Software, MicroShield Version 9.06 and MCNP6, the staff performed some scoping calculations to ascertain the potential effect on MCR operator dose. Based on the results of these calculations, it was not clear to the staff that the AP1000 design ensured that MCR operator doses would be maintained less than the requirements of GDC 19. Therefore, in RAI 8028 Questions 12.03-8 and 12.03-9, the staff asked the applicant to provide sufficient information to demonstrate that the shielding provided for MCR operators would be sufficient to maintain

MCR operator doses within the limits of GDC 19, under the conditions analyzed in the DCD. The applicant's response stated that the AP1000 DCD specified the use of the Westinghouse Quality Program to define how the company meets customer and regulatory requirements. This program was designed to meet the quality requirements of the U.S. nuclear industry including 10 CFR Part 50 Appendix B and ASME NQA-1. Westinghouse procedures control the use of external computer software applied in safety-related design applications (in this case, the MCNP5 software) acquired from Non-Qualified Suppliers. The inputs to the MCNP5 code were made in accordance with the high-level Westinghouse Policies and Procedures, and the related configuration control procedures in place for design analysis applications. The applicant and Westinghouse further noted that information regarding shield walls and dimensions are noted in Tier 1, Table 3.3-1, of the licensing basis, and that the ITAAC text that introduces this table (Tier 1, Section 3.3, Item 3) states that this information is for "shielding during normal operations." Therefore, information in this table is not indicative of methods and inputs used in post-accident radiation shielding calculations and is not intended to be used for post-accident MCR operator dose calculations. The applicant and Westinghouse also stated that other conservative assumptions, such as source term assumptions, elemental make up, and concrete density during construction versus concrete density specified within the MCNP input files, provided sufficient margin to ensure that MCR dose remained within the GDC 19 dose criterion.

Following staff scoping calculations performed to evaluate the effects on MCR dose from MCR shield wall penetrations and changes in shielding thicknesses and densities, and technical discussions with the applicant during the audit, the applicant made available for audit additional information about MCR penetrations. After reviewing the additional information, the staff continued audit discussions with the applicant and Westinghouse shielding design technical experts. The applicant agreed to provide additional information about: (1) some additional specific penetrations that were being evaluated, (2) treatment of penetrations and embedded piping running through floor shielding, (3) relative value of assumed conservatisms, and (4) a discussion of conservative assumptions that would balance against non-conservatisms (ADAMS Accession No. ML16020A355).

The applicant submitted additional information to address these concerns in NPD-NRC-2016-010, dated February 9, 2016 (ADAMS Accession No. ML16042A081). As stated above, in RAI 8028 Question 12.03-2, the staff asked the applicant to provide information about potential dose to MCR operators due to radiation streaming through penetrations in the MCR shield wall envelope. The supplemental response contained in NPD-NRC-2016-010 described a sensitivity study used to ascertain the total effect of all existing penetrations included in the MCNP model to the calculated MCR operator dose. The applicant's supplemental response provided additional information to address the staff's concerns. The response stated that these studies showed that the dose resulting from penetrations was a small fraction of the total direct dose to the MCR operators. The response compared the existing modeled penetrations to the penetrations identified during the staff review. Most of the extra penetrations identified by the staff were similar in size and location to already modeled *penetrations, so any incremental increase in dose from those penetrations should be small. The response provided information showing that in several cases, such as for horizontal runs of piping through shielding material, the actual dose rates within the areas adjacent to the location of the lines were only a fraction of the maximum dose rate listed for the zone.*

The staff also used the response to assess treatment of penetrations and embedded piping running through floor shielding. The information contained in DCD Tier 2 Figure 3H.5-9, Sheet 2 of 3,) "Auxiliary Building Finned Floor," showing the steel plate referenced in the response, in conjunction with the note on Figure 3H.5-9 stating that staff approval is required prior to implementing a change to Figure 3H.5-9, provided confirmation to the staff that other structural components not credited in the MCNP calculations were present in the design. The staff used MicroShield scoping calculations to assess the relative attenuation of an air-filled void horizontal drain system pipe combined with the additional steel plate not credited in the applicant's MCNP calculation to a solid concrete floor without the void and steel plate. The attenuation provided by the void and steel plate appeared to be less than a solid concrete floor. However, by using the information provided in the supplemental response about the localized dose rates in the adjacent rooms, the conservatisms used in the model for the operation of the VBS system, and the directional nature of the radiation in the adjacent rooms, the staff ascertained that any incremental increase in MCR dose resulting from the embedded pipe would be insignificant.

The information in supplemental response NPD-NRC-2016-010 also addressed the potential contribution to MCR dose from some staff-identified penetrations in the MCR shield wall into an area of the plant next to the Shield Building. This area contains large penetrations through the Shield Building wall which can result in radiation streaming. The response noted that the radiation zoning for the room is due to the radiation levels next to the Shield Building penetrations. Because of the location of the penetrations in the MCR wall with respect to the Shield Building penetrations, the dose rates near the MCR wall penetrations would be significantly lower than the maximum dose rate associated with the zone designation of the room. The response also noted that because of the directional nature of the radiation streaming through the MCR wall penetrations and the location of the dose receptor point of interest inside of the MCR area, further attenuation would occur. Staff-based MCNP6 scoping calculations to assess the magnitude of the expected attenuation were consistent with the information provided in the supplemental response.

The supplemental response contained in NPD-NRC-2016-010, also addressed the staff request to have information demonstrating an understanding of the full extent of penetrations through the MCR shield wall envelope. To help quantify direct dose to operators in the MCR from the existing AP1000 control room penetrations, Westinghouse stated that, based on their analysis, the contribution from the existing penetrations was a small fraction of the total direct dose to the MCR operators. Westinghouse stated that they reviewed archived concrete drawings, reviewed archived penetration drawings, and reviewed completed design change packages, to ensure that that the full scope of penetrations were identified and considered. Through reviews of the AP1000 plant *three-dimensional software model, they verified that all penetrations into radiologically significant areas were identified.*

Because the information provided in the supplemental response contained in NPD-NRC-2016-010 shows that the contribution to MCR operator dose from penetrations through the MCR shielding envelope would not result in exceeding the operator dose requirements of GDC 19, under the conditions analyzed in the DCD, the staff considers the issue identified in RAI 8028 Question 12.03-2 resolved.

As stated above in RAI 8028 Questions 12.03-8 and 12.03-9, the staff asked the applicant to provide sufficient information to demonstrate that the shielding provided for MCR operators would be sufficient to maintain MCR operator doses within the limits of GDC 19. The supplemental response contained in NPD-NRC-2016-010 discussed materials and construction details of the Shield Building wall that were not echoed in the applicant's/Westinghouse's MCNP shielding model. The staff also performed some scoping calculations using MCNP6 to evaluate the relative effectiveness of regular concrete versus regular concrete with embedded rebar. The staff scoping calculations showed that the degree of radiation attenuation is sensitive to variations in the location, size, or distribution of the rebar material. The level of detail in the DCD regarding location of rebar within walls and rebar size used in various walls of the plant does not support the staff performing a reliable evaluation of the relative attenuation effectiveness for generic walls.

To address the staff concerns related to the shielding design assumptions, the applicant provided a description of the conservatisms present in other portions of the MCR dose calculation, to show that any realistic non-conservatisms in the shielding design assumptions were well exceeded by the conservatisms present in the airborne activity dose calculations. In the supplemental response contained in NPD-NRC-2016-010, the applicant quantitatively discussed the relative significance of operation of the VBS system below the safety-related High-2 setpoint that would result in the transition from the non-safety-related VBS system to the safety-related VES system. The calculation used by the applicant estimated the total dose resulting from exclusive use of the VBS system without transitioning to the safety-related VES system, even though the VBS inlet airborne radioactivity concentrations would exceed the High-2 setpoints. Because the calculation assumes the non-safety related VBS system continues to operate with inlet airborne radioactivity levels above the safety related High-2 setpoint (the threshold at which the safety-related VES system actuates), this results in over estimating MCR operator dose because of airborne activity concentrations within the MCR. This is a very conservative approach, and unnecessary for the staff to reach a safety finding. As a result, a large margin exists between the 0.05 Sv (5 rem) TEDE criterion used for evaluating the VBS system performance and the total dose estimate derived from operating the VBS system below the High-2 setpoint. Because this margin ensures that the potential additional contribution to MCR operator dose resulting from the use of minimum wall thicknesses would not result in exceeding the operator dose requirements of GDC 19, under the conditions analyzed in the DCD, the staff *considers the issue identified in RAI 8028 Question 12.03-8 and 12.03-9 to be resolved.*

B.2 Control room filter direct dose

In its initial response to RAI 7661, dated February 6, 2015, the applicant identified that radiation contributions from MCR HVAC filters were not considered in the MCR dose analyses reported in the AP1000 DCD, Chapters 6.4 and 15. The applicant's revised DBA dose analyses include the contribution to the total MCR operator dose due to direct radiation from radioactive material estimated to accumulate on the VES and VBS filters during the accident.

The staff reviewed applicant-provided information about the direct dose from the VES and VBS filters. Because the VBS filter is located outside of the MCR envelope shielding boundary, the direct radiation dose from the VES filter is more limiting than the direct radiation dose from the VBS filter. Based on this consideration, the staff developed a scoping model using MCNP6 for the VES filter. The scoping model developed by the staff did not indicate the presence of any significant differences between the staff approach and that evidenced in the applicant's MCNP input and output files for the VES and VBS reviewed by the staff. The applicant's submittal dated July 1, 2015, states that shielding of the VES filtration unit is accomplished by safety-related metal shielding. The attenuating capability that is required is stated using tungsten as a reference. An equivalent amount of attenuation using stainless steel is also acceptable. However, neither AP1000 DCD Tier 1, Table 3.3-1, "Definition of Wall Thicknesses for Nuclear Island Buildings, Turbine Building, and Annex Building," nor DCD Tier 1, Section 2.2.5, "Main Control Room Emergency Habitability System," including Table 2.2.5-5, "Inspections, Tests, Analyses, and Acceptance Criteria," and Figure 2.2.5-1, "Main Control Room Emergency Habitability System," describe an ITAAC for verifying the presence, quantity, and the material properties of the VES shielding material. Therefore, in RAI 8028 Question 12.03-5, the staff asked the applicant whether an ITAAC for verifying the installation of the VES shielding material required to ensure compliance with GDC 19 is necessary. In the response dated November 2, 2015, the applicant revised the proposed departure to identify the VES filter shield in Tier 1, Tables 2.2.5-1 and 2.2.5-5, including a new ITAAC item 7e, which is consistent with modifications to Tier 2 of the licensing basis presented in the proposed FSAR Section 12.3.2.2.7. Because an ITAAC exists to ensure installation of design features needed to meet the regulatory requirements of GCD 19, the staff finds this response acceptable. The staff did not identify any additional issues associated with direct radiation exposure from the VES or VBS filters.

Through the addition of the additional shielding at the VES filter and the addition of the related ITAAC, the deficiency in the DCD analysis related to the direct dose contribution from the VES filter identified in the applicant's revised analysis provided as part of LNP DEP 6.4-1 is resolved. Because additional shielding ensures that the incremental increase to MCR operator dose resulting from the use of the VES filter would not result in exceeding the operator dose requirements of GDC 19, under the conditions analyzed in the DCD. Therefore, the staff finds the proposed changes acceptable.

B.3 Radiation monitor setpoint changes

As discussed in the response to RAI 7661, dated July 1, 2015, during its re-evaluation of MCR doses to include the direct dose contribution from HVAC filters, the applicant identified that the VBS radiation monitor setpoints in the AP1000 DCD, which were based on LOCA releases, were not selected in a manner that ensures that GDC is met for non-LOCA DBAs. In addition, they determined that the setpoints did not ensure the AP1000 design objective that the non-safety-related VBS supplemental filtration mode would be used when available, instead of initiating the safety-related VES. As stated in item 4 on page 5 of Enclosure 1 to the response to RAI 7661:

For postulated accident conditions involving a reduced source term or release rate other than evaluated for DBAs as part of the certified design, there may not be sufficient radioactivity within the MCR Envelope to prompt actuation of VES, and yet, enough radioactivity could exist that would lead to operator doses in excess of 5 rem [0.05 Sv] without manual actuation. The radiation monitor setpoint values are therefore updated to ensure VBS or VES filtration mode actuation occurs for any radiological release event that could result in MCR operator doses in excess of GDC-19.

Specifically, the applicant stated on page 3 of Enclosure 1 to the response to RAI 7661:

To ensure that GDC-19 is met for all design basis accidents, site-specific revisions to the radiation monitor setpoints will be included in the LNP COL application. These revised setpoints for MCR VES actuation will be based upon concentrations for any particular monitoring channel (particulate or iodine) not exceeding an operator dose of 1 rem [0.01 Sv]—regardless of release or accident scenario. This methodology will allow for airborne radioactivity in the control room to reach concentrations in each of the three channels at the setpoint and maintain compliance with GDC-19.

The applicant ensured that the postulated radioactive material releases for each DBA were conservatively compared to the setpoints to determine the timing of the initiation of the VES or the non-safety-related VBS supplemental filtration mode used as input to the MCR dose analyses. As the staff verified through audit of the proprietary radiation monitor setpoint calculation, the radiation monitor setpoints are calculated to correspond to a radioactive material concentration at the MCR HVAC intake that results in an MCR operator dose of 0.01 Sv (1 rem) in any channel because of the airborne release. Therefore, although the calculation of the VBS radiation monitor setpoints does not explicitly include the direct dose component of the MCR operator dose, the setpoint radioactive material concentration values provide sufficient margin to accommodate the addition of direct dose in the MCR and ensure that the GDC 19 dose criterion of 0.05 Sv (5 rem) TEDE is met. The staff finds these changes related to the VBS radiation monitor setpoints acceptable because they appropriately reflect the expected MCR HVAC system operation and provide acceptable input assumptions for use in each of the revised DBA dose analyses.

B.4 DBA dose analysis changes that affect the MCR airborne dose calculation

In addition to making changes to the DBA dose analyses to correct errors in the AP1000 DCD analysis of the direct dose component of the MCR dose as described above, the applicant revised the modeling of the MCR in the calculation of the dose to MCR operators from immersion in and inhalation of the airborne release. The applicant made these changes to the AP1000 DCD Chapter 15 analyses modeling of the MCR to partially offset the increase in MCR operator dose because of the revised direct dose calculations and to reflect general updates to the detailed design. The staff's review of these DBA dose analysis changes that affect the calculation of MCR airborne dose are discussed in the following B.4 subsections.

Although LNP DEP 6.4-1 is a site-specific departure from the AP1000 DCD, the revised DBA dose analyses provided by the applicant are generic analyses in that they use the same short-term (accident) atmospheric dispersion factor (χ/Q) values given as site parameters in AP1000 DCD, Section 2.3.4. For LNP DEP 6.4-1, no changes were made to the LNP site characteristic short-term χ/Qs given in FSAR 2.3.4; therefore, in accordance with the discussion of LNP COL 2.3-4 in Section 15A.4 of this safety evaluation, the LNP site-specific short-term χ/Q values are less than those used in the revised generic analysis supporting LNP DEP 6.4-1. The applicant did not provide site-specific doses at the EAB, LPZ, or MCR for the DBAs referenced in AP1000 DCD, Chapter 15, but instead provided the results of the revised generic DBA dose analysis, which are bounding for the LNP site.

The estimated DBA dose calculated for a particular site is affected by the site characteristics through the calculated χ/Q input to the analysis; therefore, the resulting dose would be different than that calculated generically for the AP1000 design in the revised generic analyses. All other inputs and assumptions in the radiological consequences analyses remain the same as in the revised generic analyses. Smaller χ/Q values are associated with greater dilution capability, resulting in lower radiological doses. When comparing a DCD site parameter χ/Q value and a site characteristic χ/Q value, the site is acceptable for the design if the site characteristic χ/Q value is smaller than the site parameter χ/Q value. Such a comparison shows that the site has better dispersion characteristics than that required by the reactor design.

For each of the DBAs, the LNP site-specific χ/Q values for each time averaging period are less than the comparable design reference χ/Q values used in the AP1000 DCD and the revised DBA dose analyses provided in LNP DEP 6.4-1. Because the result of the radiological consequences analysis for a DBA during any time period of radioactive material release from the plant is directly proportional to the χ/Q for that time period, and because the LNP site-specific χ/Q values are less than the comparable AP1000 design reference χ/Q values for all time periods and all accidents, the LNP site-specific estimated total dose at the EAB, LPZ, and the MCR for each DBA is, therefore, less than the generic *revised estimated total dose at the same receptor location for each DBA, as provided in LNP DEP 6.4-1.*

B.4.1 Increase in VES filter efficiency for organic iodine

As discussed in the response to RAI 7661, dated July 1, 2015, the applicant increased the assumed VES charcoal filter efficiency for organic iodine to 90 percent from the 30 percent value used in the AP1000 DCD Chapter 15 DBA dose analyses and the estimation of the DBA dose to the MCR operators as reported in AP1000 DCD Chapter 6.4. The applicant proposed this change to partially offset increases in the total dose to the operators related to the revised consideration of direct dose from VES filter shine and other refinements in the MCR direct dose calculations. The change in the VES filter organic iodine efficiency is noted as a revision to DCD Table 15.6.5-2, Sheet 2 of 3. The change in the assumed organic iodine efficiency for the VES filter is based upon the applicant's updated evaluation of the relative humidity expected in the MCR during post-accident operation of the VES and upon conformance with the guidance in RG 1.52, Revision 2, "Design, Testing, and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

As stated in Section 6.4.2.3 of the DCD incorporated by reference in the LNP COL application, the LNP VES charcoal adsorber is designed in accordance with ASME AG-1, Section FD, and RG 1.52. Each charcoal adsorber is an assembly with 2-inch deep Type II adsorber cells. RG 1.52 specifies the use of a safety factor of at least 2 when determining the appropriate methyl iodide penetration acceptance criterion in the TS for the representative sample of the charcoal adsorber. According to NRC Generic Letter 99-02, "Laboratory Testing of Nuclear-Grade Activated Charcoal," the following equation is used to determine the appropriate methyl iodide allowable penetration:

penetration = (100% - organic iodide efficiency credited in accident analysis)/safety factor)

In AP1000 DCD, Table 15.6.5-2, the charcoal filter efficiency for organic iodine credited in accident analysis has been revised from 30 percent to 90 percent. The efficiencies for elemental iodine, 90 percent, and particulates, 99 percent, remain the same. Section 5.5.13 of the LNP TS requires the laboratory testing of the VES charcoal filters at 30 degrees Celsius (C) (86 degrees Fahrenheit (F)) and 95 percent RH using the American Society for Testing and Materials standard ASTM D3803, "Standard Test Method for Nuclear-Grade Activated Carbon," with a test penetration of 5 percent.

Appling the above equation, the safety factor of two is satisfied.

Therefore, the required LNP TS laboratory test will ensure that the DBA dose analysis credited efficiency of 90 percent organic iodine will conservatively be met with margin (i.e. safety factor of 2) which accounts for potential degradation over the 24-month operating cycle.

B.4.2 Changes to MCR design input assumptions

The applicant's DBA dose analyses included revisions to the analysis input assumptions on MCR and MCR HVAC volume based on updated detailed design data. In addition, the VBS intake and VBS ancillary fan intake flow rates include a 10-percent uncertainty on the nominal flow rates used in the DCD Revision 19 Chapter 15 DBA dose analyses.

The staff finds these changes acceptable because they are based on detailed design data and include appropriate consideration of uncertainty.

As discussed in the response to RAI 7661, dated July 1, 2015, the applicant determined that the time modeled in the AP1000 DCD, Chapter 15, DBA analyses for the switchover from VBS normal operation to the VBS supplemental filtration mode based on the VBS radiation monitor reaching the non-safety-related High-1 MCR HVAC system setpoint was not bounding for non-LOCA analyses when the updated detailed design information was taken into account. Similarly, the VES initiation time assumed in the DCD non-LOCA DBA analyses was not bounding. To address this concern, the applicant revised the DBA dose analyses using updated detailed design information and included a longer delay interval between the time that the VBS radiation monitor reaches the High-1 setpoint concentration and the time when the non-safety-related VBS supplemental filtration mode is operational. The applicant's revised DBA dose analyses that show compliance with GDC 19 included consideration of a longer delay interval between the time that the VBS radiation monitor reaches the High-2 setpoint concentration and the time when the safety-related VES is operational, based on updated detailed design information.

In RAI Letter No. 129, dated July 13, 2015 (ADAMS Accession No. ML15194A263), RAI 8004 Question 06.04-10, the staff asked for more information on the calculated time after the beginning of the accident that the VBS radiation monitor setpoints are reached and the timing of initiation of the VES or VBS supplemental filtration mode. The applicant's response, dated October 13, 2015 (ADAMS Accession No. ML15289A228), provided information that listed the calculated times that the radiation monitor setpoints are reached and the times that the VES or VBS supplemental filtration mode begins operation for each of the DBAs based on the calculated radioactive material release for the specific DBA. Additional proprietary information was also provided on the estimated delay time for each event related to system initiation, including the time to detect the radioactive material, time for signal processing, and time to complete damper movement. The staff determined that the more detailed information supports the changes to the assumptions on timing of the VES and VBS systems operation made in the revised DBA dose analyses. The staff also determined that the proposed changes to DBA dose analysis input related to MCR HVAC system operation appropriately address the issue that the applicant identified where the DCD MCR dose analysis would not be bounding for non-LOCA DBAs. Therefore, the staff finds acceptable the proposed changes to the MCR design assumptions used as input to the DBA dose analyses, and RAI 8004, Question 06.04-10, is resolved.

B.5 Other DBA dose analysis changes that affect both the MCR dose and the offsite dose results

The applicant made additional changes to selected DBA dose analysis assumptions to reflect general detailed design updates. Because the proposed analysis changes result in a change of the calculated amount of radioactive material that is assumed to be released to the environment, the offsite dose results are also affected. The staff's review of these DBA dose analysis changes are discussed below in the following B.5 subsections.

B.5.1 Iodine re-evolution modeling in LOCA dose analysis

As discussed in the response to RAI 7661, dated July 1, 2015, to partially offset increases in the MCR operator dose because of addition of the VES filter shine and other analyses changes proposed in LNP DEP 6.4-1, the applicant made changes to the modeling assumptions regarding iodine re-evolution from the IRWST in the DBA LOCA dose analysis. Specifically, the proposed changes involve refining the assumed water/vapor partition factor for elemental iodine to be consistent with guidance in RG 1.183 and using updated AP1000 design information to determine revised timing associated with the conversion of elemental iodine to organic iodine and its availability for release from the IRWST fluid.

On page 6 of Enclosure 1 of the July 1, 2015, submittal, the applicant provided the following description of the specific proposed changes:

The iodine source term applied in the LOCA dose analysis supporting DCD Revision 19 is based upon the NUREG-1465 source term described in Regulatory Guide 1.183. The analysis models a staged release of core activity (i.e. gap release and early in-vessel) to the containment atmosphere over the first 2 hours following the start of the event. The chemical form of iodine released is assumed to be 95% particulate, 4.85% elemental, and 0.15% organic, consistent with Regulatory Guide 1.183. Particulate removal via passive processes (i.e., diffusiophoresis, thermophoresis, and sedimentation) and elemental iodine removal via deposition are modeled. Organic iodine removal via processes other than decay or leakage from containment is not modeled.

Particulates removed to the containment shell are assumed to be washed off the shell by the flow of water resulting from condensing steam (i.e. condensate flow). The particulates may be either washed into the sump, which is controlled to a pH > 7 post-accident or into the IRWST, which is not pH controlled post-accident. Due to the assumed conditions in the IRWST, the particulate iodine washed into the IRWST may chemically convert to an elemental form and re-evolve, subject to partitioning, as airborne. A portion (3%) of that airborne elemental iodine is then assumed to convert to an organic form. This is consistent with elemental organic split assumed for the initial release from the core (4.85/0.15 = 97/3) and is consistent the Regulatory Guide 1.183 guidance for other events.

The calculational approach to account for the iodine that is assumed to re-evolve from the IRWST post-LOCA is overly conservative in the certified design *analysis. The certified design analysis applies a water-steam partition factor of 5 for elemental iodine and neglects the time dependent formation of organic iodine from elemental iodine; the organic iodine that would be formed over time is assumed to be present at time zero.*

NUREG-1465 states that "It is unduly conservative to assume that organic iodine is not removed at all from containment atmosphere, once generated, since such an assumption can result in an overestimate of the long-term doses to the thyroid." The revised analysis approach applies a conservative water/vapor elemental iodine partition factor of 10, selected to conservatively bound the time-dependent partition factors calculated using the NUREG/CR-5950 models and IRWST temperature and pH as a function of time. Additionally, the conversion of elemental iodine to organic iodine is modeled on a time-dependent basis in which 3% of the evolved elemental iodine is assumed to convert to an organic form upon its release to containment. It is noted that this does not impact the percentage of iodine assumed to convert to the organic form.

Although this description of the proposed changes to the modeling of iodine re-evolution from the IRWST fluid during a DBA LOCA was given in Enclosure 1 of the submittal dated July 1, 2015, no markup of DCD text was given to document the site-specific changes in the LNP FSAR. In RAI Letter No. 129, the staff issued RAI 8005 Question 15.00.03-4 asking for additional detail on the revised modeling of iodine re-evolution from the IRWST, including values for the time-dependent pH and partition coefficients for the water in the IRWST. The staff also asked that the applicant document the specifics of this departure from the DCD dose analysis in the LNP FSAR.

In the response to RAI 8005 Question 15.00.03-4, dated October 13, 2015, the applicant provided the requested detailed information marked as proprietary information. The staff was able to audit the proprietary LOCA DBA calculation package and verified that the LOCA DBA dose calculation inputs agreed with the information given in the RAI response. The response to Question 15.00.03-4 also provided text to describe the LNP DEP 6.4-1 change to iodine re-evolution modeling, which the staff verified was added to Revision 8 of the LNP FSAR, Section 15.6.5.3.2.

The staff finds through review of the description of the departure that the applicant's revisions to the iodine re-evolution analysis use models and methods that have been previously found acceptable to the staff, as noted in RG 1.183. The staff also determined through review of the proprietary information provided that the applicant's inputs and assumptions reflect the AP1000 design information and are acceptable. A description of the changes made to the LOCA dose analysis modeling of iodine re-evolution from the IRWST was added to the LNP FSAR. Therefore, the staff finds the proposed changes to the modeling of IRWST iodine re-evolution acceptable and RAI 8005, Question 15.00.03-4, is resolved.

B.5.2 Increase in containment elemental iodine deposition removal coefficient

In the revised LOCA and REA dose analyses, the applicant increased the *passive containment elemental iodine deposition coefficient value to 1.9 hr^{-1} from the AP1000 DCD value of 1.7 hr^{-1}. The change in the deposition removal coefficient value was calculated based on a larger containment surface area available for deposition, as determined in the AP1000 updated detailed design.*

Through audit of the revised LOCA and REA dose analyses, the staff verified that the calculations used the increased containment elemental iodine deposition coefficient as input. The staff finds the increased containment elemental iodine deposition coefficient acceptable because the value was calculated using the same method that was found acceptable in review of the DCD, with the only change the incorporation of updated detailed design information as input to the calculation of the deposition coefficient.

B.5.3 Revised steam release rates for the MSLB dose analysis

The applicant calculated revised steam release rates from the secondary coolant system based on calculation of an earlier time for steam generator dry-out, which would be limiting for MCR dose estimation. As stated on page 7 of Enclosure 1 to the response to RAI 7661, dated July 1, 2015:

The AP1000 steam line break accident analysis described in DCD Revision 19 assumes a 10 minute faulted steam generator (SG) blowdown based on a Hot Zero Power (HZP) SG mass released at an average rate. This HZP case is conservative for offsite dose. It was determined, however, that a full power SG mass could lead to SG dry-out occurring at ~200 seconds. Earlier dry-out is more limiting for the purposes of operator post-accident dose calculations. To ensure a conservative dose for both offsite and MCR, the HZP initial mass was retained, a bounding release rate was modeled until 300 seconds, and any remaining activity was released thereafter.

Through audit of the revised MSLB dose analyses, the staff verified that the calculation used revised steam release rates as input. Calculating an earlier time for steam-generator dry-out results in an earlier increase in the estimated release of radioactive material to the environment because of reduced retention in the steam generators. Because there is a delay in the timing of the control room VES initiation, the calculation of the MCR dose is more sensitive to the timing of the increase in the SGTR releases, as compared to the calculation of the offsite doses. The staff finds the revised steam release rates acceptable because the values were calculated using the same method that was found acceptable in review of the DCD, with the only change to the calculation of the mass releases being the use of a more limiting power condition for the estimation of the timing of steam generator dry-out and the subsequent effect on the calculation of the MCR dose.

B.5.4 TS secondary coolant iodine activity concentration limit reduced to 0.01 µCi/gm DEI-131

In the revised dose analyses for the MSLB, REA, SGTR and LRA, in order to offset increases in the calculated MCR operator dose due to other changes in the DBA dose analyses, particularly the MSLB steam releases as discussed above in

Section B.5.3, the applicant reduced the assumed secondary coolant iodine activity concentration to 0.01 µCi/gm DEI-131. To reflect this change, the applicant also proposed to revise the TS LCO 3.7.4 limit for secondary coolant iodine concentration from the AP1000 generic value of 0.1 µCi/gm DEI-131 to 0.01 µCi/gm DEI-131.

The site-specific departure on the TS LCO limit for secondary coolant allowable iodine concentration results in a lower amount than allowed by the AP1000 generic TS of radioactive material available for release during DBAs that include release of the secondary coolant through break flow or through steaming to cool down the RCS). The staff verified that the revised MSLB, REA, SGTR and LRA dose analyses assume that the secondary coolant is at the TS allowable limit at the beginning of the accident in accordance with the guidance in RG 1.183. Therefore, the staff finds that the proposed LNP DEP 6.4-1 change to TS LCO 3.7.4 was appropriately accounted for in the safety analyses provided to support the departure.

B.5.5 Change in methodology to estimate fuel damage in the REA dose analysis

The applicant revised the method to estimate fuel damage for the REA to be based on an updated accepted methodology. As stated on page 8 of Enclosure 1 to the response to RAI 7661, dated July 1, 2015:

The method for performing the REA dose analysis has changed from that applied in DCD Revision 19. As stated in NUREG-1793, the NRC accepted the use of NUREG-0800 Section 4.2 Revision 2 for design certification of the AP1000 plant. However, in NUREG-1793 Supplement 2 it is stated that:

"For COL applicants or licensees who reference the AP1000 or AP600 certified designs, the staff will review any change or departure from the certified design that requires prior NRC approval as specified in Section VIII of Appendices C and D to 10 CFR Part 52, respectively.

The staff will evaluate the reactivity-initiated accidents such as rod ejection accidents based on the acceptance criteria in effect 6 months before docketing the amendment request, such as the interim acceptance criteria specified in Appendix B to NUREG-0800 Section 4.2, Revision 3, if a change or departure in fuel design or other aspects is proposed that requires a reevaluation of final safety evaluation report Chapter 4, "Reactor," or Chapter 15, "Transient and Accident Analysis."

Due to the need to incorporate other design changes in the REA MCR operator dose calculations, NUREG-0800, Section 4.2, Revision 3, is used for recalculation of the rod ejection dose analysis, which results in a significant impact to the rod ejection dose analysis. NUREG-0800, Section 4.2, Revision 3, precludes fuel melt, providing a dose benefit, but also connects the source term to the fuel enthalpy increase, which is a significant dose penalty. The dominant contributor to the increased dose is the increase by a factor of more than 5 in alkali metal releases.

The staff evaluated the information provided in the July 1, 2015, response to RAI 7661 and through audit of the proprietary calculation package verified that the revised fuel failure assumptions were reflected in the revised REA dose analysis. The method the applicant used to estimate fuel failure and fission product release during the REA is in conformance with the guidance in SRP, Revision 3, Section 4.2, which the staff stated in NUREG-1793 is an acceptable methodology for this purpose. The staff also determined that the fuel enthalpy input to the calculation of the fuel failure was consistent with the AP1000 design information. Therefore, the staff finds acceptable the proposed changes in LNP DEP 6.4-1 related to the estimation of fuel failure for the REA dose analysis.

B.5.6 Increase in SG moisture carryover assumptions

In the revised dose analyses for the REA, SGTR, and LRA, the assumed full-power moisture carryover from the steam generators was increased from the value of 0.1 percent used in AP1000 DCD to 0.35 percent to be consistent with the updated AP1000 detailed design.

In RAI Letter 129, RAI 8005, Question 15.00.03-2, dated July 13, 2015, the staff noted that using the increased full-power moisture carryover from the steam generators of 0.35 percent to model alkali metal releases to the environment in the revised DBA analyses that assume release through the secondary system is consistent with guidance in Appendix E of RG 1.183 (ADAMS Accession No. ML15194A263). However, the staff also noted that the value for the full-power moisture carryover is larger than the maximum weight percent moisture carryover value of 0.25 percent listed in AP1000 DCD Table 5.4-4, "Steam Generator Design Requirements," and asked that applicant clarify this apparent discrepancy. In its response to RAI 8005, Question 15.00.03-2, dated October 13, 2015, the applicant stated that the value of 0.35 percent for moisture carryover used in the REA, SGTR, and LRA dose analyses was chosen to be a conservative bounding value for analysis purposes, and is considered to be an upper bound for the amount of moisture carryover that could be expected during plant operation and is consistent with the value considered in RCS design (ADAMS Accession No. ML15289A228). The staff agrees that using the larger moisture carryover assumption in the DBA dose analyses is conservative for the design. Therefore, the staff finds that the use of a conservative steam generator moisture carryover assumption in the DBA dose analyses is acceptable, and RAI 8005, Question 15.00.03-2, is resolved.

B.5.7 Additional changes to SGTR dose analysis assumptions

In addition to changes to the steam generator moisture carryover and the assumed secondary coolant iodine activity concentration in the revised SGTR dose analysis, the applicant proposed to increase the duration of steam releases from the values used in the AP1000 DCD and decrease the initial values assumed for the reactor coolant mass and secondary coolant mass.

In RAI Letter 129, RAI 8005, Question 15.00.03-3, the staff requested that the applicant provide the basis for these proposed changes to the SGTR dose *analysis. In the response to RAI 129, Question 15.00.03-3, the applicant stated that the changes were conforming changes to reflect the updated AP1000 detailed design and are conservative values to provide additional margin for future design updates. Through audit of the revised SGTR dose analyses, the staff verified that the calculation used the proposed revisions to the duration of steam release and the primary and secondary coolant mass values as input to the analyses. Because the applicant made these changes to reflect the updated detailed design and to provide additional analysis margin, the staff finds the changes acceptable, and RAI 8005, Question 15.00.03-3, is resolved.*

B.5.8 Change in assumed fuel radial peaking factor to account for advanced first core design

In the revised dose analyses for the REA, LRA, and FHA, the applicant changed the fuel radial peaking factor to a value of 1.75, which is higher than the value of 1.65 used in the AP1000 DCD DBA dose analyses. The increase in the fuel radial peaking factor was proposed in order to provide additional analysis margin for future core design changes. This results in a 6 percent increase to the estimated amount of radioactive material released from the fuel.

Through audit of the revised REA, LRA, and FHA dose analyses, the staff verified that the calculations used the increased fuel radial peaking factor as input to the analyses. Because the applicant proposed the increased fuel radial peaking factor as a conservative multiplying factor to provide additional analysis margin, the staff finds the increased radial peaking factor acceptable.

B.5.9 Small line break flashing fraction increased based on updated detailed design

The applicant's revised small line break dose analysis included an increase in the assumed fraction of reactor coolant flashing to steam from the value that was used in AP1000 DCD small line break dose analysis. The flashing fraction is increased from 0.41 to 0.47 based on the updated AP1000 detailed design and the determination that the RCS hot leg temperature should be used to calculate the flashing fraction instead of basing it on the vessel average temperature as was done in the AP1000 DCD small line break dose analysis.

Through audit of the revised small line break dose analyses, the staff verified that the calculation used increased flashing fraction as input. The staff finds the revised flashing fraction acceptable because the value was calculated using the same method that was found acceptable in review of the AP1000 DCD, with the only change to the calculation of the flashing fraction being the correction of the coolant temperature, which was based on updated detailed design information.

B.6 Comparison of revised DBA doses to regulatory criteria

Because the revised generic DBA dose analyses that support LNP DEP 6.4-1 show that the offsite radiological consequences meet the regulatory dose requirements of 10 CFR 52.79(a)(1)(vi), and because, by the reasoning above in Section B.4, the LNP site-specific DBA radiological consequences are estimated to be less than those calculated in the revised generic DBA dose analyses, the applicant has sufficiently shown that the DBA offsite radiological consequences meet the requirements 10 CFR 52.79(a)(1)(vi).

Because the revised generic DBA dose analyses that support LNP DEP 6.4-1 show that the DBA MCR radiological consequences meet the regulatory dose requirements of GDC 19, and because, by the reasoning above in Section B.4, the LNP site-specific DBA MCR radiological consequences are estimated to be less than those calculated in the revised generic DBA MCR dose analyses, the applicant has sufficiently shown that the DBA MCR radiological consequences meet the requirements of GDC 19.

Based on the technical evaluation discussion above in Section B, the staff finds that LNP DEP 6.4-1 sufficiently addresses the concerns raised in RAI 7661, Question 06.04-2. Therefore, RAI 7661, Question 06.04-2 is resolved.

B.7 Risk Results and Insights

This design departure does not alter the description of AP1000 design features relevant to human performance in the control room. It does not modify the plant-specific PRA model used for licensing. Consequently, there is no change to the risk profile described in the COL application or the risk insights concerning the control room AP1000 DCD Revision 19, Table 19.59-18, item 20. Instead, the change improves confidence in the validity of the reported risk results and insights. Consistent with DC/COL ISG 003, "PRA Information to Support Design Certification and Combined License Applications," the plant-specific PRA remains acceptable to the staff.

21.2.5 Post Combined License Activities

For the reasons discussed in the technical evaluation section above, the staff finds acceptable item 7e proposed to be inserted in DCD Table 2.2.5-5, reproduced below in Table 21.2-1.

Table 21.2-1: DCD ITAAC item 7e from DCD Table 2.2.5-5, as revised by WLS DEP 6.4-1

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7e) Shielding below the VES Filter is capable of providing attenuation that is sufficient to ensure main control room doses are below an acceptable level during VES operation.	Inspection will be performed for the existence of a report verifying that the as-built shielding meets the requirements for functional capability.	A report exists and concludes that the as-built shielding identified in Table 2.2.5-1 meets the functional requirements and exists below the filtration unit, and within its vertical projection.

21.2.6 Conclusion

The staff reviewed the application for proposed departure number WLS DEP 6.4-1 and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to the departure, including the design change and revised DBA dose analyses related to addressing errors in the AP1000 DCD MCR dose assessment, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the regulatory requirements and guidance discussed in Section 21.2.3 of this SER. The staff based its conclusion on the following:

- Based on the evaluation discussed above, the staff concludes that the revised DBA dose departure from the AP1000 design certification rule at the WLS Units 1 and 2 site meets the 10 CFR 52.79(a)(1)(vi) dose criteria and the offsite dose acceptance criteria, as given in SRP 15.0.3 and RG 1.183 for these accidents.

- The staff finds reasonable assurance that the VES, under High-2 radiological conditions as described in FSAR Section 6.4 and WLS DEP 6.4-1, can mitigate the dose in the MCR following DBAs to meet the dose acceptance criterion specified in GDC 19.

- The staff finds it reasonable that, if available, the non-safety-related VBS as described in FSAR Sections 6.4 and 9.4.1, and in WLS DEP 6.4-1 can mitigate the dose in the MCR following DBAs to be within 0.05 Sv (5 rem) TEDE.

21.3 <u>Main Control Room Heat Load</u>

21.3.1 Introduction

The AP1000 DCD Tier 2, Section 6.4.3.2, describes how the temperature and humidity in the MCR pressure boundary remain within limits for reliable human performance over a 72-hour period. At a public meeting held on July 23, 2014 (ADAMS Accession Nos. ML14192A803 and ML14220A113), with Westinghouse, the staff received information that a more limiting transient had been identified and that additional heat sources exist in the control room that were not accounted for in the original analysis that may challenge the ability of the plant to meet control room habitability requirements and equipment qualification limits.

The AP1000 design normally uses the non-safety related nuclear island nonradioactive ventilation system (VBS) to provide heating, ventilation, cooling, and filtration to the MCR when power is available. During events where VBS is unavailable, however, the MCR emergency habitability system (VES) uses a combination of bottled air and passive heat sinks to maintain the MCR in a habitable state. As a result of development of the detailed AP1000 design, the applicant identified that the VES is not capable of maintaining the MCR in an acceptable condition for human performance during certain transients. Acceptability, in the certified design, is defined as an MCR effective temperature of 85 °F (29 °C), which corresponds to a dry bulb temperature of 95 °F (35 °C) with a relative humidity (RH) of 50 percent.

During events where the MCR is isolated (e.g., because of radiological conditions exceeding the VES actuation setpoint or both trains of VBS are unavailable) and VES is actuated, but offsite power is available to power other plant equipment, the heat loads in the MCR further exceed those set forth in the certified design. In a letter dated February 9, 2016 (ADAMS Accession No. ML16043A123), the applicant endorsed RAI responses on the Levy docket stating that the heat sources in the MCR exceeded those assumed in the DCD. As such, an event resulting in MCR isolation with offsite power available would result in significantly higher heat loads than described in the DCD, and so a revised approach to evaluate the heat load in the MCR was required. The applicant proposed a design change to add a load shedding arrangement to some of the MCR heat loads, changed the acceptance criteria for the MCR temperature for human performance to a wet bulb globe temperature of 90 °F (32 °C) (consistent with NUREG-0700, Revision 2, "Human-System Interface Design Review Guidelines" for an unlimited stay time), revised the curve defining equipment qualification limits, revised the analysis supporting the habitability of the MCR to incorporate the new heat loads and other analysis changes, and changed the classification of a set of valves in the VES from inactive to active.

21.3.2 Summary of Application

DEC incorporated in WLS COL application, dated April 11, 2016 (ADAMS Accession No. ML16124A854), the same information that DEF incorporated into the LNP COL application related to the voluntary submittal of an exemption request and design change description for departure from the AP1000 DCD to address main control room heat load. The information was originally submitted in endorsement and exemption request letter dated February 9, 2016 (ADAMS Accession No. ML16043A123).

Tier 1 and Tier 2 Departure

The applicant included the following Tier 1 and Tier 2 departure from the AP1000 DCD:

- WLS DEP 6.4-2

AP1000 DCD, Revision 19, Tier 2 Section 6.4.3.2, describes how the temperature and humidity in the MCR are maintained within the limits for reliable human performance. The applicant requested an exemption and site specific departure WLS DEP 6.4-2 from the AP1000 DCD, Revision 19, for the WLS Units 1 and 2 COL application to address newly identified limiting transients and heat sources in the MCR.

This exemption request includes changes to plant-specific DCD Tier 1 information and generic TS with other Tier 2 involved departures. Therefore, these departures require NRC approval and are evaluated below.

21.3.3 Regulatory Basis

The acceptance criteria for the staff review of the design and qualification of the main control room habitability system include the following:

- 10 CFR Part 50, Appendix A, GDC 2 requires that safety-related portions of the control room ventilation system be designed to withstand the effects of natural phenomena.

Meeting the requirements associated with GDC 2 provides assurance that the habitability of the control room area will be maintained and that equipment in the control room will operate as designed, thereby minimizing the potential for loss of function.

- GDC 4 requires that SSCs important to safety be designed to accommodate the effects of environmental conditions of normal operation, maintenance, testing, and postulated accidents. Meeting the requirements associated with GDC 4 provides assurance that control room ventilation system will support the functioning of systems and components important to safety by maintaining suitable environmental conditions for performance of safety functions.

- GDC 19 requires that the control room remain functional to the degree that actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain the plant in a safe condition under accident conditions. This is accomplished by providing adequate protection to equipment and operators to permit access to and occupy the control room under accident conditions.

The acceptance criteria associated with the human factors review include the following:

- 10 CFR 50.34(f)(2)(iii), which requires a control room design that reflects state-of-the-art human factor principles. Guidance applicable to design-related human factors principles is set out in NUREG-0700.

The acceptance criteria for the staff review of the design and qualification of the instrumentation and controls include the following:

- 10 CFR 50.55a(h)(3), “Protection and Safety Systems,” requires compliance with Institute of Electrical and Electronics Engineers (IEEE) Std. 603-1991, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” and the correction sheet dated January 30, 1995. Clause 5.1 of IEEE Std. 603-1991, “Single Failure Criterion,” requires, in part, that safety systems shall perform all safety functions required for a design-basis event in the presence of (1) any single detectable failure within the safety systems concurrent with all identifiable but non-detectable failures, (2) all failures caused by the single failure, and (3) all failures and spurious system actuations that cause or are caused by the design-basis event requiring the safety functions. Clause 5.6.3 of IEEE Std. 603-1991, “Between Safety Systems and Other Systems,” requires, in part, that the safety system design shall be such that credible failures in and consequential actions by other systems, as documented in Clause 4.8 of the design basis, shall not prevent the safety systems from meeting the requirements of this standard.

- GDC 13, “Instrumentation and Control,” requires, in part, that instrumentation shall 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 assure adequate safety.

- Clause 5.4 of IEEE Std. 603-1991, “Equipment Qualification,” requires safety system equipment be qualified by type test, previous operating experience, or analysis, or any combination of these three methods, to substantiate that it will be capable of meeting, on a continuing basis, the performance requirements as specified in the design basis.

The acceptance criteria for the staff review of the design, qualification (functional, seismic, and environmental), and inservice testing (IST) programs for safety-related valves include the following:

- GDC 1 requires that valves important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Meeting the requirements of GDC 1 provides assurance that valves important to safety are capable of performing their intended safety functions.

- GDC 2 requires that components important to safety be designed to withstand the effects of expected natural phenomena, combined with appropriate effects of normal and accident conditions, without loss of capability to perform their safety functions. Meeting the requirements of GDC 2 provides assurance that valves important to safety are capable of withstanding the effects of expected natural phenomena while performing their safety functions during and after the occurrence of those phenomena, as applicable.

- GDC 4 requires that components important to safety be designed to accommodate the effects of, and be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. Meeting the requirements of GDC 4 provides assurance that the components can withstand those effects and perform their intended safety functions.

- 10 CFR 50.55a(f) requires that applicable valves whose function is required for safety be assessed for operational readiness in accordance with the applicable revision to the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code). Meeting the requirements of 10 CFR 50.55a(f) provides assurance that applicable valves important to safety are capable of performing their intended safety function.

21.3.4 Technical Evaluation

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff’s findings on standard content that were documented in the SER for the reference COL application (LNP Units 1 and 2) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the LNP COL FSAR, Revision 9 to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.

- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.
- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting.

Tier 1 and Tier 2 Departures

- WLS DEP 6.4-2

Although the staff concluded that the majority of the evaluation performed for the standard content is directly applicable to the WLS applicant from that provided by the LNP applicant regarding ISG-11 changes for main control room heat load, there was a site specific difference. This difference is evaluated by the staff below. The following portion of this technical evaluation section is reproduced from Section 21.3.4 of the LNP COL application FSER.

- *LNP DEP 6.4-2*

LNP DEP 6.4-2 proposes to change the safety-related MCR VES to control the heat-up of the MCR envelope (MCRE) following VES actuation to meet the licensing basis requirements for equipment qualification and human factors engineering, described in DCD Tier 1 Subsection 2.2.5 and would also add generic TS to conduct surveillances of the revised components of the VES. The proposed changes do not change the VES safety-related design requirements and design functions.

The staff reviewed a request for an exemption submitted by the applicant. The request proposed changes to Tier 1 Tables 2.5.2-3, 2.5.2-4, 2.2.5-4, and 2.2.5-1 in the AP1000 DCD and generic TS 3.3.2, TS Table 3.3.2-1, TS 3.7.6, and TS surveillances (SRs) 3.7.6.3, 3.7.6.8, and 3.7.6.12. Additionally, the staff reviewed the associated changes to Tier 2 information for potential effects on safety functions of the MCR VES and the associated TS Bases in Chapter 16. The regulatory evaluation of the exemption request appears in Subsection A, below, and the technical evaluation of the exemption request and departure appears in Subsection B, below.

A. Regulatory Evaluation of Exemption Request

A.1 Summary of Exemption

The applicant requested an exemption from the provisions of 10 CFR Part 52, Appendix D, Section III.B, "Design Certification Rule for the AP1000 Design, Scope and Contents," that require the applicant referencing a certified design to incorporate by reference Tier 1 information. Specifically, the applicant proposed to revise Tier 1 Tables 2.5.2-3, 2.5.2-4, 2.2.5-4, and 2.2.5-1 (1) to ensure the VES design functions to maintain heat loads inside the MCRE within design-basis assumptions to limit the heat-up of the room, (2) to ensure a

72-hour supply of breathable-quality air for the occupants of the MCRE, (3) to maintain the MCRE pressure boundary at a positive pressure with respect to the surrounding areas, and (4) to provide a passive recirculation flow of MCRE air to maintain MCR dose rates below an acceptable level during VES operation.[5]

A.2 Regulations

- *10 CFR Part 52, Appendix D, Section VIII.A.4 states that exemptions from Tier 1 information are governed by the requirements of 10 CFR 52.63(b)(1) and 10 CFR 52.98(f). It also states that the Commission will deny such a request if the design change causes a significant reduction in plant safety otherwise provided by the design. This subsection of Appendix D also provides that a design change requiring a Tier 1 change shall not result in a significant decrease in the level of safety otherwise provided by the design.*

- *10 CFR Part 52, Appendix D, Section VIII.C.4 states that an applicant may request an exemption from the generic TS or other operational requirements. The Commission may grant such a request only if it determines that the exemption will comply with the requirements of 10 CFR 52.7.*

- *10 CFR 52.63(b)(1) allows an applicant or licensee to request NRC approval for an exemption from one or more elements of the certification information. The Commission may only grant such a request if it complies with the requirements of 10 CFR 52.7, which in turn points to the requirements listed in 10 CFR 50.12 for specific exemptions, and if the special circumstances present outweigh the potential decrease in safety due to reduced standardization. Therefore, any exemption from the Tier 1 information certified by Appendix D to 10 CFR Part 52 must meet the requirements of 10 CFR 50.12, 52.7, and 52.63(b)(1).*

[5] Although the applicant describes the requested exemption as being from Section III.B of 10 CFR Part 52, Appendix D, the entirety of the exemption pertains to proposed departures from Tier 1 information and generic TS in the generic DCD. In the remainder of this evaluation, the NRC will refer to the exemption as an exemption from Tier 1 information and generic TS to match the language of Sections VIII.A.4 and VIII.C.4 of 10 CFR Part 52, Appendix D, which specifically govern the granting of exemptions from Tier 1 information and generic TS.

A.3 Evaluation of Exemption

As stated in Section VIII.A.4 of Appendix D to 10 CFR Part 52, an exemption from Tier 1 information is governed by the requirements of 10 CFR 52.63(b)(1) and 52.98(f). Additionally, the Commission will deny an exemption request if it finds that the requested change to Tier 1 information will result in a significant decrease in safety. Pursuant to 10 CFR 52.63(b)(1), the Commission may, upon application by an applicant or licensee referencing a certified design, grant exemptions from one or more elements of the certification information, so long as the criteria given in 10 CFR 50.12 are met and the special circumstances as defined by 10 CFR 50.12 outweigh any potential decrease in safety due to reduced standardization.

As stated in Section VIII.C.4 of Appendix D to 10 CFR Part 52, the Commission may grant an exemption from generic TS of the DCD only if it determines that the exemption will comply with the requirements of 10 CFR 52.7. As stated above, Section 52.7 points to 10 CFR 50.12 for specific exemptions.

Applicable criteria for when the Commission may grant the requested specific exemption are provided in 10 CFR 50.12(a)(1) and (a)(2). Section 50.12(a)(1) provides that the requested exemption must be authorized by law, not present an undue risk to the public health and safety, and be consistent with the common defense and security. The provisions of 10 CFR 50.12(a)(2) list six special circumstances for which an exemption may be granted. It is necessary for one of these special circumstances to be present in order for NRC to consider granting an exemption request. The applicant stated that the requested exemption meets the special circumstances of 10 CFR 50.12(a)(2)(ii). That subsection defines special circumstances as when "[a]pplication of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule." The staff's analysis of each of these findings is presented below.

A.3.1 Authorized by Law

This exemption would allow the applicant to implement approved changes to Tier 1 Tables 2.5.2-3, 2.5.2-4, 2.2.5-4, and 2.2.5-1 and generic TS 3.3.2, TS Table 3.3.2-1, TS 3.7.6, and TS SRs 3.7.6.3, 3.7.6.8, and 3.7.6.12. This is a permanent exemption limited in scope to particular Tier 1 information and generic TS, and subsequent changes to this information or any other Tier 1 information or generic TS would be subject to full compliance with the change processes specified in Sections VIII.A.4 and VIII.C.4 of Appendix D to 10 CFR Part 52. As stated above, 10 CFR 52.63(b)(1) allows the NRC to grant exemptions from one or more elements of the certification information, namely, as discussed in this exemption evaluation, the requirements of Tier 1. Moreover, Section VIII.C.4 allows the NRC to grant exemptions from generic TS if the exemption meets the requirements of 10 CFR 52.7 and 50.12. The staff has determined that granting of the applicant's proposed exemption will not result in a violation of the Atomic Energy Act of 1954, as amended, or the NRC's regulations. Therefore, as required by 10 CFR 50.12(a)(1), the exemption is authorized by law.

A.3.2 No Undue Risk to Public Health and Safety

The underlying purpose of AP1000 Tier 1 Tables 2.5.2-3, 2.5.2-4, 2.2.5-4, and 2.2.5-1 and generic TS 3.3.2, TS Table 3.3.2-1, TS 3.7.6, and TS SRs 3.7.6.3, 3.7.6.8, and 3.7.6.12 is to ensure that the plant will be constructed and operated with a safe and reliable VES in the event of an accident.

The changes to the VES system description and associated TS (1) ensure the VES design functions to maintain heat loads inside the MCRE within design-basis assumptions to limit the heat-up of the room, (2) ensure a 72-hour supply of breathable-quality air for the occupants of the MCRE, (3) maintain the MCRE pressure boundary at a positive pressure with respect to the surrounding areas, and (4) provide a passive recirculation flow of MCRE air to maintain MCR dose rates below an acceptable level during VES operation. The changes to the VES system therefore support the system's intended design functions. The plant-specific Tier 1 DCD and TS will continue to meet regulatory requirements for protecting public health and safety and will maintain a level of detail consistent with what is provided elsewhere in Tier 1 of the plant-specific DCD. The affected design description in the plant-specific Tier 1 DCD will continue to provide the detail necessary to support the performance of the associated ITAAC. The proposed changes to Tier 1 information and generic TS are evaluated and found to be acceptable in Section 21.3 of this safety evaluation. Therefore, the staff finds the exemption presents no undue risk to public health and safety as required by 10 CFR 50.12(a)(1).

A.3.3 Consistent with Common Defense and Security

The proposed exemption would allow the applicant to implement modifications to the Tier 1 information and generic TS requested in the applicant's submittal. This is a permanent exemption limited in scope to particular Tier 1 information and a specific TS. Subsequent changes to this information or any other Tier 1 information or generic TS would be subject to full compliance with the change processes specified in Sections VIII.A.4 and VIII.C.4 of Appendix D to 10 CFR Part 52. This change is not related to security issues. Therefore, as required by 10 CFR 50.12(a)(1), the staff finds that the exemption is consistent with the common defense and security.

A.3.4 Special Circumstances

Special circumstances, in accordance with 10 CFR 50.12(a)(2)(ii), are present whenever application of the regulation in the particular circumstances would not serve the underlying purposes of the rule or is not necessary to achieve the underlying purpose of the rule. The underlying purposes of the specific Tier 1 Tables 2.5.2-3, 2.5.2-4, 2.2.5-4, and 2.2.5-1 modified in the exemption request is (1) to ensure the VES design functions to maintain heat loads inside the MCRE within design-basis assumptions to limit the heat-up of the room, (2) to ensure a 72-hour supply of breathable-quality air for the occupants of the MCRE, (3) to maintain the MCRE pressure boundary at a positive pressure with respect to the surrounding areas, and (4) to provide a passive recirculation flow of MCRE air to maintain MCR dose rates below an acceptable level during VES operation. The *underlying purposes of the specific generic TS 3.3.2, TS Table 3.3.2-1, TS 3.7.6, and TS SRs 3.7.6.3, 3.7.6.8, and 3.7.6.12 modified in the exemption request is to identify and conduct surveillances of the components that will be revised in the design of the VES. The revised components and new surveillance requirements for those components ensure that the VES can perform its intended function.*

Application of the requirements in Tier 1 Tables 2.5.2-3, 2.5.2-4, 2.2.5-4, and 2.2.5-1 and generic TS 3.3.2, TS Table 3.3.2-1, TS 3.7.6, and TS SRs 3.7.6.3, 3.7.6.8, and 3.7.6.12 is not necessary to achieve the underlying purpose of those portions of the rule. The proposed revisions to the VES support the system's intended design functions, as does the addition of generic TS to conduct surveillances of those revised components. The system and tables listing its components and surveillances, as modified in the requested exemption, will continue to perform its intended function and will, therefore, meet the underlying purpose of the rule. Accordingly, because application of the requirements in Tier 1 Tables 2.5.2-3, 2.5.2-4, 2.2.5-4, and 2.2.5-1 and generic TS 3.3.2, TS Table 3.3.2-1, TS 3.7.6, and TS SRs 3.7.6.3, 3.7.6.8, and 3.7.6.12 is not necessary to achieve the underlying purpose of the rule, special circumstances are present. Therefore, the staff finds that special circumstances required by 10 CFR 50.12(a)(2)(ii) for the granting of an exemption from the Tier 1 information and generic TS described above.

A.3.5 Special Circumstances Outweigh Reduced Standardization

This exemption, if granted, would allow the applicant to change certain Tier 1 information incorporated by reference from the AP1000 DCD into the LNP COL application. An exemption from Tier 1 information may only be granted if the special circumstances of the exemption request, required to be present under 10 CFR 52.7 and 10 CFR 50.12, outweigh any reduction in standardization. The proposed exemption would modify the VES to support the system's intended design functions. The proposed additions to the system support the system's intended design functions and the key design functions of the VES will be maintained.[6]

As described below in the technical evaluation, the changes to the VES (1) maintain heat loads inside the MCRE within design-basis assumptions to limit the heat-up of the room, (2) ensure a 72-hour supply of breathable-quality air for the occupants of the MCRE, (3) maintain the MCRE pressure boundary at a positive pressure with respect to the surrounding areas, and (4) provide a passive recirculation flow of MCRE air to maintain MCR dose rates below an acceptable level during VES operation. While there is a small possibility that standardization may be slightly reduced by granting the exemption from the specified Tier 1 requirements, the proposed exemption modifying the VES will result in no reduction in the level of safety. For this reason, the staff determined that, even if other AP1000 licensees and applicants do not request similar departures, the special circumstances supporting this exemption outweigh the

[6] Based on the nature of the proposed changes to the generic Tier 1 information in Tables 2.5.2-3, 2.5.2-4, 2.2.5-4, and 2.2.5-1, which maintain and support the design functions of the VES, other AP1000 licensees and applicants may request the same exemption, preserving the intended level of standardization.

potential decrease in safety because of reduced standardization of the AP1000 design, as required by 10 CFR 52.63(b)(1).

A.3.6 No Significant Reduction in Safety

The proposed exemption would modify the VES from the design presented in the original application. As described below in the technical evaluation, the changes to the VES (1) maintain heat loads inside the MCRE within design-basis assumptions to limit the heat-up of the room, (2) ensure a 72-hour supply of breathable-quality air for the occupants of the MCRE, (3) maintain the MCRE pressure boundary at a positive pressure with respect to the surrounding areas, and (4) provide a passive recirculation flow of MCRE air to maintain MCR dose rates below an acceptable level during VES operation. Because the proposed changes will ensure that the VES design will support the system's intended design functions and will not adversely affect its function, there is no reduction in the level of safety. Therefore, the staff finds that granting the exemption would not result in a significant decrease in the level of safety otherwise provided by the design, as required by 10 CFR Part 52, Appendix D, Section VIII.A.4.

A.4 Conclusion

The staff has determined that, as required by Section VIII.A.4 of Appendix D to 10 CFR Part 52, the exemption: (1) is authorized by law, (2) presents no undue risk to the public health and safety, (3) is consistent with the common defense and security, (4) has special circumstances that outweigh the potential decrease in safety because of reduced standardization, and (5) does not significantly reduce the level of safety at the applicant's facility. The staff has also determined, pursuant to Section VIII.C.4 of Appendix D to 10 CFR Part 52 , that the generic TS portion of the exemption request: (1) is authorized by law, (2) presents no undue risk to the public health and safety, (3) is consistent with the common defense and security, and (4) demonstrates the existence of special circumstances. Therefore, the staff grants the applicant an exemption from the requirements of Tier 1 Tables 2.5.2-3, 2.5.2-4, 2.2.5-4, and 2.2.5-1 and generic TS 3.3.2, TS Table 3.3.2-1, TS 3.7.6, and TS SRs 3.7.6.3, 3.7.6.8, and 3.7.6.12.

B. Technical Evaluation of Exemption Request and Departure

B.1 Main Control Room Temperature and Humidity

To maintain conditions in the control room within limits for reliable human performance and maintain equipment within qualified limits, the applicant proposed changes to the calculated heat loads, as well as changes to the acceptance criteria for conditions resulting in no restrictions to stay times for operators. Because in events where the MCR is isolated—for instance, because of radiological conditions exceeding the VES actuation setpoint or having both trains of VBS out of service at the onset of an accident—and VES is actuated, but offsite power is available to power other plant equipment, the heat loads in the MCR exceed those set forth in the certified design. The applicant's proposed changes to rectify this issue are evaluated below.

<u>FSAR Tier 1 Departure</u>

FSAR Tier 1, Section 2.2.5, "Main Control Room Habitability System," provides a functional description of the MCR VES. This includes a limit on the heat-up of the MCR, instrumentation and control (I&C) equipment rooms, and dc equipment rooms to provide assurance that acceptance criteria for reliable human performance and equipment qualification are not exceeded. This is accomplished by limiting the heat loads in these rooms to values specified in FSAR Tier 1, Table 2.2.5-4. The proposed departure includes changes to the table for the values in the control room based on the new load shedding scheme and expectation of the as-installed heat loads, including operators. The staff finds this change acceptable, given that the proposed limiting heat loads are reflected in the GOTHIC analysis (discussed further below) and that the values in Table 2.2.5-4 will be confirmed as limiting in the as-built design by ITAAC 7.c in Table 2.2.5-5. In addition, these values correspond with the changes to FSAR Tier 2, Table 6.4-3.

<u>FSAR Tier 2 Departure</u>

In a letter dated November 12, 2015, the applicant proposed to change the acceptance criteria for acceptable conditions for control room habitability from the effective temperature of 85 °F (29 °C) in the certified AP1000 design to a wet bulb globe temperature of less than 90 °F (32 °C) in the LNP FSAR. The wet bulb globe temperature (WBGT) is defined as 0.7 times the natural wet bulb temperature of the air plus 0.3 times the dry bulb temperature of the air. The WBGT stay-time criteria, defined in NUREG-0700, was referenced by the applicant. The staff considered that, according to NUREG-0700, Table 12.6, at less than 90 °F (32 °C) WBGT, there is no stay time limit if workers are performing low-metabolism work. The temperature ranges in Table 12.6 are intended to minimize performance decrements and potential harm to workers because of excessive heat. These temperature ranges are ceiling values (i.e., they assume that protective practices, such as acclimatization, training, and a cool place to rest, are in place). Further discussion related to this topic is located in the "Impact of control room habitability changes on operator performance" subsection presented below.

The staff views an unlimited stay time as an appropriate method for meeting the GDC 19 requirement to permit operators to occupy the control room under accident conditions. The other aspect required by GDC 19, adequate protection for equipment, is addressed via maintaining MCR conditions under those specified in revised FSAR Figure 3D-201, "Typical Abnormal Environmental Test Profile: Main Control Room (Sheet 1 of 3)," which the applicant identified as a departure from AP1000 DCD Figure 3D.5-1, Sheet 1 of 3. The staff's review of the applicant's analysis justifying that limits for reliable human performance and equipment qualification, following the limiting DBA conditions, is below, and is divided into two parts: the first 72 hours, during which the VES system operates to provide air to the main control room, and post-72 hours, when ancillary fan(s) are placed in operation to ventilate the MCRE.

<u>*First 72 hours*</u>

As discussed earlier, the heat loading values in FSAR Tier 2, Table 6.4-3, have been changed to correspond with the new load shedding design and revised LNP FSAR heat loads expected in the MCR for the limiting DBA with ac power still available. The staff reviewed the GOTHIC calculations supporting the temperature evaluation, and the revised heat loads including the new timing resulting from the load shed are reflected in the GOTHIC analyses.

The applicant's GOTHIC heat load analyses calculated MCR and I&C equipment room temperatures during a DBA. The temperature and RH values calculated during the 72 hours following a DBA with ac power available equate to a maximum average WBGT index for the control room of less than 90 °F (32 °C). The 90 °F (32 °C) WBGT index is the design limit for minimizing performance decrements and potential harm, and preserving well-being and effectiveness of the control room staff for an unlimited duration. Under the load shed, non-1E MCR heat loads are de-energized by automatic actions of the protection and safety monitoring system (PMS) within 3 hours after VES is actuated, and the 24-hour battery heat loads are terminated or exhausted at 24 hours to maintain the assumed heat load values, which then maintain the occupied zone of the MCR and the zones containing qualified safety-related equipment within the temperature constraints at 72 hours following VES actuation. The occupied zone is considered to be the area between the raised floor and 7 ft (2.13 m) above the floor, which encompasses the reactor operators and senior reactor operator consoles. In the event that power to the VBS is unavailable for more than 72 hours, MCR habitability is maintained by operating one of the two MCR ancillary fans to supply outside air to the MCR. Discussion of the post-72-hour conditions can be found below in the "Post 72 hours" subsection below. These conditions are reflected in the GOTHIC model, which was audited by the staff.

The GOTHIC calculation used the following conservatisms:

- *Finned surfaces areas are conservatively reduced to account for construction tolerances and embedments in the as-built design that could inhibit the heat transfer from the fins*
- *Heat transfer is conservatively calculated to account for thermal resistances associated with coatings and fouling (minimal fouling is expected over the life of the plant)*
- *Initial room temperatures are conservatively initialized above expected conditions*

Related to the above, the applicant revised the FSAR to include new TS surveillance requirements (and changes to the associated TS Bases) for the rooms surrounding the MCR, as well as the I&C and dc equipment rooms, to verify the average temperature is less than 85 °F (29 °C). This is conservative with respect to the value used in the applicant's analysis and therefore is acceptable to the staff, as provisions to ensure that the initial values are *bounded, in concert with limits on the design heat loads, are necessary to meet GDC 4 (specifically, the aspect of maintaining operation under the environmental conditions associated with both normal operations and following a postulated accident).*

The applicant proposed to revise LNP FSAR Subsection 6.4.3.2 to state that the bounding initial values of temperature and RH in the MCR are 75 °F (24 °C)/60 percent. The temperature and RH values calculated during the 72 hours following a DBA equate to a maximum average WBGT Index for the control room of less than 90 °F (32 °C).

The humidity of the air in the MCR also represents an important parameter in the acceptance criteria of the WBGT and is not calculated in the applicant's GOTHIC analysis. The applicant instead calculated the moisture content in the MCR in a separate spreadsheet calculation. During the first 72 hours, the safety-related VES system supplies air to the MCR.

During the first 72 hours, the RH in the control room (and therefore the wet bulb temperature) is a function of the initial moisture in the room, any moisture input from heat loads in the room (e.g., the operators), and any moisture stored in the VES bottles. Uncertainty regarding the allowed level of moisture in the VES bottles led staff to ask RAI 09.04.01-1, as the DCD did not specify a moisture specification for the air stored in the VES bottles. This lack of a moisture specification had potential effects on both the MCR analysis for human performance limits and operability of the VES system under conditions that could lead to freezing of the VES regulator.

In the certified design, given a potential scenario where the VES moisture content was sufficiently high, the potential existed to cause freezing at the VES regulator because of the Joule-Thomson effect. The air stored in the VES bottles is at high pressure. It is expanded through a pressure regulator before being supplied to the main control room. During the expansion process, the air cools below the freezing point for water. At higher moisture contents (a higher dew point or wet bulb temperature), moisture could condense out of the air and form ice on the regulator, potentially inhibiting the expected flow of air from the VES system to the MCR. In addition, a higher moisture content input from the VES bottled air could result in humidity values in the MCR that may challenge the human performance acceptance criteria outlined above.

In a letter dated December 22, 2015, the applicant submitted a revised RAI response proposing revisions to the FSAR and the TS. The proposed changes to FSAR Sections 6.4.5.3 and 9.3.1.1.2, TS Surveillance 3.7.6.8, and the associated TS bases state that the air in the VES bottles will be supplied as ANSI/CGA-7.1 Quality Level E with a pressure dew point temperature not to exceed 40 °F at 3,400 psig (4.4 °C at 23.5 MPa) or greater. Adding a VES moisture specification to the licensing basis that requires a relatively low-pressure dew point (i.e., dry air) in VES prevents moisture from affecting proper operation of VES components, such as the pressure regulator, given that the VES temperatures are maintained in a temperature range of 60–80 °F (16–

27 °C) (from TS Bases Figure B3.7.6-2, "VES Operability Requirements") and the VES has insulated piping and components.

In addition, the applicant states that the moisture specification is conservative with respect to maintaining acceptable conditions for habitability in the MCR during the first 72 hours following a transient even with maximum occupancy in the MCR. The staff audited the calculation supporting the RH in the MCR with maximum occupancy. The applicant calculated the humidity content of the control room under limiting conditions with 11 operators and initial values of 75 °F (24 °C) and 60 percent RH, and found that humidity conditions in the control room asymptotically approach a roughly steady-state condition because control room air is exhausted at the same rate it enters the control room not long into the transient (as the control room does not continually increase in pressure). The staff audited the applicant's calculation, which showed the control room reached a limiting humidity content of approximately 78 °F (26 °C) wet bulb. Because the TS do not impose a limit on the humidity in the control room, the staff performed confirmatory calculations using initial values of 75 °F (24 °C), 100 percent RH with the limiting moisture content added by 11 operators to determine the effect of adding the small amount of moisture present in the bottles using a 40 °F (4.4 °C) pressure dew point at 3,400 psig (4.4 °C at 23.5 MPa). The staff calculated a dew point in the control room of approximately 79 °F (26 °C) wet bulb at 72 hours, less than the value of 80.1 °F (26.7 °C) assumed by the applicant in the submittal. Given the above discussion, staff finds the proposed changes to the air quality acceptable. The staff is tracking the revisions discussed above to the FSAR as ***LNP Confirmatory Item 21.3-1****.*

<u>Resolution of LNP Confirmatory Item 21.3-1</u>

LNP Confirmatory Item 21.3-1 is a commitment by the applicant to revise the LNP COL application to provide additional information in the FSAR as indicated in the letters dated November 12, December 11, and December 22, 2015, including information related to limiting moisture content in the VES bottled air. The staff confirmed that the LNP COL FSAR has been appropriately revised. As a result, LNP Confirmatory Item 21.3-1 is now closed.

<u>Post 72 hours</u>

After 72 hours, the bottled air in the VES system has been depleted. If no non-safety system recovery has taken place, one of two ancillary fans is placed in operation to blow approximately 1,500 cfm (42,475 lpm) of outside air through the MCR envelope such that the maximum average WBGT index for the control room is less than 90 °F (32 °C). Likewise, outside air is supplied to Division B and C I&C rooms in order to maintain the ambient temperature below the qualification temperature of the equipment. In an RAI response dated July 17, 2015 (ADAMS Accession No. ML15201A540), the applicant stated that beyond 7 days, if VBS is still not operable, offsite support is available to extend habitability system operations. As such, the post-72-hour analyses are performed for a four-day period beginning at 72 hours and ending at 7 days after the onset of the transient.

Operation of the ancillary fans results in conditions in the MCR closely resembling ambient outdoor air conditions. In a November 12, 2015, RAI response (ADAMS Accession No. ML15322A009), the applicant performed an MCR habitability analysis in GOTHIC using a diurnal outdoor air input, with a maximum of 101 °F (38.3 °C) and a minimum of 86 °F (30 °C) for the dry bulb temperature. The corresponding wet bulb temperature in the analysis was assumed to be a constant 82.4 °F (28.0 °C) for 4 days. The applicant stated 101 °F (38.3 °C) is the maximum normal temperature for the certified design (FSAR Tier 2, Table 2-1); this value corresponds to the 1 percent seasonal exceedance temperature (or 0.4 percent annual exceedance temperature) for sites referencing the AP1000. The staff has evaluated the applicability of these values to the LNP site and found them acceptable, and further discussion of the staff evaluation is located in Section 2.3 of this SER. The constant 82.4 °F (28.0 °C) wet bulb temperature is a bounding assumption with respect to the value of 80.1 °F (26.7 °C) corresponding wet bulb coincident with the maximum normal dry bulb temperature as reflected in FSAR Tier 2, Table 2-1. FSAR Tier 2, Sections 6.4.2, 9.4.1.1.2, and 9.4.1.2.3.1 have been revised to reflect that, post-72 hours, the ventilation system is designed to maintain the MCR below the limits associated with reliable human performance, as defined in the "Impact of Control Room Habitability Changes on Operator Performance," section of this SER, below, and the equipment qualification limits in DCD Figure 3D.5-1, Sheet 2 of 3, based on operation at the maximum normal site ambient temperature.

Using the temperature data discussed above, the applicant's analysis demonstrated that the MCR remained below a WBGT index of 90 °F (32 °C) during the 4-day period between 72 hours and 7 days. The staff reviewed the temperature input values and assumptions in the applicant's analysis and performed its own analysis to confirm the acceptability of the temperature inputs. The staff analysis consisted of reviewing data from National Weather Service stations near the Levy site. As part of its review, the staff identified the worst consecutive 4-day period with respect to the WBGT index, and compared this data set to the applicant's inputs and assumptions. The staff found that the applicant's analysis conservatively bounds the staff calculated WBGT index recorded near the site. In addition, in the staff's analysis, the staff found that the dry and wet bulb temperatures for the entirety of the 4-day period that resulted in the worst WBGT index were bounded by the applicant's assumption of a daytime peak of 101 °F (38.3 °C) with an 15 °F (8.3 °C) diurnal swing and a wet bulb temperature of 82.4 °F (28.0 °C).

In addition, the staff also identified the worst 1-hour period with respect to the WBGT index that was recorded at National Weather Service stations near the Levy site. The staff compared this data to the applicant's MCR habitability inputs and assumptions. Using the worst 1-hour data, the staff found that the applicant's peak conditions bound the staff calculated peak WBGT index recorded near the site.

The staff recognizes that the use of a WBGT index as an appropriate metric to assess MCR habitability consists of a calculation that combines the dry bulb and wet bulb temperatures using appropriate scaling factors. In the staff's review of

> *the worst recorded 1-hour WBGT index, an individual temperature input that contributed to calculating the WBGT index (i.e., wet bulb temperature) exceeded the assumed value in the applicant's analysis. However, when the wet bulb temperature was combined with the coincident dry bulb temperature to form the calculated WBGT index, the staff found that the WBGT index was bounded by the applicant's analysis.*

The staff reviewed temperature data for National Weather Service stations near the William States Lee site. Similar to the LNP review, the staff identified the worst consecutive 4-day period with respect to the WBGT index, and compared this data set to the applicant's inputs and assumptions. The staff found the values used in the applicant's analysis conservatively bounds the staff calculated WBGT index recorded near the site. In the staff's review of the worst consecutive 4-day period, the dry bulb temperature exceeded the daytime peak of 101 °F (38.3 °C) for a brief period of time. However, at all other times the temperature was lower, generally substantially so, than the diurnal curve used by the applicant. In all cases when the dry bulb temperature was combined with the coincident wet bulb temperature to form the calculated WBGT index, the applicant's analysis remained bounding.

In addition, the staff also identified the worst 1-hour period with respect to the WBGT index that was recorded at National Weather Service stations near the Lee site. The staff compared this data to the applicant's MCR habitability inputs and assumptions. Using the worst 1-hour data, the staff found that the applicant's peak conditions bound the staff calculated peak WBGT index recorded near the site. In the staff's review of the worst recorded 1-hour WBGT index, an individual temperature input that contributed to calculating the WBGT index (i.e., dry bulb temperature) exceeded the assumed value in the applicant's analysis. However, when the wet bulb temperature was combined with the coincident dry bulb temperature to form the calculated WBGT index, the staff found that the WBGT index was bounded by the applicant's analysis.

The following portion of this technical evaluation section is reproduced from Section 21.3.4 of the LNP COL application FSER:

> *Humidity in the control room after 72 hours is primarily a function of the initial humidity of the control room at 72 hours combined with the moisture content of the outside ambient air, as an ancillary fan operates to blow approximately 1500 cfm of air through the MCR and Division B and C I&C rooms. The FSAR was revised to state the fans are expected to maintain the environment in the MCR near the daily average outdoor air temperature. Operators inside the control room represent a substantially smaller contribution to the ambient humidity as compared to the case prior to 72 hours, given the flow rate through the MCR from the fans. As stated earlier, the applicant uses conservative values for the temperature and moisture content of the air.*
>
> *Finally, the applicant revised FSAR Figure 3D-201 to reflect the post-72-hour limits for equipment qualification to 110 °F (43.3 °C) with 35 percent RH at this temperature. This change results in different acceptance criteria for equipment qualification and human performance after 72 hours. In addition, staff audited an analysis performed by the applicant demonstrating that even in conditions where 101 °F (38.3 °C) outside air was input to the control room for the entirety of the period between 72 hours and 7 days, the limits in FSAR Figure 3D-201 were not*

exceeded. As such, based on the above discussion, staff finds the proposed change to the FSAR acceptable, as the applicant's analysis provides reasonable assurance that the requirements associated with GDC 2 (with respect to natural phenomena, including ambient conditions) and GDC 4 are met. The calculated dry bulb temperature in the control room in this analysis was lower than the equipment qualification curve in Figure 3D-201, demonstrating further margin as compared to the diurnal temperature analysis discussed above.

The applicant's calculation showed that the WBGT remains below the 90-degree F (32.2-degree C) index associated with unlimited stay times for the operators. Additionally, the temperatures remain within the bounds for equipment qualification specified in DCD Figure 3D.5-1, Sheet 2 of 3. Based on the above review, the conservatism used by the applicant, and the staff's confirmatory analysis, the staff believes that the applicant's control room temperature calculation is acceptable, and therefore meets NRC regulations as specified in GDC 2, GDC 4, and GDC 19.

B.2 Impact of Control Room Habitability Changes on Operator Performance

In response to an RAI on control room habitability dated October 10, 2014 (ADAMS Accession No. ML14283A522), the applicant submitted a response dated March 26, 2015 (ADAMS Accession No. ML15089A193) stating that:

The MCRE temperature profile contained in the DCD is incorrect because of the following errors:

(1) MCRE heat loads during operation with or without normal ac power sources exceed the values documented in the DCD.

(2) Analyses that were performed to support the DCD were non-conservative because these analyses assumed that:

- *VES actuation is always coincident with station blackout (SBO); however, MCRE heat load challenge is most severe during events that result in isolation of the control room with offsite power available.*
- *EDS batteries are exhausted at exactly 1 hour beyond minimum mission time when there is a high probability that these batteries would last considerably longer.*

These errors could result in the MCR becoming a limited tolerance hot zone according to the referenced licensing basis standard, MIL-STD-1472E. This results in a 2- to 4-hour stay time for control room personnel, as stated in the applicant's RAI response dated July 17, 2015 (ADAMS Accession No. ML15201A540).

In the applicant's RAI responses dated November 12, 2015 (ADAMS Accession Nos. ML15320A025, ML15320A028, and ML15322A009), the applicant proposed to change the acceptance criteria for control room habitability from the effective *temperature of 85 °F (29 °C) in the certified AP1000 design to a WBGT of less than 90 °F (32 °C) in the LNP FSAR. NUREG-0700, Table 12.6, "Ranges of WBGT for Different Ranges of Stay Times," was used by the applicant as the basis for stay time limits. In accordance with NUREG-0700, Table 12.6, at 90 °F (32 °C) WBGT or less under control room working conditions (low-activity levels, normal work clothing), there is no stay time limit. The temperature ranges in Table 12.6 are intended to minimize performance decrements and potential harm to workers because of excessive heat. These temperature ranges are ceiling values (i.e., they assume that protective practices, such as acclimatization, training, and a cool place to rest, are in place).*

The staff finds the change in licensing basis from MIL-STD-1472E to NUREG-0700 to be acceptable and confirmed that the change was incorporated into the FSAR. Both documents establish stay time limits above 90-degree F (32.2-degree C) WBGT with NUREG-0700 providing a more detailed set of limitations based on temperature, clothing, and work activity. NUREG-0700 is also the established NRC-approved standard for human factors guidance. The staff finds the change of acceptance criteria for control room habitability from the effective temperature of 85 °F (29 °C) in the certified AP1000 design to a WBGT of less than 90 °F (32 °C) in the LNP FSAR to be acceptable. The new limit, as did the old limit, maintains an unlimited stay time in the control room and provides reasonable assurance that operator performance will not be affected by the control room environment.

B.3 Addition of Load Shed

The safety-related PMS and post-accident monitoring (PAM) system in the certified AP1000 DCD, Revision 19, as modified by LNP DEP 6.4-2, were reviewed to meet the above regulatory requirements. Chapter 7 of AP1000 DCD, Revision 19, as incorporated by reference in the LNP COL application includes the certified PMS and PAM systems. However, in response to RAI Question 06.04-4 on the MCR heat-up concern, dated October 10, 2014, the LNP COL applicant proposed in a submittal dated March 26, 2015, two new safety-related load shedding panels with associated other components to receive commands from the PMS to de-energize some non-safety-related electrical loads in the MCR (ADAMS Accession Nos. ML14283A522 and ML15089A193). In the RAI response, the applicant also stated that the PAM system would be revised to include some status signals. The above design changes were assessed below by the staff to ensure the regulatory requirements in Section 21.3.3 of this SER are still met. In addition, in response to RAI Question 06.04-4 on the MCR heat-up issue, the applicant stated the environmental conditions in the MCR after a design-basis event are changed from the certified, original conditions of 95 °F (35 °C) and 70 percent RH to 115 °F (46.1 °C) and 35 percent RH for an extended time duration of 4 days. The above changes to the environmental conditions in the MCR were also evaluated below by the staff to ensure the related regulatory requirement on equipment qualification in Section 21.3.3 of this SER is still met for the safety-related I&C equipment located in the MCR.

In order for the safety-related main control room VES to maintain heat loads for the MCRE within design-basis assumptions to limit the heat-up of the MCR, the *applicant stated in response to NRC RAI Question 06.04-4 that two safety-related MCR load shedding panels containing Class 1E equipment will be added to automatically or manually de-energize some non-safety-related electrical loads in the MCR. The applicant also stated in response to NRC RAI Question 06.04-4 that automatic actuation of the two new MCR load shedding panels is added to the existing PMS VES system actuation signal for VES MCRE isolation, pressurization, and filtration on a high iodine or particulate MCRE air supply radioactivity signal or a loss of all ac power for longer than 10 minutes signal by the low Class 1E battery charger input voltage parameter. In addition, the existing manual actuation signal for VES MCRE isolation, pressurization, and filtration is added to the two new MCR load shedding panels. De-energized, non-safety-related electrical loads are separated into two stages (Stage 1 and Stage 2) to maximize the availability of some non-safety-related wall panel information system, which is de-energized with other Stage 2 loads. Timers controlling the de-energization of electrical loads in both Stage 1 and Stage 2 are internal to each MCR load shedding panel and actuate relays to de-energize the associated loads. Stage 1 loads are de-energized by both panels immediately after the timers in each load shedding panel receive the PMS VES system actuation signal. Stage 2 loads are de-energized by both load shedding panels within 180 minutes after the timers in each load shedding panel receive the PMS VES system actuation signal. Component Interface Modules (CIMs) in PMS Divisions A and C are provided to de-energize non-safety-related electrical loads powered by the two MCR load shedding panels. In the staff's evaluation, it was not clear in the response to NRC RAI Question 06.04-4 how the above proposed design changes meet the regulatory requirement for the single failure criterion, as required in Clause 5.1 of IEEE Std. 603-1991, for the two new load shedding panels. Hence, the staff issued RAI Question 07.03-1 requesting the applicant to provide design information to demonstrate its compliance with the single failure criterion. In the response to RAI Question 07.03-1, the applicant stated that either PMS Division A or C is capable of de-energizing the two new MCR load shedding panels. Each load shedding panel de-energizes separate, non-essential, non-safety-related electrical loads from both Stage 1 and Stage 2. Each MCR load shedding panel contains redundant load shedding relays and timers that are actuated by both PMS Divisions A and C; therefore, actuation of either PMS Division A or C de-energizes all required non-safety-related electrical loads. The staff found that the additional information submitted in the RAI response demonstrated the compliance with Clause 5.1 of IEEE Std. 603-1991 for the single failure protection.*

During the staff's evaluation, it was not clear in the response to NRC RAI Question 06.04-4 how physical separation and electrical isolation were achieved between the two safety-related MCR load shedding panels and non-safety electrical loads controlled by them. In addition, the description on how the non-safety-related electrical loads will be controlled by the two new MCR load shedding panels was not clear in the response to RAI Question 06.04-4. For example, in Section 3.0 of Enclosure 2 in its response to RAI Question 06.04-4, the applicant states that two redundant MCR load shedding panels are added. However, later it states that each panel de-energizes separate nonessential non-safety-related electrical loads. Therefore, in RAI Question 07.03-1 dated

May 20, 2015, the staff requested the applicant to demonstrate clearly how the proposed changes meet the regulatory requirements for separation and isolation between safety systems and other systems, as required in Clause 5.6.3 of IEEE Std. 603-1991 (ADAMS Accession No. ML15140A475). In its response dated July 16, 2015, the applicant stated that each of the two load shedding panels contains two independent, isolated, in-series sets of relay contacts, one controlled by PMS Division A and the other controlled by PMS Division C (ADAMS Accession No. ML15201A542). In the RAI response, the applicant also provided schematic diagrams showing how the control and feedback signals are designed. Power for the non-safety-related loads, which may be de-energized, passes through both sets of relay contacts in one of the two new load shedding panels. Spatial separation between PMS Division A and Division C within the panel and between Class 1E and non-Class 1E circuits on the two load shedding panels is also provided to meet the requirements of IEEE Std. 384 and Regulatory Guide 1.75, "Criteria for Independence of Electrical Safety Systems," in accordance with the certified AP1000 commitments and exceptions. The applicant also stated in its response that the non-Class 1E loads to be shed by the two MCR load shedding panels are isolated from each of the Class 1E PMS Divisions A and C through the use of two fuses in series. These fuses provide Class 1E to non-Class 1E isolation and PMS Division to Division isolation. The staff found that the additional design information and schematic diagrams provided by the applicant in its response to RAI Question 07.03-1 demonstrated compliance with the regulatory requirements in Clause 5.6.3 of IEEE Std. 603-1991 regarding separation and isolation between safety systems and other systems.

In response to NRC RAI Question 06.04-4, the applicant stated the PAM system will be revised to include the status of the two new MCR load shedding panels. However, the revised Table 7.5-1 provided in the response only identified the MCR electrical load status, which would be added as PAM parameters. The staff found there is an inconsistency in the above description on what new parameters will be added to the PAM system. Therefore, the staff issued RAI Question 07.03-1 requesting the application to clarify what parameters will be added to the existing PAM system. In its response dated July 16, 2015, the applicant stated that each load shedding panel provides feedback to the PMS through individual digital input and output for affirmative display of de-energization of non-safety MCR electrical load status on the primary dedicated safety panel. Two Stage 1 feedbacks and two Stage 2 feedbacks per Division (a total of eight signals) are provided. Each MCR electrical load status signal is reported as closed when the contactor is closed (and MCR loads are energized). When the contactor input is open, the PMS inverts the signal to report that the contactor is open (and MCR loads are de-energized). The staff found that the above additional design information clarified which new parameters will be added to the existing PAM system. Therefore, the staff found that the response to RAI Question 07.03-1 is acceptable to meet the regulatory requirements in GDC 13 for variables to be monitored.

The staff found that electrical loads to be shed includes non-safety-related electrical equipment, such as wall panel information system displays, office equipment, water heater, kitchen appliances, and non-emergency lighting. However, it does not include the non-safety-related, but important to safety *diverse actuation system equipment. Therefore, the staff found that the proposed changes do not affect the certified design in the AP1000 DCD, Revision 19, approach to diversity and defense-in-depth.*

Safety-related I&C equipment located in the MCR must meet the regulatory requirements on equipment qualification as entailed in Clause 5.4 of IEEE Std. 603-1991. Chapter 7 of AP1000 DCD, Revision 19, as incorporated by reference in the LNP COL application, includes description of the PMS hardware, which will use the approved Common Qualified (Common-Q) platform, as described in Topical Report WCAP-16097-P-A, Revision 2, "Common Qualified Platform Topical Report." Table 7-1 in Topical Report WCAP-16097-P-A identifies the environmental design requirements for the Common-Q equipment, which includes a maximum temperature at 120 °F (48.9 °C) and 95 percent RH, and a minimum temperature of 40 °F (4.4 °C) and 20 percent RH for a time duration of 12 hours. In response to NRC RAI Question 06.04-4, the applicant stated the potential environmental conditions in the MCR after a design-basis event need to be revised from 95 °F (35 °C) and 70 percent RH, to 115 °F (46.1 °C) and 35 percent RH for an extended time duration of 4 days (between 4th and 7th day after a design-basis event).[7] However, the response to NRC RAI Question 06.04-4, lacked discussion on how the safety-related Common-Q equipment, such as flat display panels, node boxes, AP1000 modems and their processors located in the MCR, is qualified for the changed environmental conditions and time duration. It was not stated in the response to NRC RAI Question 06.04-4 whether the qualification already conducted for the Common-Q platform equipment was to be credited for the COL application. Therefore, the staff issued RAI Question 07.01-1, dated October 1, 2015, requesting the applicant to demonstrate how the safety-related Common-Q equipment is qualified for the revised higher temperature with an extended time duration after a design-basis event (ADAMS Accession No. ML15275A000). The staff also requested the applicant to clarify whether the qualification conducted for the Common-Q equipment is credited for the LNP COL application, or if additional testing needs to be performed on safety-related Common-Q equipment in the MCR.

In its response to RAI Question 07.01-1 dated November 12, 2015, the applicant stated that qualification performed with the Common-Q platform is not utilized as the only basis for the environmental qualification for the AP1000 safety-related Common-Q equipment in the MCR (ADAMS Accession No. ML15320A022). Topical Report WCAP-16097-P-A provides a qualification basis for the Common-Q system as a whole, but is not specific to the MCR installation of the Common-Q equipment. The MCR safety-related I&C equipment is listed in Table 3.11-1 of the AP1000 DCD, Revision 19. According to AP1000 DCD Tier 2 Appendix 3D, "Methodology for Qualifying AP1000 Safety-Related Electrical and Mechanical Equipment," the safety I&C equipment in the MCR requires an equipment qualification data package to demonstrate environmental qualification. After the proposed changes in potential environmental conditions to 115 °F (46.1 °C) and 35

[7] Subsequent to the RAI response discussed here, the applicant decreased the proposed limit for the environmental conditions during the period between 72 hours and 7 days from 115 °F (46.1 °C) to 110 °F (43.3 °C).

percent RH post-72 hours, various test programs that environmentally qualified similar safety-related equipment were used to show the safety Common-Q equipment is qualified for the changed environmental conditions. No further additional testing is expected because these safety-related I&C components have been qualified in other test programs.[8] The equipment qualification data package for the Common-Q equipment in the MCR, which are lower-level design documents, is being updated to reflect the revised environmental conditions in the MCR and reference the evaluation performed to ensure the Common-Q equipment in the MCR remains qualified for the changed environmental conditions with an extended time duration. The staff found the additional design information provided by the applicant demonstrated compliance with Clause 5.4 of IEEE Std. 603-1991.

Based on the evaluation above on meeting regulatory requirements for protection and safety systems, the staff finds the design changes meet the requirements identified in 10 CFR 50.55a(h)(3) and GDC 13.

B.4 Impact of Load Shed on Operator Performance

To limit control room maximum temperature during VES operation, a two-stage load shed of selected MCR equipment is automatically initiated on a high iodine or particulate MCRE air supply radioactivity signal or a loss of all ac power for greater than 10 minutes. Select, non-safety loads are de-energized by the Stage 1 load shed, which occurs coincident with VES actuation. Consisting primarily of office equipment and non-battery-backed lighting, specific loads include:

- *large screen displays used for weather or plan of the day information*
- *water heater*
- *coffee machine*
- *refrigerator*
- *microwave*
- *dishwasher*
- *drinking fountain/icemaker*
- *site-supplied desktop computer, monitors, copy machine, printers*
- *normal ELS lighting (i.e., not battery-backed)*
- *convection heater (2)*
- *non-safety-related MCR area radiation monitor*

Additional non-safety-related loads de-energized by the Stage 2 load shed include the

- *local area network consoles*
- *wall panel information system (WPIS) Displays.*

This occurs 3 hours after the Stage 1 load shed.

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[8] Subsequent to the RAI response discussed here, the applicant decreased the proposed limit for the environmental conditions during the period between 72 hours and 7 days from 115 °F (46.1 °C) to 110 °F (43.3 °C).

The staff confirmed that the Stage 1 load shed, with the exception of normal lighting, does not affect operational decision making or plant control. The applicant stated in the July 1, 2015, supplement (ADAMS Accession No. ML15187A039) that the plant lighting system (ELS) in the control area will continue to be available throughout the event using Class 1E battery-backed power. This battery-backed lighting provides the necessary illumination for safe operation.

With battery-backed lighting available, the staff concludes the Stage 1 load shed does not affect operator performance.

The staff identified two concerns with the proposed Stage 2 load shed:

(1) The WPIS is credited with supporting teamwork, situational awareness, and command and control as part of the "control room design that reflects state-of-the-art human factor principles" required by 10 CFR 50.34(f)(2)(iii).

(2) It is not clear whether the plant would remain at power and for how long it would stay at power following the initiation of VES followed by the subsequent load shed.

The staff requested additional information on how the load shed affected these issues in RAI Letter No. 128, issued June 29, 2015 (ADAMS Accession No. ML15180A275). The applicant provided additional information addressing these issues in their RAI response dated August 5, 2015 (ADAMS Accession No. ML15219A202).

The July 1, 2015, supplement states that the Two-Stage Automatic Load Shed does not de-energize all non-safety equipment and that although the WPIS displays are de-energized, the information shown on these panels can be readily retrieved and displayed on any available console that is not de-energized. The consoles that are not de-energized are identified as:

- *shift manager office console*
- *senior reactor operator console*
- *reactor operator consoles (excluding business LAN)*

The staff concludes that the command and control and situational awareness functions are not significantly affected because the WPIS information is available to the control room personnel at their normal work station consoles, which are not de-energized. The information available on the WPIS is high-level, fundamental safety information that is available on the work station consoles typically at the first or second information level so information accessibility remains reasonably quick and simple. Also the safety-related consoles display the minimum inventory parameters that are used to monitor the status of critical safety functions and to manually actuate the safety-related systems that achieve these critical safety functions.

While the loss of the WPIS places additional emphasis on communications between operators, the staff concludes the control room communications are also not significantly affected. The normal conduct of operations for MCR communications includes repeat backs, status announcements, and independent verifications to minimize human error and are used for normal and abnormal operations. During normal operations these communication practices reinforce information made readily available to the control room team via WPIS. During abnormal operations, the same practices would supplement the information each operator has available at his control station and compensate for loss of the centralized information on WPIS.

Although the control room design is sufficiently diverse to compensate for loss of the WPIS information, the reduction in defense-in-depth strategy within the control room human factors design caused by the removal of common indications, instantly and simultaneously available to all control room personnel that supports analysis and decision making warrants a better understanding of the conditions under which the loss of WPIS would occur. The staff prepared the following table based on the August 5, 2015, RAI response.

Table 21.3-1. VBS/VES Functionality

	Scenario	***Response***	***Standby Diesel Generator (DG) Functionality***	***VBS Functionality***
1	*Station blackout*	*Rx trip; VES actuates 10 min after power loss; WPIS is de-energized 2 hours after power loss because of battery limit or immediately if non-safety EDS batteries are not functioning*	*None—Cannot be credited under definition of station blackout*	*VBS not functional, but after 72 hours, operators may be able to align the ancillary DG to the VBS fans*
2	*Loss of switchyard only (offsite power) with runback (rapid power reduction)*	*Rx power reduced to meet plant loads. VBS continues to operate.*	*Available but not needed*	*Fully functional*
3	*Loss of switchyard and turbine generator trip*	*Rx trip; VES 10-minute timer starts on loss of battery charger input voltage. If DGs not functional then plant is in a station blackout condition*	*Standby DG starts and provides power to VBS system*	*Fully functional on power from standby DG.*

	Scenario	*Response*	*Standby Diesel Generator (DG) Functionality*	*VBS Functionality*
4	*Spurious VES actuation because of component failures.*	*Simultaneous, independent failures actuate VES and isolate VBS. If repairs unsuccessful WPIS de-energized by auto load shed at 3 hours. Mode 3 required by TS about 26 hours from VES actuation. Exact time to shutdown is dependent on component(s) which failed.*	*No impact, failures assumed to be independent of power supply*	*After verification of plant condition, operators override VBS isolation and return system to service.*
5	*VBS isolation occurs because of simultaneous, independent component failures*	*Operator manually initiates VES. If VBS repairs unsuccessful, WPIS de-energized by auto load shed at 3 hours. Mode 3 required by TS about 26 hours from VES actuation.*	*No impact; failures assumed to be independent of power supply*	*System is unavailable*
6	*LOCA with fuel failure and leakage from containment. Offsite ac available.*	*Rx trip; High-1 setpoint shifts VBS to recirc mode. VBS designed to maintain MCR doses below GDC 19 limits during design-basis events.*	*Available but not needed*	*Fully functional*
7	*LOCA with fuel failure and leakage from containment. Offsite ac not available.*	*Rx trip; VES 10-minute timer starts. If DG not credited then plant is in a station blackout condition with LOCA.*	*Standby DG starts and provides power to VBS system; High-1 shifts system to recirc*	*Fully functional on power from standby DG.*
8	*LOCA with fuel failure and leakage from containment from adjacent plant.*	*High-1 setpoint shifts VBS to recirc mode. VBS designed to maintain MCR doses below GDC 19 limits during design-basis events.*	*Available but not needed*	*Fully functional*
9	*LOCA with fuel failure and leakage from containment from adjacent plant with concurrent, simultaneous, independent failure of two VBS recirculation trains on intact unit*	*High-2 actuates VES on intact unit. WPIS de-energized by auto load shed at 3 hours. Mode 3 required by TS about 26 hours from VES actuation.*	*No impact; failures assumed to be independent of power supply*	*System is unavailable*

In summary:

(1) If the VES actuation occurs from a loss of power the plant is in a station blackout condition and the WPIS would not be available regardless of the load shed feature. This condition was accepted as part of the AP1000 design certification. If power is available either from offsite or the standby diesel generator, then the VBS system remains functional and VES actuation is unnecessary. The VBS system is designed to maintain MCR doses below GDC 19 limits.

(2) If the VES actuation occurred because of spurious component failures or a valid High-2 actuation signal, then TS associated with room temperature limits would require a plant shutdown within 26 hours. These scenarios require multiple independent system or component failures to cause VES actuation.

Scenarios 4, 5, and 9 would be most limiting in that the unit continues at power for up to 26 hours followed by a plant shutdown. However, these scenarios assume multiple, independent failures occur. The incorporation of independent systems and components into a design is a defense-in-depth strategy credited to effectively minimize the scenarios being postulated. Therefore the staff concludes that there is reasonable assurance that Scenarios 4, 5, and 9 will not occur because of the low probability of concurrent independent failures. If they should occur, the MCR operating staff still has the information necessary to evaluate and diagnose plant condition and implement the necessary actions to place the plant in a safe condition. It should be noted that many of the scenarios evaluated above are beyond design requirements. They are being used to illustrate intersystem functionality and the defense-in-depth provided by the design and are not part of the applicant's design basis.

The combination of failures and/or events that would cause VES actuation are either beyond the design basis and already addressed in the station blackout regulation or require failure combinations that are beyond what regulation addresses because of their low probability of occurrence.

Regardless, should such a combination of events occur, the defense-in-depth strategy inherent to the control room design would be reduced. Given the limited time at power at which the condition exists, the fact that that time is governed by technical specifications, and that redundant information is readily available on each of the operator consoles the staff concludes there is reasonable assurance that the operators could complete the actions necessary to maintain plant safety. Accordingly, the staff finds that, given the low probability of events resulting in WPIS load shed and the availability of alternate indications, the WPIS load shed does not undermine the acceptability of the WPIS system under 10 CFR 52.34(f)(2).

B.5 Reclassification of VES-PL-V018 and VES-PL-V019 as Active Safety-Related Valves

This section evaluates provisions for the functional design, qualification (functional, seismic, and environmental), and IST for safety-related valves *identified in the LNP Units 1 and 2 request for exemption regarding MCR heat load.*

The staff reviewed the following proposed departures from DCD Revision 19 to verify that the appropriate provisions are specified for the design, qualification, and IST of valves VES-PL-V018 and VES-PL-V019.

FSAR Tier 1 Departures

DCD Tier 1, Section 2.2.5, "Main Control Room Habitability System," describes the design-related information for valves VES-PL-V018 and VES-PL-V019. The applicant proposed a departure from DCD Tier 1, Table 2.2.5-1, to add valves VES-PL-V018 and VES-PL-V019, and identified the design requirements as ASME Boiler & Pressure Vessel Code (BPV Code), Section III, and seismic Category I, with an active function as "Transfer Open." The proposed departure to DCD Tier 1, Table 2.2.5-1 also specifies that the valve design does not include remote operators, safety-related displays, or PMS controls.

DCD Tier 2, Section 3.9.3, "ASME Code Classes 1, 2, and 3 Components, Component Supports, and Core Support Structures," states that pressure retaining components classified as Class 1, 2, or 3, are constructed according to the rules of ASME BPV Code, Section III, Division 1. Also, DCD Tier 2, Section 3.10, "Seismic and Dynamic Qualification of Seismic Category I Mechanical and Electrical Equipment," describes seismic qualification requirements for seismic Category I valves.

The staff finds the applicant's proposal to add valves VES-PL-V018 and VES-PL-V019 to DCD Tier 1, Table 2.2.5-1, to be acceptable because it includes the correct identification of the design criteria for the valves. The valves are designed and constructed in accordance with ASME BPV Code, Section III, requirements to withstand seismic design-basis loads without a loss of safety function to transfer open. Therefore, provisions are specified to meet the design and construction requirements of GDC 1 and the design requirements to withstand the effects of natural phenomena requirements of GDC 2. The valves are located in Environmental Zone 7 of the auxiliary building (not in the MCR itself), and are accessible for manual operation during normal, abnormal, and accident conditions as identified in Tables 3D.5-1, 3D.5-4, and 3D.5-5 of DCD Tier 2, and therefore do not require automatic operators.

FSAR Tier 2 Departures

The capability provisions for valves VES-PL-V018 and VES-PL-V019 are specified in DCD Tier 2, Section 3.9.3.2.2, "Valve Operability." DCD Tier 2, Section 3.9.3.2.2 states that prior to installation, qualification of the functional capability of active valve assemblies is performed in accordance with the requirements of ASME Standard QME-1-2007, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants," and that Tier 2, Table 3.9-12, "List of ASME Class 1, 2, and 3 Active Valves," identifies the active valves in the AP1000 design. The applicant proposed a departure to add valves

VES-PL-V018 and VES-PL-V019 to FSAR Tier 2, Table 3.9-12, and to classify the valve function as active.

The staff finds the applicant's proposal to reclassify the function of valves VES-PL-V018 and VES-PL-V019 in DCD Tier 2, Table 3.9-12, from inactive valves to "active valves" to be acceptable because it is consistent with the active safety-related function of the valves, and provides identification of the functional qualification requirements in accordance with the provisions of ASME QME-1-2007 where implemented as accepted in NRC Regulatory Guide 1.100, "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants" (Revision 3).

The IST (including preservice testing) provisions for valves VES-PL-V018 and VES-PL-V019 are described in DCD Tier 2, Section 3.9.6, "Inservice Testing of Pumps and Valves." DCD Tier 2, Section 3.9.6, specifies that inservice testing of ASME BPV Code, Section III, Class 1, 2, and 3 valves is performed in accordance with the ASME OM Code as required by 10 CFR 50.55a(f), and that DCD Tier 2, Table 3.9-16, "Valve Inservice Test Requirement," identifies components subject to the IST program. Table 3.9.6 also identifies the method and frequency of inservice testing for each valve. The applicant proposed a departure from DCD Tier 2, Table 3.9-16, to add valves VES-PL-V018 and VES-PL-V019, and identified the following test requirements: (1) the valves are active manual valves with a safety-related mission to maintain closed, transfer open, and maintain open, (2) the valves are ASME BPV Code, Class 3 and ASME OM Code, IST Category B, and (3) the IST type is full stroke and the test frequency is 2 years.

The staff finds the applicant's proposal to be acceptable because the IST provisions are consistent with the requirements specified in ASME OM Code, Subsection ISTC, "Inservice Testing of Valves in Light-Water Reactor Nuclear Power Plants." The staff notes that leak testing and position indication testing per ASME OM Code, Subsection ISTC are not required because these valves are classified as Category B and do not have remote position indication.

The environmental qualification provisions for valves VES-PL-V018 and VES-PL-V019 are specified in DCD Tier 2, Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment." Section 3.11 states that mechanical components identified in DCD Tier 2, Table 3.11-1, "Environmentally Qualified Electrical and Mechanical Equipment," are qualified to perform their required functions under the appropriate environmental effects of normal, abnormal, accident, and post-accident conditions. For mechanical equipment, DCD Tier 2, Section 3.11, specifies two categories of components: (1) active equipment that performs a mechanical motion as part of its safety-related function, and (2) non-active equipment whose only safety function is to maintain its structural integrity. For active components, the environmental qualification program is based on a combination of design, test, and analysis of critical sub-components, which is supported by maintenance and surveillance programs. For non-active equipment, the only safety-related function is to maintain the structural integrity according to the ASME BPV Code, Section III. The applicant proposed a *departure from DCD Tier 2, Table 3.11-1, to reclassify the function of valves VES-PL-V018 and VES-PL-V019 from "non-active valves" to "active valves."*

The staff finds the applicant's proposal to be acceptable because reclassification of the valves VES-PL-V018 and VES-PL-V019 in DCD Tier 2, Table 3.11-1, from "non-active valves" to "active valves" is consistent with the active safety-related function of the valves, and provides identification of the environmental qualification requirements associated with active valves. Therefore, provisions are specified to meet the environmental requirements of GDC 4. Valves VES-PL-V018 and VES-PL-V019 are located in Environmental Zone 7 (auxiliary room). In addition, other mechanical equipment listed in DCD Tier 2, Table 3.11-1, and located in Environmental Zone 3 (MCR) is required to be environmentally qualified to the revised test profile identified in FSAR Figure 3D-201. Use of this revised test profile for environmental qualification is acceptable to the staff because it is consistent with the environmental assumptions for the location.

DCD Tier 2, Appendix 3I, "Evaluation for High Frequency Seismic Input," states that the seismic analysis and design of the AP1000 plant is based on the Certified Seismic Design Response Spectra (CSDRS). Ground Motion Response Spectra (GMRS) for some Central and Eastern United States rock sites show higher amplitude at high frequency than the CSDRS. Appendix 3I describes the methodology and criteria to evaluate equipment that might be sensitive to the high-frequency input. Equipment that is not sensitive to high frequency input is listed in DCD Tier 2, Table 3I.6-3, "List of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive," and does not require high frequency evaluation per Appendix 3I. The applicant proposed a departure to classify valves VES-PL-V018 and VES-PL-V019 as being "not high frequency sensitive," and added the valves to FSAR Tier 2, Table 3I.6-3.

The staff finds the applicant's proposal to classify valves VES-PL-V018 and VES-PL-V019 as "not high frequency sensitive," and add the valves to Tier 2, Table 3I.6-3, to be acceptable because the valves are not within the high frequency sensitive criteria listed in Tier 2, Table 3I.6-1, "Potential High Frequency Sensitive Equipment List." The criteria include attributes such as: (1) equipment or components with moving parts that are required to perform a switching function during the seismic event, and (2) components with moving parts that may bounce or chatter, such as relays and actuation devices.

The staff concludes that the LNP proposed departure to DCD, Revision 19, to reclassify valves VES-PL-018 and VES-PL-019 from non-active valves to active valves is acceptable because the applicant specified appropriate provisions for the design, qualification, and IST of valves VES-PL-V018 and VES-PL-V019 and meets NRC regulations as specified in GDC 1, GDC 2, GDC 4, and 10 CFR 50.55a(f).

B.6 Technical Specifications

In a letter dated March 26, 2015, the applicant submitted its response to RAI Letter 122, Question 06.04-4, related to a revised Auxiliary Building heat-up *analysis to adequately support the safety function of the VES. This revised analysis results in modification of the VES design to add two new safety-related load-shed panels to allow automatic shutting off of various non-safety electrical loads during certain design-basis events, and a need to monitor the initial air temperatures in the MCRE as well as in selected adjacent rooms around the MCRE. These modifications result in changes in a few sections of the TS and TS Bases (Chapter 16) in the COL application.*

In letters dated July 17 and November 12, 2015, the applicant submitted its responses to follow-up RAI Letter 126, Question 16-3, and RAI Letter 134, Question 16-4, to address the staff's concerns related to proposed TS requirements and insufficient level of details provided in the TS Bases. Also, in its response letter dated December 22, 2015, to RAI Letter 132, Question 09.04.01-1, regarding the freezing issue in the VES air distribution lines, the applicant proposed changes to existing SR 3.7.6.5 (renumbered as SR 3.7.6.8) to address the potential high-moisture content of the air stored in the VES storage tanks.

These changes are necessary to ensure that the TS and TS Bases accurately reflect the updated design and are described below, with deleted text lined out and added text underlined.

- *LCO 3.3.2 (engineered safety features actuation system (ESFAS) Instrumentation)*

Required Action F.2.2 and Function 20 in Table 3.3.2-1 are revised to include the actuation of the new MCR Load Shed function as follows (with added text underlined):

The description of Function 20 is revised to read "Main Control Room Isolation, Air Supply Initiation, and Electrical Load De-energization" including a minor editorial correction for the input sensor description to read "a. Main Control Room Air Supply Radiation – High-2"

Required Action F.2.2 is revised to read "[V]erify main control room isolation, air supply initiation and electrical load de-energization manual controls are OPERABLE"

- *Applicable Safety Analyses, LCOs, and Applicability (ASA) Section of TS Bases B3.3.2 (ESFAS Instrumentation)*

On page B3.3.2-45, the discussion of Function 20 is revised as follows (with deleted text lined out and added text underlined):

"Main Control Room Isolation, Air Supply Initiation, and Electrical Load De-energization

Isolation of the main control room and initiation of the VES air supply provides a ~~protected environment from which operators can control the plant following an~~

~~uncontrolled release of radioactivity~~ <u>breathable air supply for the operators following an uncontrolled release of radiation. De-energizing non-essential main control room electrical loads maintains the room temperature within habitable limits</u>. This Function is required to be OPERABLE in MODES 1, 2, 3, and 4, and during movement of irradiated fuel because of the potential for a fission product release following a fuel handling accident, or other DBA.

20.a. <u>Main</u> Control Room Air Supply Radiation – High 2"

- *Actions Section of TS Bases B3.3.2 (ESFAS Instrumentation)*

On pages B3.3.2-55 and 57, in the first and second paragraphs under Actions F.1, F.2.1, and F.2.1 and in the second paragraph under Action K.1, the phrase "main control room isolation and air supply initiation" is revised as follows (with deleted text lined out and added text underlined):

"Condition F is applicable to the Main Control Room (MCR) isolation, ~~and~~ air supply initiation<u> and electrical load de-energization</u> function which has only two channels of the initiating process variable …"

"Alternatively, radiation monitor(s) which provide equivalent information and main control room isolation, ~~and~~ air supply initiation<u> and electrical load de-energization</u> manual controls may be verified to be OPERABLE ..."

"Condition K is applicable to the Main Control Room Isolation, ~~and~~ Air Supply Initiation,<u> and Electrical Load De-energization</u> (Function 20), during movement of irradiated fuel assemblies ..."

The staff finds the above proposed changes to TS LCO 3.3.2 and its associated bases acceptable because they reflect the change in the VES actuation logics described in FSAR Chapter 7.

- *LCO 3.7.6 (VES)*

A new condition, required action, and its associated completion time are added to address failure of the MCR load-shed panels to perform their safety function, as follows:

Condition B which reads "<u>One PMS division inoperable in MCR load shed panel(s)</u>"

Required Action B.1 which reads "<u>Restore MCR load shed panel(s) to OPERABLE status</u>" with a Completion Time of "<u>7 days</u>"

A new condition, required action, and its associated completion time are added to address nonconformance issues with monitored air temperature in adjacent rooms around the MCRE, as follows:

Condition D which reads "Air temperature in one or more required rooms not within limit"

Required Action D.1 which reads "Restore air temperature of required room(s) to within limit" with a Completion Time of "24 hours"

A new surveillance requirement is added to monitor the air temperature in the adjacent rooms around the MCRE, as follows:

SR 3.7.6 3 which reads "[V]erify the air temperatures of required rooms are ≤ 85°F" with a Frequency of "24 hours"

A new surveillance requirement is added to verify the automatic response of the electrical load shed function, as follows:

SR 3.7.6.12 which reads "[V]erify the MCR load shed function actuates upon receipt of an actual or simulated actuation signal" with a Frequency of "24 months"

The existing SR 3.7.6.5 for the verification of air quality in the VES high-pressure storage tanks is revised to address the freezing of air distribution lines because of high relative humidity condition of air in the tanks, as follows:

"Verify that the air quality of the air storage tanks meets the requirements of Appendix C, Table C-1 of ASHRAE Standard 62 with a pressure dew point of 40°F or lower at 3400 psig or greater."

In addition, the order of all SRs is changed such that the one with the shorter Frequency would come first, and the one with the longer Frequency would come last to be consistent with the convention used in the STS.

- *Background Section of TS Bases B3.7.6*

On page B3.7.6-1, in the first paragraph, the last line is revised as follows (with added text underlined):

"… functional during an accident, via de-energizing (load shedding) non-essential, non-safety main control room (MCR) electrical equipment (e.g., wall panel information system displays, office equipment, water heater, kitchen appliances, and non-emergency lighting) and the heat absorption of passive heat sinks. The VES limits the maximum temperature in DC Equipment Rooms (12201, 12202, 12203, 12204, 12205, and 12207), I&C rooms (12301, 12302, 12304, and 12305), as well as the MCRE.

On page B3.7.6-2, the fourth paragraph is revised as follows (with deleted text lined out and added text underlined):

"~~Sufficient thermal mass exists in the surrounding concrete structure (including walls, ceiling and floors) to absorb the heat generated inside the MCRE, which is initially at or below 75°F~~The VES also provides emergency passive heat sinks for

the main control room (Room 12401), instrumentation and control rooms (Rooms 12301, 12302, 12304, and 12305), and dc equipment rooms (Rooms 12201, 12202, 12203, 12204, 12205, and 12207). Provided air temperatures in the rooms requiring monitoring are within their Surveillance Requirement limits, the VES passive heat sinks limit the temperature rise inside each room during the 72-hour period following VES actuation. Heat sources inside the MCRE include operator workstations, emergency lighting and occupants. ~~Sufficient insulation is provided surrounding the MCRE pressure boundary to preserve the minimum required thermal capacity of the heat sink. The insulation also limits the heat gain from the adjoining areas following the loss of VBS cooling~~."

On page B3.7.6-2, new 5th through 13th paragraphs are added as follows:

"During normal operation, temperatures in the main control room, instrumentation and control rooms, dc equipment rooms, Class 1E electrical penetration rooms, and adjacent rooms are maintained within a specified range by the VBS. As described in Section 9.4.1.2, the VBS consists of independent subsystems, including the main control room / control support area HVAC subsystem and the Class 1E Electrical Room HVAC subsystem. The Class 1E Electrical room HVAC subsystem is further divided into two independent subsystems, with one serving the Division A & C Class 1E electrical division rooms and the other serving Division B & D Class 1E electrical division rooms. Each independent subsystem serves its associated rooms with two redundant, 100 percent capacity equipment trains, maintaining temperatures within the specified range.

Surveillance limits are required for rooms which have limits on allowable temperature increase, and conservatively established for some adjacent rooms of the VES passive heat sinks. Monitoring the air temperature is required for the rooms with the following numerical designators: 12201, 12202, 12203, 12204, 12205, 12207, 12300, 12301, 12302, 12303, 12304, 12305, 12313, 12401, 12412, and 12501.

Initial temperatures assumed for remaining rooms modeled in the VES passive heat sinks analysis are selected to maximize ~~operational~~ flexibility in responding to abnormal conditions or equipment failures, while still maintaining sufficient margin below safety analysis limits.

Access corridors, stairwells, rooms separated by an air gap, and other rooms without significant heat loads are not monitored because these areas do not contain significant heat sources and their temperatures are assumed to match the connected spaces. The numerical designators for these unmonitored rooms are 12211, 12311, 12400, 12405, 12411, 21480, 40400, and Stairwells.

Initial temperatures assumed for remaining rooms are conservatively selected to match the outdoor ambient or do not have an appreciable impact on the analyses. The numerical designators of these unmonitored rooms are 12212, 12213, 12306, 12312, 12404, 12406, 12504, 12505, 12506, and Level 1 rooms.

Non-essential, non-safety MCR heat loads are de-energized by the PMS VES actuation signal, which is generated by the "Main Control Room Isolation, Air

Supply Initiation and Electrical Load De-energization" ESFAS function, to maintain the MCRE within habitable limits for 72 hours.

Upon receipt of a "Main Control Room Isolation, Air Supply Initiation and Electrical Load De-energization" ESFAS signal, PMS Divisions A and C energize associated redundant relays in each of the two safety-related electrical panels (VES-EP-01 and VES-EP-02). Energizing one set of relays in each panel disconnects non-safety related electrical power to the non-safety electrical loads in the MCRE. Energizing just one set of relays in one panel de-energizes non-safety loads associated only with that panel.

De-energized non-safety loads are separated into stage 1 and stage 2 to maximize the availability of the non-safety related wall panel information system which is deenergized with stage 2 ~~loads~~. Timers and associated relays, which actuate to deenergize the stage 1 and stage 2 non-safety loads, are internal to each safety-related load shed panel. Stage 1 loads are de-energized by both panels immediately after the timers in each panel receive the PMS VES system actuation signal. Stage 2 loads are de-energized by both panels within 180 minutes after the timers in each panel receive the "Main Control Room Isolation, Air Supply Initiation, and Electrical Load Deenergization" ESFAS signal.

OPERABILITY of two redundant divisions of MCR Class 1E load-shed relays and timers located in two safety-related panels is required to meet the single failure criteria. Each panel contains redundant load-shed relays and timers actuated by the two PMS divisions, such that actuation of either division de-energizes all required loads."

- *LCO Section of TS Bases B3.7.6*

On page B3.7.6-3, in the third paragraph, the phrase "[T]his includes components listed in SR 3.7.6.3 through 3.7.6.10" is changed to read "[T]his includes components monitored under surveillance requirements" to accommodate the renumbering of all SRs as mentioned above.

On page B3.7.6-3, a new paragraph is added after the fourth paragraph as follows:

"The initial MCRE temperature (75°F), DC Equipment and I&C Rooms, and required room temperatures (≤85°F) are initial conditions required to both meet the maximum MCRE temperature limit 72 hours after VES actuation, and to maintain DC Equipment and I&C rooms below the equipment qualification temperature limit throughout the duration of the postulated accidents."

On page B3.7.6-4, a new paragraph is added at the end of the LCO Section as follows:

"All PMS divisions in the two safety-related electrical panels are required to be OPERABLE, so that non-safety stage 1 ~~and~~ stage 2 MCR heat loads can be de-

energized by the VES system actuation signal within the required time. This maintains the MCR temperature within habitable limits."

- *Actions Section of TS Bases B3.7.6*

On page B3.7.6-4, a discussion of the new Action B.1 is added as follows:

"If one division of MCR load shed panel(s) is inoperable, all divisions of both MCR load shed panels must be restored to OPERABLE status within 7 days. In this condition, the OPERABLE unaffected division of the panels is capable of providing 100% of the load shed function.

A Completion Time of 7 days is permitted to restore the inoperable division of MCR load shed panel(s) to OPERABLE status before action must be taken to reduce power. The Completion Time of 7 days is based on engineering judgment, considering the low probability of an accident that would require VES actuation, and that the remaining panel division can provide the required load shed function.

As described in Subsection 6.4.2.3 of Ref.1, any component failure in a PMS division of the load shed panel(s) renders that division inoperable. If this failure affects only one PMS division, leaving the remaining division of PMS unaffected, including the associated power and control circuit, it renders the panel(s) inoperable, while still maintaining the full load shed function.

An event or action that impacts both PMS divisions in either panel does not maintain the full load shed function, and Condition G or H of LCO 3.7.6 would apply."

On page B3.7.6-5, a discussion of the new Action D.1 is added as follows:

"When the air temperature in one or more of the rooms requiring temperature monitoring is not within the required limit, action is required to restore it to within the limit. A Completion Time of 24 hours is based on ~~engineering~~ judgment, considering the low probability of an accident that would require VES actuation under the worst case temperature conditions. It is judged to be a sufficient amount of time allotted to correct the deficiency in the non-safety ventilation system before shutting down."

On pages B3.7.6-6 and 7, in the discussions of Actions E.1, E.2, and F.1 (renumbered G.1, G.2, and H.1), editorial corrections are made to reflect the renumbered applicable Conditions which use the specified action to exit the Modes of Applicability.

- *Surveillance Requirements Section of TS Bases B3.7.6*

On page B3.7.6-7, the discussion of SR 3.7.6.1 is revised to clarify that temperature of air in the return air duct can be used for the performance of this surveillance.

On page B3.7.6-7, a discussion of the new SR 3.7.6.3 (for monitoring of air temperature in the required adjacent rooms around the MCRE) is added as follows:

"<u>Using indication from temperature elements in each room, the air temperatures in the following rooms are checked at a Frequency of 24 hours: 12202, 12204, 12300, 12303, 12313, 12412, and 12501.</u>

<u>Using indication from temperature elements located in shared return air ducting, the air temperatures in the following rooms are checked at a Frequency of 24 hours: 12201/12301, 12203/12302, 12205/12305, and 12207/12304.</u>

<u>This is done to verify that the VBS is performing as required to maintain the initial conditions assumed in the safety analyses, and to show that the VES heat sinks provide adequate thermal capacity to limit the temperature increase in the MCRE, DC Equipment Rooms, and I&C Rooms from exceeding the allowable limits after VES actuation. The surveillance limit of 85°F is below the initial temperature assumed in the analysis.</u>

<u>The 24 hour Frequency is acceptable based on the availability of automatic VBS temperature controls, alarms and indication in the MCRE. Air temperatures may also be verified using local measurement</u>."

On page B3.7.6-10, a discussion of SR 3.7.6.5 (renumbered as SR 3.7.6.8) is revised as follows:

"Verification that the air quality of the air storage tanks meets the requirements of Appendix C, Table C-1 of ASHRAE Standard 62 <u>with a pressure dew point of 40°F or lower at 3400 psig or greater,</u> is required every 92 days. If air has not been added to the air storage tanks since the previous verification, verification may be accomplished by confirmation of the acceptability of the previous surveillance results along with examination of the documented record of air makeup. The purpose of ASHRAE Standard 62 states: "This standard specifies minimum ventilation rates and indoor air quality that will be acceptable to human occupants and are intended to minimize the potential for adverse health effects." Verification of the initial air quality (in combination with the other surveillances) ensures that breathable air is available for 11 MCRE occupants for at least 72 hours. <u>Verification of the pressure dew point ensures that no water will form in the line, eliminating the potential for freezing at the pressure regulating valve during VES operation. In addition, the dry air ensures the MCRE will remain below the maximum relative humidity to support the 90°F WBGT required for human factors performance</u>."

On page B3.7.6-10, a discussion of the new SR 3.7.6.12 (for automatic response of the new MCR load shed panels) is added as follows:

"Verification that the MCR load shed function actuates on an actual or simulated signal from each PMS Division is required every 24 months to ensure that the non-safety stage 1 and stage 2 MCR heat loads can be de-energized by the VES system actuation signal within the required time. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, to minimize the potential for adversely affecting MCR operations."

The staff finds the above proposed changes to TS LCO 3.7.6 and its associated Bases acceptable because the newly established TS requirements are consistent with guidance in the STS with regards to format and content, the specified completion times and SR frequencies are consistent with those in similar LCOs in the AP1000 TS that are specifically relevant to this modified VES design, and these revised and new TS requirements also reflect the modified VES design described in FSAR Sections 6.4 and 9.4.1.

Based on the above evaluation, and pending the staff's confirmation that the proposed revisions are incorporated in Part 4 of the LNP Units 1 and 2 COL application, the staff finds the proposed TS and Bases revisions meet the requirements of 10 CFR 50.36, "Technical specifications." The staff is tracking these revisions as ***LNP Confirmatory Item 21.3-1****.*

Resolution of LNP Confirmatory Item 21.3-1

LNP Confirmatory Item 21.3-1 is a commitment by the applicant to revise the LNP COL application to provide additional information as indicated in the letters dated November 12, December 11, and December 22, 2015, including changes to TS and TS Bases. The staff confirmed that the TS and TS Bases have been appropriately revised. As a result, LNP Confirmatory Item 21.3-1 is now closed.

B.7 Risk Results and Insights

This design departure does not alter the description of AP1000 design features relevant to human performance in the control room. It does not modify the plant-specific probabilistic risk assessment (PRA) model used for licensing. Consequently, there is no change to the risk profile described in the COL application or the risk insights concerning the control room AP1000 DCD Table 19.59-18, item 20. Instead, the change improves confidence in the validity of the reported risk results and insights. Consistent with DC/COL-ISG-3, "PRA Information to Support Design Certification and Combined License Applications," the plant-specific PRA remains acceptable to the staff.

21.3.5 Post Combined License Activities

There are no post-COL activities related to this section.

21.3.6 Conclusion

The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to the design change of the VES, and there is no outstanding information expected to be addressed in the WLS COL FSAR related to this section. As discussed above in the technical evaluation section, the staff finds the departure acceptable, as it meets the requirements associated with GDCs 1, 2, 4, 13, and 19, 10 CFR 50.34(f)(2)(iii); 10 CFR 50.55a(h)(3); and 10 CFR 50.55a(f).

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR is acceptable and meets the regulatory requirements and guidance discussed in Section 21.3.3 of this SER. The staff based its conclusion on the following:

- WLS DEP 6.4-2 is acceptable because the described changes permit the applicant to meet the licensing basis within the bounds of the updated licensing document.

21.4 Hydrogen Vent ITAAC

21.4.1 Introduction

The applicant requests a change to the AP1000 DCD Revision 19 information. The WLS COL application incorporates the AP1000 DCD by reference. The change involves a departure from DCD Tier 1 ITAAC as well as an associated DCD Tier 2 departure.

The applicant determined that the ITAAC described in Tier 1 Table 2.3.9-3 cannot be met by the certified design. Instead, the applicant requested to revise the ITAAC described in Tier 1 Table 2.3.9-3, Item 3, Acceptance Criterion iii. This ITAAC requires that 98 percent of the primary openings through the ceilings of the PXS valve/accumulator rooms in containment must be at least 19 feet (5.8 meters) away from the containment shell and all other openings must be at least 3 feet (0.9 meters) away.

The applicant also proposes to depart from Tier 2, Section 6.2.4.5.1, “Preoperational Inspection and Testing, Hydrogen Ignition Subsystem,” and Tier 2, Section 19.41.7, “Diffusion Flame Analysis.”

21.4.2 Summary of Application

DEC incorporated in WLS COL application, dated April 11, 2016 (ADAMS Accession No. ML16124A854), the same information that DEF incorporated into the LNP COL application related to the voluntary submittal of an exemption request and design change description for departure from the AP1000 DCD to address hydrogen vent ITAAC. The information was originally submitted in endorsement and exemption request letters dated January 29, 2016 (ADAMS Accession No. ML16034A062).

Tier 1 and Tier 2 Departure

The applicant included the following Tier 1 and Tier 2 departure from the AP1000 DCD:

- WLS DEP 6.2-1

WLS DEP 6.2-1 proposes to change the acceptance criteria to be applied to a specific ITAAC design commitment and associated inspection, test, or analysis in Tier 1 Table 2.3.9-3, Item 3 to establish consistency with the current detailed design of the plant. The ITAAC currently contained in the AP1000 DCD, Tier 1 Table 2.3.9-3, Item 3, for control of containment hydrogen concentration for beyond-design-basis accidents, was based on the original AP600 and AP1000 design. The applicant determined that changes during the development of the current detailed design have resulted in inconsistencies between the design and the ITAAC acceptance criteria for (1) the primary vent paths through the ceilings of the PXS valve/accumulator rooms and (2) the proximity of these paths to the containment shell.

The staff reviewed the applicant's request for an exemption. The request included changes to Tier 1 Table 2.3.9-3, Item 3. Additionally, the staff reviewed the Tier 2 changes for potential effects on safety functions and design criteria of the PXS valve/accumulator room vents as described in DCD Sections 6.2.4.5.1 and 19.41.7. Subsection A of this SER (below) shows the staff's regulatory evaluation of the exemption. Subsection B of this SER (below) shows the staff's technical evaluation of the exemption request and departure.

Below are the specific ITAAC and DCD changes the applicant included under WLS DEP 6.2-1.

- Tier 1, Table 2.3.9-3, Item 3, Acceptance Criteria iii, be revised to state:

  "The equipment access opening and CMT-A opening constitute at least 98% of vent paths within Room 11206 that vent to Room 11300. The minimum distance between the equipment access opening and containment shell is at least 24.3 feet. The minimum distance between the CMT-A opening and the containment shell is at least 9.4 feet. The CMT-B opening constitutes at least 98% of vent paths within Room 11207 that vent to Room 11300 and is a minimum distance of 24.6 feet away from the containment shell. Other openings through the ceilings of these rooms must be at least 3 feet from the containment shell."

- Tier 2, Chapter 6.2.4.5.1 Preoperational Inspection and Testing, Hydrogen Ignition Subsystem, second paragraph be revised to read:

  "Pre-operational inspection is performed to verify the location of openings through the ceilings of the passive core cooling system valve/accumulator rooms with respect to the containment pressure boundary. The primary openings are those that constitute 98% of the opening area. The primary openings in Room 11206 that vent to Room 11300 are the equipment access opening and CMT-A opening. These openings are verified to be a minimum distance of 24.3 feet and 9.4 feet, respectively, from the containment shell. The primary opening in Room 11207 that vents to Room 11300 is the CMT-B opening, which is verified to be a minimum distance of 24.6 feet away from the containment shell. Other openings through the ceilings of these rooms are verified to be at least 3 feet from the containment shell."

- Tier 2, chapter 19.41.7, “Diffusion Flame Analysis” the last two paragraphs should be revised to read:

  “In the event that ADS stage 4 fails to adequately direct hydrogen away from combined compartments, the compartment vents are designed to release the hydrogen at locations where it burns, but does not challenge the containment shell integrity.

  Vents from the PXS and CVS compartments to the CMT room are located away from the containment shell and containment penetrations. Access hatches to the subcompartments that are near the containment shell are covered and secured closed such that they will not open as a result of a pipe break inside the compartment. Therefore, hydrogen releases to the CMT room from the subcompartments have been shown to not challenge the containment integrity.”

This exemption request involves a departure from Tier 1 Table 2.3.9-3, with a Tier 2 involved departure. Therefore, these departures require NRC approval and are evaluated below.

21.4.3 Regulatory Basis

The regulatory basis for evaluating the requested departures is provided by the applicable change processes in the AP1000 design certification rule. Departures from Tier 1 and Tier 2 requirements shall comply with Appendix D to Part 52, Design Certification Rule for the AP1000 Design, Section VIII, “Processes for Changes and Departures.” Specifically, the Tier 1 departure shall comply with the requirements for exemptions from Tier 1 information, which are governed by the applicable requirements in 10 CFR 52.63(b)(1) and 52.98(f). The Commission will deny a request for an exemption from Tier 1 if it finds that the design change will result in a significant decrease in the level of safety otherwise provided by the design. An applicant may depart from Tier 2 information without prior NRC approval, subject to the conditions of 10 CFR Part 52, Appendix D, Section VIII.B.5.

The regulatory guidance applicable for this technical evaluation is found in SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs,” issued April 2, 1993, and the corresponding staff requirements memorandum (SRM), issued July 21, 1993, Section I.J, “Containment Performance,” which states that the containment should maintain its role as a reliable, leak-tight barrier by ensuring that containment stresses do not exceed ASME Service Level C limits for a minimum period of 24 hours following the onset of core damage, and that following this 24-hour period the containment should continue to provide a barrier against the uncontrolled release of fission products.

21.4.4 Technical Evaluation

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff’s findings on standard content that were documented in the SER for the reference COL application (LNP Units 1 and 2) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the LNP COL FSAR, Revision 9 to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.
- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.
- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting.

The following portion of this technical evaluation section is reproduced from Section 21.4.4 of the LNP COL application FSER.

A. Regulatory Evaluation of Exemption Request

A.1 Summary of Exemption

The applicant requested an exemption from the provisions of 10 CFR Part 52, Appendix D, Section III.B that require the applicant referencing a certified design to incorporate by reference Tier 1 information. Specifically, the applicant proposed to revise Tier 1 Table 2.3.9-3, Item 3, Acceptance Criteria iii, to make it consistent with the current detailed design of the plant.[9]

A.2 Regulations

- *10 CFR Part 52, Appendix D, Section VIII.A.4 states that exemptions from Tier 1 information are governed by the requirements of 10 CFR 52.63(b) and 10 CFR 52.98(f). It also states that the Commission may deny such a request if the design change causes a significant reduction in plant safety otherwise provided by the design. This subsection of 10 CFR Part 52 Appendix D also provides that a design change requiring a Tier 1 change shall not result in a significant decrease in the level of safety otherwise provided by the design.*
- *10 CFR 52.63(b)(1) allows an applicant or licensee to request NRC approval for an exemption from one or more elements of the certification information. The Commission may only grant such a request if it complies with the requirements of 10 CFR 52.7, “Specific Exemptions,” which in turn points to the requirements listed in 10 CFR 50.12, “Specific Exemptions,” for specific exemptions. In addition, the*

[9] While the applicant describes the requested exemption as being from Section III.B of 10 CFR Part 52, Appendix D, the entirety of the exemption pertains to proposed departures from Tier 1 information in the generic DCD. In the remainder of this evaluation, the NRC will refer to the exemption as an exemption from Tier 1 information to match the language of Section VIII.A.4 of 10 CFR Part 52, Appendix D, which specifically governs the granting of exemptions from Tier 1 information.

special circumstances present outweigh the potential decrease in safety due to reduced standardization. Therefore, any exemption from the Tier 1 information certified by Appendix D to 10 CFR Part 52 must meet the requirements of 10 CFR 50.12, 10 CFR 52.7, and 10 CFR 52.63(b)(1).

A.3 Evaluation of Exemption

As stated in Section VIII.A.4 of Appendix D to 10 CFR Part 52, an exemption from Tier 1 information is governed by the requirements of 10 CFR 52.63(b)(1) and 10 CFR 52.98(f). Additionally, the Commission will deny an exemption request if it finds that the requested change to Tier 1 information will result in a significant decrease in safety. Pursuant to 10 CFR 52.63(b)(1), the Commission may, upon application by an applicant or licensee referencing a certified design, grant exemptions from one or more elements of the certification information, so long as the criteria given in 10 CFR 50.12 are met and the special circumstances as defined by 10 CFR 50.12 outweigh any potential decrease in safety due to reduced standardization.

The guidance of 10 CFR 50.12(a)(1) and 10 CFR 50.12(a)(2) provide the applicable criteria for when the Commission may grant the requested specific exemption. Section 50.12(a)(1) provides that the requested exemption must be authorized by law, not present an undue risk to the public health and safety, and be consistent with the common defense and security. The provisions of 10 CFR 50.12(a)(2) list six special circumstances for which an exemption may be granted. In order for NRC to consider granting an exemption request, at least one of these six special circumstances must be present. The applicant stated that the requested exemption meets the special circumstances of 10 CFR 50.12(a)(2)(ii). That subsection defines special circumstances as when "[a]pplication of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule." The staff's analysis of each of these findings is presented below.

A.3.1 Authorized by Law

This exemption would allow the applicant to implement approved changes to Tier 1 Table 2.3.9-3, Item 3. This is a permanent exemption limited in scope to particular Tier 1 information; subsequent changes to this information or any other Tier 1 information would be subject to full compliance with the change processes specified in Section VIII.A.4 of Appendix D to 10 CFR Part 52. As stated above, 10 CFR 52.63(b)(1) allows the NRC to grant exemptions from one or more elements of the certification information, namely, as discussed in this exemption evaluation, the requirements of Tier 1. The NRC staff has determined that granting the applicant's proposed exemption will not result in a violation of the Atomic Energy Act of 1954, as amended, or the Commission's regulations. Therefore, as required by 10 CFR 50.12(a)(1), the exemption is authorized by law.

A.3.2 No Undue Risk to Public Health and Safety

The underlying purpose of AP1000 Tier 1 Table 2.3.9-3, Item 3 is to ensure that in the postulated beyond-design-basis accident scenarios discussed in DCD Subsections 19.34 and 19.41, hydrogen generated as a result of the accident which migrates to the PXS compartments is vented through large openings in the ceilings of these rooms such that, in the event of ignition of the hydrogen plume, the containment shell will not fail.

A change to Tier 1 Table 2.3.9-3, Item 3, Acceptance Criteria iii, is necessary to establish consistency with the current detailed design of the plant by changing the ITAAC acceptance criteria for the primary ventilation paths through the ceilings of the PXS valve/accumulator rooms and the proximity of the paths to the containment shell. This change maintains the design margins of the Containment Hydrogen Control System; therefore, the change supports the intended design functions. The plant-specific Tier 1 DCD will continue to protect public health and safety and will maintain a level of detail consistent with that which is currently provided elsewhere in Tier 1 of the plant-specific DCD. The affected design description in the plant-specific Tier 1 DCD will continue to provide the detail necessary to support the performance of the associated ITAAC. In Section 21.4.4 of this safety evaluation, the NRC staff evaluates the proposed changes to Tier 1 information and finds them to be acceptable. Therefore, the staff finds the exemption presents no undue risk to public health and safety as required by 10 CFR 50.12(a)(1).

A.3.3 Consistent with Common Defense and Security

The proposed exemption would allow the applicant to implement modifications to the Tier 1 information requested in the applicant's submittal. This is a permanent exemption limited in scope to particular Tier 1 information. Subsequent changes to this information or any other Tier 1 information would be subject to full compliance with the change processes specified in Section VIII.A.4 of Appendix D to 10 CFR Part 52. This change is not related to security issues. Therefore, as required by 10 CFR 50.12(a)(1), the staff finds that the exemption is consistent with the common defense and security.

A.3.4 Special Circumstances

Special circumstances, in accordance with 10 CFR 50.12(a)(2)(ii), are present whenever application of the regulation in the particular circumstances would not serve the underlying purposes of the rule or is not necessary to achieve the underlying purpose of the rule. The underlying purpose of the specific Tier 1 Table 2.3.9-3, Item 3, Acceptance Criteria iii, modified in the exemption request, is to ensure that, in the postulated beyond-design-basis accident scenarios discussed in DCD Subsections 19.34 and 19.41, the following will happen: hydrogen generated as a result of the accident which migrates to the PXS compartments is vented through large openings in the ceilings of these rooms such that, in the event of ignition of the hydrogen plume, the containment shell *will not fail. A change to the ITAAC acceptance criteria is necessary to establish consistency with the current detailed design of the plant.*

Application of the requirements in Tier 1 Table 2.3.9-3, Item 3, Acceptance Criteria iii, as stated in the certified design, is not necessary to achieve the underlying purpose of those portions of the rule. The proposed change to the ITAAC acceptance criteria maintains the design margins of the Containment Hydrogen Control System, therefore supporting the intended design functions. This change does not impact the ability of any structures, systems, or components to perform their functions or negatively impact safety; therefore, the change meets the underlying purposes of the rule. Because application of the current requirements in Tier 1 Table 2.3.9-3, Item 3 is not necessary to achieve the underlying purpose of the rule, special circumstances are present. Therefore, the staff finds that special circumstances exist, as required by 10 CFR 50.12(a)(2)(ii) for the granting of an exemption from the Tier 1 information described above.

A.3.5 Special Circumstances Outweigh Reduced Standardization

This exemption, if granted, would allow the applicant to change certain Tier 1 information incorporated by reference from the AP1000 DCD into the LNP COL application. An exemption from Tier 1 information may only be granted if the special circumstances of the exemption request, required to be present under 10 CFR 52.7 and 10 CFR 50.12, outweigh any reduction in standardization. The proposed exemption would modify the ITAAC acceptance criteria for the primary ventilation paths through the ceilings of the PXS valve/accumulator rooms and the proximity of the paths to the containment shell. The proposed changes to the ITAAC acceptance criteria maintain the design margins of the Containment Hydrogen Control System, therefore supporting the intended design functions.[10]

As described below in the technical evaluation, the change to the ITAAC acceptance criteria for the primary ventilation paths through the ceilings of the PXS valve/accumulator rooms and the proximity of the paths to the containment shell is necessary to establish consistency with the description of the hydrogen ventilation paths in the current detailed design of the plant. While there is a small possibility that standardization may be slightly reduced by granting the exemption from the ITAAC acceptance criteria in Tier 1 Table 2.3.9-3, Item 3, the proposed exemption modifying the ITAAC acceptance criteria for combustible gas control will allow for application of acceptance criteria that are appropriate to evaluate a plant built according to the current detailed design. The proposed exemption modifying the ITAAC acceptance criteria for combustible gas control does not reduce the design margins of the Containment Hydrogen Control System and will result in no reduction in the level of safety. For this reason, the staff determined that even if other AP1000 licensees and applicants do not request similar

[10] Based on the nature of the proposed change to the Tier 1 Table 2.3.9-3, Item 3, Acceptance Criteria iii, and the understanding that this change is necessary to establish consistency with the current detailed design of the plant and does not impact the design function of the Containment Hydrogen Control System, other AP1000 licensees and applicants may request the same exemption, preserving the intended level of standardization.

departures, the special circumstances supporting this exemption outweigh the potential decrease in safety due to reduced standardization of the AP1000 design, as required by 10 CFR 52.63(b)(1).

A.3.6 No Significant Reduction in Safety

The proposed exemption would modify the ITAAC acceptance criteria for combustible gas control presented in the original application. As described below in the technical evaluation, the change to the ITAAC acceptance criteria for the primary ventilation paths through the ceilings of the PXS valve/accumulator rooms and the proximity of the paths to the containment shell is necessary to establish consistency with the current detailed design of the plant. Because the proposed change does not reduce the design margins of the Containment Hydrogen Control System, there is no reduction in the level of safety. Therefore, the staff finds that granting the exemption would not result in a significant decrease in the level of safety otherwise provided by the design, as required by 10 CFR Part 52, Appendix D, Section VIII.A.4.

A.4 Conclusion

The staff has determined that pursuant to Section VIII.A.4 of Appendix D to 10 CFR Part 52, the exemption: (1) is authorized by law, (2) presents no undue risk to the public health and safety, (3) is consistent with the common defense and security, (4) has special circumstances that outweigh the potential decrease in safety due to reduced standardization, and (5) does not significantly reduce the level of safety at the licensee's facility. Therefore, the staff grants the applicant an exemption from the requirements of Tier 1 Table 2.3.9-3, Item 3, Acceptance Criteria iii.

B. Technical Evaluation of Exemption Request and Departure

As discussed in Section 21.4.3 of this report, SECY-93-087 states that the containment should maintain its role as a reliable, leak-tight barrier by ensuring that containment stresses do not exceed ASME Service Level C limits for a minimum period of 24 hours following the onset of core damage, and that following this 24-hour period the containment should continue to provide a barrier against the uncontrolled release of fission products.

The purpose of the ITAAC in Tier 1 Table 2.3.9-3, Item 3 is to keep postulated diffusion flame sources away from the containment pressure boundary to mitigate potential for over temperature leading to failure of the containment shell, hatches, and penetrations.

The applicant's review of the assessment of the hydrogen diffusion flame locations and zones of influence for equipment survivability showed that a burning hydrogen plume from the passive core cooling system (PXS)-A compartment (Room 11206) to the core makeup tank (CMT)-A Room 11300 in the current detailed design could potentially challenge containment thermal limits.

The staff's technical evaluation is largely based on the following Westinghouse documents, which were reviewed during an audit conducted by the staff (ADAMS Accession No. ML15156B062).

- *WEC Document No. APP-VLS-M3C-008, Revision 0, "Hydrogen Diffusion Flame and Containment Integrity Analysis," dated October 15, 2015.*

- *WEC Engineering & Design Coordination Report No. APP-VLS-GEF-017, Revision 0, "Containment Structural Assessment for Hydrogen Venting," which includes Appendix A, "Structural Assessment for Equipment Survivability of the Containment Pressure Boundary during Diffusion Flame in CMT Compartment." Appendix A will be added to the APP-VLS-M3C-008 calculation.*

- *WEC Document No. APP-VLS-M3C-008, Appendix A, which calculates temperature distributions on the containment pressure boundary near the lower equipment hatch for a hydrogen diffusion flame from the PXS-A room vent exit to the CMT-A room. The temperature distribution will be input to a containment structural model to assess the containment pressure boundary severe accident survivability under the heat load of a hydrogen diffusion flame.*

- *WEC Document No. APP-VLS-M3C-007, Revision 0, "Thermal Analysis of Hydrogen Venting and Burning from the PXS-A compartment." This document describes a computational fluid dynamics (CFD) analysis which models a hydrogen diffusion flame in the CMT-A room that creates a containment wall temperature response. The CFD analysis, which models the hydrogen plumes exiting both the CMT-A opening and the floor hatch opening, shows that plume behavior is affected by the cutout for the equipment hatch in the CMT-A compartment ceiling. The hot plume is drawn toward the containment wall at the location of the lower equipment hatch, creating a hot spot. The applicant used the CFD analysis only as a sensitivity analysis and to identify non-conservative assumptions.*

B.1 Hydrogen Diffusion Flame and Temperature Distribution Evaluation

The applicant first performed a computational fluid dynamics (CFD) sensitivity analysis to evaluate location of hot spots and any flow split variation effects from the PXS-A room below. Using the insights gained from the CFD analysis, the applicant then performed a one-dimensional (1D) analysis to calculate temperature distributions on the containment pressure boundary in the CMT-A area near the lower equipment hatch for a hydrogen diffusion flame from the PXS-A room vents following a beyond design basis accident. This 1D calculation was based on first principle heat transfer and thermodynamic correlations. A conservative hydrogen plume temperature is calculated and the radiation and convection heat transfer is assessed to calculate a maximum containment wall temperature. The temperature distribution was then used as input to a *containment structural model to assess the containment pressure boundary severe accident survivability under the heat load from a hydrogen diffusion flame.*

The hydrogen venting scenario from the PXS-A room is for a beyond-design-basis event involving significant core damage and hydrogen generation due to fuel cladding oxidation. The scenario pertains to only one specific initiating event, a direct vessel injection (DVI) double-ended or large-line break which spills into the PXS-A compartment below the CMT room floor. The break must be large enough to defeat injection through the DVI line for the accident to progress to core damage. The PXS-B line must also fail to inject. Multiple failures of the ADS-4 valves must occur for the hydrogen generated in the core to reach the DVI line break and be released into the PXS-A compartment. This potential challenge applies only to a small subset of severe accident scenarios by frequency. The cut set frequency for this scenario, from the AP1000 probabilistic risk assessment (APP-GW-GL-022, Revision 8) is 6.4E-09/reactor-year.

The purpose of calculation APP-VLS-M3C-008 was to perform a simple heat transfer calculation independent of the CFD analysis, to calculate potential pressure boundary transients during a diffusion flame hydrogen burn in the CMT-A compartment for the bounding hydrogen release scenario described above. The source term for the hydrogen and steam from the PXS-A vents are from a Modular Accident Analysis Program (MAAP) analysis, referenced in APP-VLS-M3C-007.

The diffusion flame hydrogen temperature is calculated from the heat balance on the plume, which is modeled as a cylinder. The area for heat transfer to the containment wall is based on the hydraulic radius of the source, the distance from the source to the wall, and the height of the CMT-A compartment. The calculation assumed that the hydrogen igniters are operable and preventing global hydrogen combustion. The temperature distributions are based on the peak temperatures assuming that 100 percent of the hydrogen release is from the equipment access floor hatch. Sensitivity analyses in the CFD calculation showed that the hydrogen release from the floor hatch only produced the most challenging temperature results.

The APP-VLS-M3C-008, Appendix A, analysis creates two temperature distributions on the containment pressure boundary based on insights from the CFD analysis and identifies the location of maximum temperature, referred to as the hot spot. The first distribution, Temperature Distribution No. 1, assumes the plume creates a hot spot that spans the lower containment equipment hatch cover, the hatch barrel, the insert plate, and the containment shell. The second distribution, Temperature Distribution No. 2, locates the hot spot on the containment shell at the vent exit (opening in ceiling above the lower equipment hatch).

The hot spot is the local area where the hot plume impacts the containment pressure boundary. Heat transfer to the hot spot consists of radiation and convection from the hydrogen diffusion flame. Heat transfer to the containment shell away from the hot spot consists of radiation from the hydrogen diffusion flame. For the structural analysis, the allowable surface temperatures within the *hot spot are assumed to be the bounding temperature limits of the containment shell and the hatch door cover. For the hatch barrel hot spot temperature, where the hatch seals are located, the allowable average wall temperature is assumed to be the temperature limit of the* ethylene propylene diene monomer *(EPDM) rubber, and the corresponding surface temperature is reported.*

Zone 1 is the area of the containment pressure boundary above the hot spot in contact with the plume flow up the containment wall. The heat transfer consists of radiation and flat plate in parallel flow convection. Zone 2 is the area of the containment pressure boundary below the hot spot where the containment shell is not in contact with the plume flow but is receiving radiation from the plume.

Temperatures outside of Zones 1 and 2 are assumed unaffected and remain at 200 °F (93 °C). The calculations are performed to capture the maximum temperature on the inside surface of the heat sink in each region. The average temperatures in each region are also reported because the structural analysis uses the average through-wall temperatures for assessing integrity.

The peak surface and average temperatures from the limiting scenario identified by the sensitivity analysis for each of the zones are shown in the table below. The peak average through wall temperatures are assigned to the structural model. For Temperature Distribution No. 1, the temperatures were assigned as both a gradient from the hot spot outward to the base shell temperature and also as a constant value over the zone. Temperature Distribution No. 2 used the worst case from Temperature Distribution No. 1.

The component surface temperatures within each zone are calculated from these distributions.

Table 21.4-1 provides the results of the applicant's heat transfer calculations for Zone 1 and Zone 2 and compares them to the applicant's maximum allowable temperature for the hot spot. The results show that the applicant's calculated peak surface temperatures and peak average wall temperatures are below the allowable limits. The acceptability of the applicant's maximum allowable temperatures is discussed in Subsection B.2, below.

Table 21.4-1. Summary of Peak Temperature Results

	Peak Surface Temperature (°F (°C))		
Component	*Hot Spot Allowables*	*Zone 1=Radiation and Convection*	*Zone 2=Radiation Only*
CTMT shell	*650* (343)*	*470 (243)*	*436 (224)*
Insert Plate/Barrel	*488** (253)*	*366 (186)*	*344 (173)*
Hatch Cover	*800 (427)*	*591 (310)*	*543 (284)*

	Peak Average Wall Temperature (°F (°C))		
Component	*Hot Spot Allowables*	*Zone 1=Radiation and Convection*	*Zone 2=Radiation Only*
CTMT Shell	*607 (319)*	*442 (228)*	*411 (210)*
Insert Plate/Barrel	*390** (199)*	*308 (153)*	*293 (145)*
Hatch Cover	*780 (416)*	*577 (303)*	*530 (277)*

* *Allowable maximum temperature limit from ASME Code Service Level C for SA 738 Grade B.*

** *Allowable maximum temperature limit for insert plate/barrel corresponds to acceptance criterion for ethylene propylene diene monomer (EPDM) rubber.*

*The staff concludes that the methodology and assumptions in the analysis for determining the temperature source terms from the hydrogen burns are appropriately conservative, and the results are acceptable to be used as input to the structural analysis. The staff is tracking the proposed FSAR and ITAAC revisions proposed in the applicant's January 6, 2016, submittal, to be included in a future revision of the COL application, as **LNP Confirmatory Item 21.4-1**.*

<u>Resolution of LNP Confirmatory Item 21.4-1</u>

LNP Confirmatory Item 21.4-1 is a commitment by the applicant to revise the LNP COL application FSAR and ITAAC as indicated in the letter dated January 6, 2016, in areas related to combustible gas control. The staff confirmed that the LNP COL FSAR and ITAAC have been appropriately revised. As a result, LNP Confirmatory Item 21.4-1 is now closed.

B.2 Containment Structural Evaluation of Hydrogen Venting

The NRC staff considered FSAR, Revision 8, Section 3.8, "Design of Category I Structures" to perform the technical evaluation. The staff also considered portions of NUREG–1793, Supplement 2, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Plant Design" (ADAMS Accession No. ML112061231).

The applicant's January 6, 2016, submission identifies the actual design *distances between the PXS vents and the containment shell, including consideration of construction tolerances that pertain to the ITAAC in AP1000 DCD Tier 1 Table 2.3.9-3, Item 3. This submittal also contains proposed changes to AP1000 DCD Tier 2, Section 6.2.4.5.1, "Preoperational Inspection and Testing for the Hydrogen Ignition Subsystem," and Tier 2 Section 19.41, "Diffusion Flame Analysis." This section of the SER evaluates containment survivability and confirms that containment integrity is not challenged due to diffusion flame hydrogen burn in the containment compartments.*

In the letter dated January 6, 2016, the applicant discussed changes in the analytical approach for the heat transfer calculation and the analysis to confirm that the containment integrity was not challenged due to a diffusion flame hydrogen burn in the containment compartments. In the applicant's supporting analysis audited by the staff, the maximum allowable temperature of the local area at the lower equipment hatch cover (approximately 780 °F (416 °C)) exceeded the ASME NE-3000 maximum service temperature limit of 650 °F (343 °C). The applicant's supporting information audited by the staff provided further explanation of why the higher limit was acceptable. The temperature exceedance occurs at low containment pressure on order of 1.5 to 2.0 bar absolute. In order to assess the containment survivability of the hydrogen burning in the PXS-A compartment, the staff conducted an audit of the structural calculation (Westinghouse Document No. APP-VLS-GEF-017, Revision 0). As discussed above, the applicant's calculation developed two temperature distributions, each of which identified the location of a hot spot and two zones relative to the location of features on the containment shell. The calculation also performed sensitivity cases of the structural analysis. The applicant's results show Zone 1 and 2 are not affected by the hydrogen burn and remain below the service temperature limits. The hot spot area is a local area where burning plume flow impacts the containment pressure boundary. The hot spot area is about 2 meters in diameter and located on the equipment hatch at the top and covers the hatch barrel. For this hot spot, within the hatch barrel where the hatch seal is located, the peak allowable average wall temperature of 390 °F (199 °C) is based on the temperature limit of the EPDM rubber seal located within the hatch. The EPDM rubber is behind the 4-inch (10-cm) -thick lip of the hatch cover and, therefore, it is exposed to lesser temperature than the surrounding area of the hatch door. As shown in Table 21.4-1, above, the maximum average wall temperatures in Zone 1 and Zone 2 for the insert plate/barrel component are well below the applicant's 390 °F (199 °C) allowable limit.

Table 21.4-2, below, shows the applicant's calculation results of the stress analysis following ASME NE-3000, Service Level C code requirements for the containment vessel and hatch, which are fabricated from SA 738 Grade B steel.

Table 21.4-2. ASME Service Level C Limits

Location and Corresponding Maximum Allowable Temperature	*ASME Section 2, Part D Yield strength (Sy) for SA 738 Grade B*	*ASME Service Level C Allowable for SA 738 Grade B*
780 °F (416 °C)– Hot spot on equipment hatch	*42.4 ksi (292 MPa)*	*63.6 ksi (438 MPa)*
607 °F (319 °C)– Hot spot on containment shell	*46.3 ksi (319 MPa)*	*69.45 ksi (478.8 MPa)*

The applicant used an ANSYS finite element analysis (using software from ANSYS, Inc.) to calculate the maximum resultant stress intensity that would be experienced at the hot spot locations on the equipment hatch and containment shell. From the ANSYS stress analysis, the calculated maximum resultant stress intensity of 15.25 thousand pounds per square inch (ksi) (105.1 Megapascal (MPa) is less than ASME Service Level C allowable of 63.6 ksi (438 MPa).

Therefore, based on the presented results, the staff concluded that the applicant meets the Service Level C requirements of ASME Code, Section III, Division 1, Subsection NE-3230.

Further, during the staff audit, the staff discussed the containment metal creep values at peak average wall temperature with the applicant. The applicant presented to the staff results of the creep calculation that was based on EGG-EA-7431, "Creep Rupture Failure of Three Components of the Reactor Primary Coolant System during the TMLB Accident," published November 1986. Based on the creep calculation results, the time required to rupture at 800 °F (427 °C) is 6.3 E+07 hours and temperature required to rupture at stress level of 15.25 ksi (105.1 MPa) is 1291 °F (699 °C) for a 1-hour duration. Since the time at the elevated temperature exposed for the containment shell and hatch cover is short (less than 10 minutes) the staff concluded that the creep is not significant factor for the containment to rupture for the hydrogen burn event.

According to Regulatory Guide 1.216, "Containment Structural Integrity Evaluation for Internal Pressure Loadings Above Design Bases Pressure," regulatory position 2(b), an instability (buckling) calculation is not required for the steel containments. Therefore, buckling is not an issue for the hydrogen burn event.

Based on the staff's evaluation of containment survivability, discussed above, the staff finds that containment integrity is not challenged due to diffusion flame hydrogen burn in the containment CMT-A compartment from the PXS-A compartment because the containment meets the Service Level C requirements of ASME Code, Section III, Division 1 Subsection NE-3230 and Regulatory Guide 1.216. Therefore, the staff finds that applicant's FSAR and ITAAC revisions proposed in the January 6, 2016 submittal are acceptable. The staff is tracking these proposed FSAR and ITAAC revisions, to be included in a future revision of *the COL application, as **LNP Confirmatory Item 21.4-1**.*

Resolution of LNP Confirmatory Item 21.4-1

LNP Confirmatory Item 21.4-1 is a commitment by the applicant to revise the LNP COL application FSAR and ITAAC as indicated in the letter dated January 6, 2016, in areas related to combustible gas control. The staff confirmed that the LNP COL FSAR and ITAAC have been appropriately revised. As a result, LNP Confirmatory Item 21.4-1 is now closed.

B.3 Risk Results and Insights

This design departure does not materially alter the description of AP1000 design features that reduce the risk associated with generation of combustible gases. It does not modify the plant-specific probabilistic risk assessment model used for licensing. Consequently, there is no change to the risk profile described in the COL application or the risk insights concerning hydrogen control in AP1000 DCD Revision 19, Table 19.59-18, Item 31. Consistent with DC/COL-ISG-003, "PRA Information to Support Design Certification and Combined License Applications," the plant-specific PRA remains acceptable to the staff.

21.4.5 Post Combined License Activities

For the reasons discussed in the technical evaluation section above, the staff finds acceptable revised Acceptance Criteria iii, as part of DCD ITAAC Item 3 in DCD Table 2.3.9-3, reproduced below in Table 21.4-3.

Table 21.4-3. DCD ITAAC Item 3 from DCD Table 2.3.9-3, as revised by WLS DEP 6.2-1.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. The VLS provides the nonsafety-related function to control the containment hydrogen concentration for beyond design basis accidents.	i) Inspection for the number of igniters will be performed. ii) Operability testing will be performed on the igniters. iii) An inspection of the as-built containment internal structures will be performed. iv) An inspection will be performed of the as-built IRWST vents that are located in the roof of the IRWST along the side of the IRWST next to the containment shell.	i) At least 64 hydrogen igniters are provided inside containment at the locations specified in Table 2.3.9-2. ii) The surface temperature of the igniter exceeds 1700°F. iii) The equipment access opening and CMT-A opening constitute at least 98% of vent paths within Room 11206 that vent to Room 11300. The minimum distance between the equipment access opening and containment shell is at least 24.3 feet. The minimum distance between the CMT-A opening and the containment shell is at least 9.4 feet. The CMT-B opening constitutes at least 98% of vent paths within Room 11207 that vent to Room 11300 and is a minimum distance of 24.6 feet away from the containment shell. Other openings through the ceilings of these rooms must be at least 3 feet from the containment shell. iv) The discharge from each of these IRWST vents is oriented generally away from the containment shell.

21.4.6 Conclusion

The NRC staff reviewed the application and checked the referenced DCD, including the applicant's proposed changes in WLS DEP 6.2-1. The NRC staff's review confirmed that the applicant addressed the required information relating to the ITAAC and FSAR changes to be in conformance with the current detailed design while continuing to preserve the containment integrity. The staff concluded that the AP1000 containment will continue to maintain its role as a reliable leak-tight barrier in accordance with the containment performance regulatory guidance of SECY 93-087.

Based on the staff's technical evaluation documented above, the staff finds that the proposed change to allow short duration of the hydrogen burn temperature and pressure effect on the containment shell and equipment hatch with verification of the ITAAC distances from the containment shell is acceptable. The staff based its conclusion on the following:

- The methodology and assumptions used in the applicant's analysis for determining the temperature source terms from the hydrogen burns are appropriately conservative, and the result are acceptable to be used as input to the structural analysis.

- The containment meets the Service Level C requirements of ASME Code, Section III, Division 1 Subsection NE-3230 and Regulatory Guide 1.216, and the staff confirmed that the containment integrity is not challenged due to diffusion flame hydrogen burn in the containment compartment.

21.5 Source Range Neutron Flux Doubling Logic Operating Bypass

21.5.1 Introduction

The regulations in 10 CFR Part 50, Section 50.55a, "Codes and standards," cites certain standards published by the IEEE. According to 10 CFR 50.55a(h)(3), "Safety Systems," applicants for a COL must comply with IEEE Std. 603–1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," and the associated correction sheet dated January 30, 1995.

Operating bypasses are addressed in Clause 6.6 of the standard. Under certain conditions, it may be acceptable to bypass a safety function. All of the conditions that permit bypassing the function must exist before the bypass is activated. If an operating bypass has been activated and plant conditions change so that the bypass is no longer permissible, the safety system must automatically do one of three things: restore plant conditions so that bypass is permissible, remove the active bypass, or initiate the safety function.

In the AP1000 certified design, safety functions are initiated by the PMS. In Revision 19 of the AP1000 DCD, Chapter 7, all safety functions initiated by the PMS comply with IEEE Std. 603-1991, Clause 6.6, "Operating Bypasses," with one exception. The exception is the manually activated operating bypass of the safety function called the boron dilution block from the source range neutron flux doubling logic. The boron dilution blocking function is normally activated when neutron flux doubles too quickly while reactor power is in the source range. However, bypassing this block is permitted above a certain temperature when boron dilution can no longer lead to inadvertent criticality. The AP1000 design of the PMS flux doubling logic for the boron dilution block did not meet the operating bypass requirements of IEEE Std. 603-1991 because the logic programmed into the PMS did not include a permissive to allow the block of the flux doubling function under the appropriate conditions.

21.5.2 Summary of Application

DEC incorporated in WLS COL application, dated April 11, 2016 (ADAMS Accession No. ML16124A854), the same information that DEF incorporated into the LNP COL application related to the voluntary submittal of an exemption request and design change description for departure from the AP1000 DCD to address source range neutron flux doubling logic operating bypass. The information was originally submitted in endorsement and exemption request letters dated February 9, 2016 and September 29, 2015 (ADAMS Accession Nos. ML16041A586 and ML15274A134).

Tier 2 Departure

The applicant included the following Tier 2 departure from the AP1000 DCD:

- WLS DEP 7.3-1

WLS DEP 7.3-1 includes changes for the PMS source range neutron flux doubling logic to comply with the requirements of IEEE Std. 603-1991, Clause 6.6. The departure included changes to the FSAR and TS; and incorporated changes into Parts 2, 7, and 10 of the COL application.

This exemption request involves a departure from the generic TS Table 3.3.2-1, and Tier 2 involved departures. Therefore, these departures require NRC approval and are evaluated below.

21.5.3 Regulatory Basis

The regulations in 10 CFR 50.55a(h)(3) require compliance with IEEE Std. 603-1991, and the correction sheet dated January 30, 1995. Clause 5.1 of IEEE Std. 603-1991, “Single Failure Criterion,” requires, in part, that safety systems shall perform all safety functions required for a DBE in the presence of (1) any single detectable failure within the safety systems concurrent with all identifiable but nondetectable failures, (2) all failures caused by the single failure, and (3) all failures and spurious system actuations that cause or are caused by the DBE requiring the safety functions. Clause 6.6 of IEEE Std. 603-1991, requires that, whenever the applicable permissive conditions are not met, a safety system shall automatically prevent the activation of an operating bypass or initiate the appropriate safety function(s). If plant conditions change so that an activated operating bypass is no longer permissible, the safety system shall automatically accomplish one of the following actions: (1) remove the appropriate active operating bypass(es), (2) restore plant conditions so that permissive conditions once again exist, or (3) initiate the appropriate safety function(s).

The regulations in 10 CFR 52.79(a)(2) require, in part, that the description of the structures, systems, and components shall be sufficient to permit understanding of the system designs and their relationship to the safety evaluations.

The guidance of the Standard Review Plan (SRP) Appendix 7.1-C, “Guidance for Evaluation of Conformance to IEEE Std. 603,” Section 4, “Safety System Designation,” states that the information provided for the design-basis items, taken alone and in combination, should have one and only one interpretation.

21.5.4 Technical Evaluation

Section 1.2.3 of this SER provides a discussion of the strategy used by the NRC to perform one technical review for each standard issue outside the scope of the DC and use this review in evaluating subsequent COL applications. To ensure that the staff’s findings on standard content that were documented in the SER for the reference COL application (LNP Units 1 and 2) were equally applicable to the WLS Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the LNP COL FSAR, Revision 9 to the WLS COL FSAR. In performing this comparison, the staff considered changes made to the WLS COL FSAR (and other parts of the COL application, as applicable) resulting from RAIs.
- The staff confirmed that all responses to RAIs identified in the corresponding standard content evaluation were endorsed.
- The staff verified that the site-specific differences were not relevant.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the WLS COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting.

Tier 2 Departure

- WLS DEP 7.3-1

The following portion of this technical evaluation section is reproduced from Section 21.5.4 of the LNP COL application FSER.

> - *LNP DEP 7.3-1*
>
> *LNP DEP 7.3-1 proposes to make changes for the PMS source range neutron flux doubling logic to comply with the requirements of IEEE Std. 603-1991, Clause 6.6 (Operating Bypasses). The manual block of the source range neutron flux doubling logic portion of the boron dilution block logic in the AP1000 DCD, Revision 19, does not comply with the requirements contained in Clause 6.6 of IEEE Std. 603-1991, which require the PMS to accomplish one of the following actions if plant conditions change so that an activated operating bypass is no longer permissible: (1) automatically remove the appropriate active operational bypass(es), (2) automatically restore plant conditions so that permissive conditions once again exist, or (3) automatically initiate the appropriate safety functions.*
>
> *The staff reviewed a request for an exemption submitted by the applicant. The request proposed changes to generic TS Table 3.3.2-1. Additionally, the staff reviewed the associated changes to Tier 2 information, including DCD Chapters 7, 9, 14, 16, and 19. The regulatory evaluation of the exemption request appears in Subsection A, below, and the technical evaluation of the exemption request and departure appears in Subsection B, below.*
>
> *A. Regulatory Evaluation of Exemption Request*
>
> *A.1 Summary of Exemption*
>
> *The applicant requested an exemption from the provisions of 10 CFR Part 52, Appendix D, Section III.B, "Design Certification Rule for the AP1000 Design, Scope and Contents," that require the applicant referencing a certified design to incorporate by reference generic TS. Specifically, the applicant proposed to revise TS Table 3.3.2-1 by adding a P-8 permissive to the TS Table 3.3.2-1 for*

the ESFAS to provide reasonable assurance that the facility will be constructed and operated in conformity with the applicable design criteria, codes and standards.[11]

A.2 Regulations

10 CFR Part 52, Appendix D, Section VIII.C.4 states that an applicant may request an exemption from the generic TS or other operational requirements. The Commission may grant such a request only if it determines that the exemption will comply with the requirements of 10 CFR 52.7, "Specific Exemptions."

A.3 Evaluation of Exemption

As stated in Section VIII.C.4 of Appendix D to 10 CFR Part 52, the Commission may grant an exemption from generic TS of the DCD only if it determines that the exemption will comply with the requirements of 10 CFR 52.7. As stated above, Section 52.7 points to 10 CFR 50.12 for specific exemptions.

Applicable criteria for when the Commission may grant the requested specific exemption are provided in 10 CFR 50.12(a)(1) and (a)(2). Section 50.12(a)(1) provides that the requested exemption must be authorized by law, not present an undue risk to the public health and safety, and be consistent with the common defense and security. The provisions of 10 CFR 50.12(a)(2) list six special circumstances for which an exemption may be granted. It is necessary for one of these special circumstances to be present in order for NRC to consider granting an exemption request. The applicant stated that the requested exemption meets the special circumstances of 10 CFR 50.12(a)(2)(ii). That subsection defines special circumstances as when "[a]pplication of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule." The staff's analysis of each of these findings is presented below.

A.3.1 Authorized by Law

This exemption would allow the applicant to implement approved changes to TS Table 3.3.2-1. This is a permanent exemption limited in scope to particular generic TS, and subsequent changes to this information or any other generic TS would be subject to full compliance with the change processes specified in Section VIII.C.4 of Appendix D to 10 CFR Part 52. Section VIII.C.4 allows the NRC to grant exemptions from generic TS if the exemption meets the requirements of 10 CFR 52.7 and 50.12. The staff has determined that granting of the applicant's proposed exemption will not result in a violation of the Atomic

[11] Although the applicant describes the requested exemption as being from Section III.B of 10 CFR Part 52, Appendix D, the entirety of the exemption pertains to proposed departures from generic TS in the generic DCD. In the remainder of this evaluation, the staff will refer to the exemption as an exemption from generic TS to match the language of Section VIII.C.4 of 10 CFR Part 52, Appendix D, which specifically governs the granting of exemptions from generic TS.

Energy Act of 1954, as amended, or the NRC's regulations. Therefore, as required by 10 CFR 50.12(a)(1), the exemption is authorized by law.

A.3.2 No Undue Risk to Public Health and Safety

Design changes are required for the PMS source range neutron flux doubling logic to comply with the requirements of IEEE Std. 603-1991, Clause 6.6 on operating bypasses; these changes to the source range flux doubling logic therefore support the system's intended design functions. The change will enable the plant-specific TS to meet the requirements of IEEE Std. 603-1991 and therefore the TS will continue to protect public health and safety and will maintain a level of detail consistent with that which is currently provided elsewhere in the plant-specific TS of the plant-specific DCD. The proposed changes to generic TS are evaluated and found to be acceptable in Section 21.5.4 of this safety evaluation. Therefore, the staff finds the exemption presents no undue risk to public health and safety as required by 10 CFR 50.12(a)(1).

A.3.3 Consistent with Common Defense and Security

The proposed exemption would allow the applicant to implement modifications to generic TS requested in the applicant's submittal. This is a permanent exemption limited in scope to a specific TS. Subsequent changes to this information or any other generic TS would be subject to full compliance with the change processes specified in Section VIII.C.4 of Appendix D to 10 CFR Part 52. This change is not related to security issues. Therefore, as required by 10 CFR 50.12(a)(1), the staff finds that the exemption is consistent with the common defense and security.

A.3.4 Special Circumstances

Special circumstances, in accordance with 10 CFR 50.12(a)(2)(ii), are present whenever application of the regulation in the particular circumstances would not serve the underlying purposes of the rule or is not necessary to achieve the underlying purpose of the rule. The underlying purpose of TS Table 3.3.2-1 is to ensure compliance with the requirements of IEEE Std. 603-1991, Clause 6.6. Because TS Table 3.3.2-1 does not include the missing elements as described in the PMS source range neutron flux doubling logic, the proposed addition is needed to ensure that the plant-specific TS reflect the actual PMS design which meets the applicable requirements in IEEE Std. 603-1991. The additional TS requirements are needed so that the PMS source range flux doubling logic maintains the design margins of reactor startup protection.

Application of the requirements in TS Table 3.3.2-1 is not necessary to achieve the underlying purpose of those portions of the rule. The proposed changes to the PMS source range neutron flux doubling logic support the system's intended design functions, as does the proposed changes to the TS requirements. The system as modified in the requested exemption will continue to perform its intended functions and will, therefore, meet the underlying purposes of the rule. Accordingly, because application of the requirements in generic TS Table 3.3.2-1 is not necessary to achieve the underlying purpose of the rule, special *circumstances are present. Therefore, the staff finds that special circumstances exist, as required by 10 CFR 50.12(a)(2)(ii), for the granting of an exemption from generic TS described above.*

A.4 Conclusion

The staff has determined that, as required by Section VIII.C.4 of Appendix D to 10 CFR Part 52, the exemption: (1) is authorized by law, (2) presents no undue risk to the public health and safety, (3) is consistent with the common defense and security, and (4) has special circumstances. Therefore, the staff grants the applicant an exemption from the requirements of TS Table 3.3.2-1.

B. Technical Evaluation of Exemption Request and Departure

B.1 Operating Bypasses

Operating bypasses are usually included in the reactor safety I&C system design to permit some safety functions to be bypassed, so that normal plant operations can occur without actuating safety systems unnecessarily. The implementation of operating bypasses for safety functions are required to meet the requirements in Clause 6.6 of IEEE Std. 603-1991, which is required by regulation in accordance with 10 CFR 50.55a(h)(3).

The applicant has incorporated the AP1000 DCD for the LNP COL application. However, the applicant proposed this design change because it found that the design in the safety-related PMS for bypassing the source range neutron flux doubling logic input to the boron dilution block, which is a safety function as shown in Figure 7.2-1 (Sheet 3 of 21) in the AP1000 DCD, did not meet the criteria in Clause 6.6 of IEEE Std. 603-1991. Hence, the applicant submitted the exemption request from generic TS and design change description, dated September 1, 2015, for a Tier 2 departure from the AP1000 DCD in which the applicant proposed the following design changes to ensure that the regulatory criteria on operating bypasses for safety functions are met in the LNP COL application:

(1) Add a new permissive, P-8, to permit blocking the flux doubling logic during reactor startup (P-8 provides the logical permissive input to the PMS. P-8 is set to 551 degrees Fahrenheit (°F) (288 degrees Celsius (°C)) RCS temperature, the minimum temperature for criticality).

(2) Add logic that will cause the PMS to force chemical and volume CVS Valves 136A and 136B closed if the flux doubling logic is blocked when reactor temperature is less than P-8. This ensures a permissible condition exists before flux doubling is bypassed below P-8, which is one option from IEEE Std. 603-1991, the other being to perform the appropriate safety functions.

(3) When RCS temperature is below P-8 with the flux doubling signal block control logic actuated to block, reset of the flux doubling logic is required to open CVS Valves 136A and 136B.

(4) *Add an additional reset of source range flux doubling logic when RCS temperature falls below P-8. Existing PMS design resets flux doubling logic when neutron flux decreases below P-6.*

(5) *Include new permissive and actuation in TS, and describe the changes in Tier 2 information.*

In its submitted exemption request and design change description, the applicant also included revised logic Figure 7.2-1, Sheet 3 of 21, to show the incorporation of the above proposed design changes, which are evaluated below in this section of the safety evaluation.

In the AP1000-certified design, without this departure, when the reactor is shut down from power operations, the PMS design for the block of the flux doubling logic safety function met the criteria in Clause 6.6 of IEEE Std. 603-1991 regarding to the operating bypass because the flux doubling logic safety function will be automatically reset to remove its block when the neutron flux falls below the existing Permissive P-6 setpoint. However, when the reactor starts up, the certified design of the PMS did not meet the regulatory requirement to impose permissive conditions for the manual block of the flux doubling logic safety function at any time because there were no permissive conditions implemented in the PMS design for the manual block of the flux doubling logic safety function for the boron dilution block. In addition, for the flux doubling logic safety function the PMS design in the certified AP1000 DCD did not include control logic to reinstate permissive conditions or initiate appropriate safety function when the permissive conditions do not exist.

To address the above design deviations from the regulatory requirement on operating bypasses, the applicant proposed to create a new permissive, P-8, by using the RCS temperature to permit blocking the flux doubling logic during reactor startup. The setpoint for the new Permissive P-8 is selected to be at 551 °F (288.3 °C) for the RCS temperature, which is the minimum temperature for criticality for the AP1000 standard design. The staff found that this proposed design change will provide the necessary permissive condition to allow manual bypass of the flux doubling logic safety function during the plant startup. The applicant also proposed to add an additional reset of source range flux doubling logic when the RCS temperature falls below the setpoint for the new Permissive P-8. The staff found that this proposed design change will address the lack of the control logic in the current PMS design to reinstate permissive conditions to manually block the flux doubling logic safety function. When the RCS temperature falls below the setpoint for the new P-8 permissive, the applicant proposed to add logic in the PMS to force CVS Valves 136A and 136B closed. The CVS in the AP1000 DCD is designed to avoid or terminate boron dilution events by isolating sources of unborated water to the RCS during all modes of operation when signaled to do so by the PMS. Valves 136A and 136B are installed on the demineralized water supply line for isolating the unborated demineralized water to the CVS system. The staff found that this proposed change could prevent and/or terminate a boron dilution event from happening when the RCS temperature is below the new P-8 permissive setpoint if the flux doubling logic safety function is blocked.

In the revised logic Figure 7.2-1, Sheet 3 of 21, included in the submittal dated September 1, 2015, the staff noticed that there is a RESET/BLOCK momentary command for each applicable division for the "FLUX DOUBLING BLOCK CONTROL." This momentary command is used for the newly created function to close demineralized water system (DWS) isolation valves. However, the staff found that there is not a coincident voting logic used for this divisionized command. Therefore, the staff issued RAI 8404, Question 07.02-1, requesting the applicant to clarify how the single failure criterion, as required in Clause 5.1 of IEEE Std. 603-1991, is met for this newly added actuation signal sent to "CLOSE DWS ISOLATION VALVES." In its response, dated December 23, 2015, the applicant described how the DWS isolation valves are controlled by the PMS Division A for isolation Valve V136A and Division C for isolation Valve 136B, respectively. When the flux doubling block control is actuated for each division, the respective isolation valve is closed. Because the isolation valves are in series on the demineralized water supply connecting the DWS to the CVS system, the isolation function complies with the single failure criterion. In addition, this new function block to "CLOSE DWS ISOLATION VALVES" is added to prevent a boron dilution from happening if the flux doubling logic is blocked when the RCS temperature falls below the P-8 setpoint. Because this new function is not required to mitigate any DBE, it is not added as an engineered safety feature actuation function. The staff found that the response from the applicant to the above question in the RAI is appropriate and acceptable because it clarified how the design change meets the single failure criterion.

The applicant initially proposed to add logic to reset the flux doubling logic if CVS isolation Valves 136A and 136B are opened when RCS temperature is below the setpoint for the new P-8 permissive. However, the staff found that this original proposed change was not consistent with the revised logic Figure 7.2-1, Sheet 3 of 21. Hence, the staff issued RAI 8404, Question 07.02-1 requesting the applicant to explain how the proposed logic change would be implemented to match with the revised logic diagram (ADAMS Accession No. ML15329A055). In its response dated December 23, 2015, the applicant provided additional information stating that the information initially submitted is incorrect for this change, which should be changed as follows: When the RCS temperature is below the setpoint for the new P-8 permissive with the flux doubling signal block control logic actuated to block, the reset of the flux doubling block control logic is required to open CVS isolation Valves 136A and 136B (ADAMS Accession No. ML15329A055). The staff found that this modified description matches the revised logic Figure 7.2-1, Sheet 3 of 21.

Overall, the staff found that the changes to the PMS design comply with criteria in Clauses 5.1 and 6.6 of IEEE Std. 603-1991. Therefore, the staff found that the design changes proposed by the applicant are acceptable.

B.2 Boron Dilution Analysis

The staff reviewed the design change descriptions presented in the departure and exemption request (letter NPD-NRC-2015-038, dated November 12, 2015) with respect to the boron dilution analysis presented in AP1000 DCD Revision 19

Section 15.4.6. The design changes include adding a P-8 permissive which limits the ability to manually block the flux doubling calculation during plant startup and logic to force applicable CVS DWS isolation valves closed if the flux doubling logic is blocked.

The inclusion of the new permissive, P-8, does not change the approach and underlying assumptions used in the analysis for boron dilution as presented in Section 15.4.6. The logic presented in the exemption includes the automatic closure of the CVS valves if a manual block of the flux doubling logic is implemented below the P-8 permissive. This would block the potential source of unborated water and would be consistent with the termination method for a boron dilution event for modes 1 through 4 as discussed in DCD Section 15.4.6.2. When above the P-8 permissive, the manual block of the flux doubling logic may be permitted to allow for plant startup. The logic associated with the new P-8 permissive is also consistent with the description of dilution during startup (mode 2) as described in DCD Section 15.4.6.2.5.

Based on the staff's review of the new permissive and associated logic, the staff concludes that the boron dilution analysis presented in DCD Section 15.4.6 remains applicable given the changed descriptions presented in exemption request NPD-NRC-2015-038.

B.3 Technical Specifications

The design changes proposed by the applicant correspond to proposed changes in Section 3.3 of the TS and TS Bases (FSAR Chapter 16) in the COL application.

These changes, which appear in the September 1, 2015, submittal and have been incorporated into Part 4 of, Revision 8 of the COL application, submitted on December 7, 2015, are necessary to ensure that the TS and TS Bases accurately reflect the updated design and are described below.

Additionally, in a letter dated December 23, 2015, the applicant submitted its response to RAI Letter No. 135, Question 16-5, to address the staff's concerns related to proposed TS changes and insufficient level of details provided in the TS Bases. These changes, to be included in a future revision of the COL application, are among those described below and are being tracked by the staff as ***LNP Confirmatory Item 21.5-1****.*

Resolution of LNP Confirmatory Item 21.5-1

LNP Confirmatory Item 21.5-1 is a commitment by the applicant to revise the LNP COL application TS Bases as indicated in the letter dated December 23, 2015, in areas related to the flux doubling logic operating bypass. The staff confirmed that the LNP COL TS Bases have been appropriately revised. As a result, LNP Confirmatory Item 21.5-1 is now closed.

- *LCO 3.3.2 (ESFAS Instrumentation)*

In Table 3.3.2-1 (Page 9 of 13), the Mode 3 Applicability of Function 15.a, "Source Range Neutron Flux Doubling" is revised to indicate that this Function is "not applicable for valve isolation Functions whose associated flow path is isolated" (i.e., by applying Footnote (e) to the listed Mode 3).

In Table 3.3.2-1 (Page 10 of 13), a new Function 18.d, "Reactor Coolant Average Temperature, P-8" is added, with its associated requirements in columns for Applicable Modes or Other Specified Conditions, Required Channels, Conditions, and Surveillance Requirements, as follows (with added text underlined):

Applicable Modes or Other Specified Conditions	***Required Channels***	***Conditions***	***Surveillance Requirements***
2, 3(e), 4(e)	*4*	*J, T*	*SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5*
5(e)	*4*	*J, P*	*SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5*

- *Applicable Safety Analyses, LCOs, and Applicability (ASA) Section of TS Bases B3.3.2 (ESFAS Instrumentation)*

On Page B3.3.2-37, the discussion of Function 15 is revised as follows (with deleted text lined out and added text underlined) to accurately reflect the logics shown in DCD Figure 7.2-1 (Sheet 3 of 21):

"The block of boron dilution is accomplished by closing the CVS makeup line isolation ~~suction~~ valves or closing the demineralized water system isolation ~~storage tanks~~ valves to CVS~~, and aligning the boric acid tank to the CVS makeup pumps~~. This Function is actuated by Source Range Neutron Flux Doubling and Reactor Trip."

On Page B3.3.2-37, the discussion of Function 15.a is revised as follows (with added text underlined) to reflect the revised logics:

"A signal to block boron dilution in MODES 2 or 3, when not critical or during an intentional approach to criticality, and MODES 4 or 5 is derived from source range neutron flow increasing at an excessive rate (source range flux doubling). This Function is not applicable in MODES 3, 4 and 5 if the demineralized water makeup flow path is isolated. The source range neutron detectors are used for this Function. The LCO requires four divisions to be OPERABLE. There are four divisions and two-out-of-four logic is used. On a coincidence of excessively increasing source range neutron flux in two of the four divisions, demineralized water is isolated (CVS demineralized water system isolation valves closed) from the makeup pumps and reactor coolant makeup is isolated (CVS makeup line isolation valves closed) from the reactor coolant system to preclude a boron dilution event. In MODE 6, a dilution event is precluded by the requirement in

LCO 3.9.2 to close, lock and secure at least one valve in each unborated water source flow path."

On Page B3.3.2-37, the discussion of Function 15.b is revised, in part, as follows (with deleted text lined out and added text underlined) to clarify the specific components actuated by the permissive P-4:

"A P-4 signal initiates isolation of RCS makeup from the CVS ~~Demineralized Water Makeup is also isolated~~ by closing the demineralized water system isolation valves, and ~~aligned to the CVS makeup pumps)~~ aligning the CVS makeup pump suction to the boric acid tank. Unborated water source makeup isolation is initiated by all the Functions that initiate a Reactor Trip."

On Page B3.3.2-41, the discussion of Function 18.c, "Intermediate Range Neutron Flux, P-6," is revised as follows (with deleted text lined out and added text underlined) to reflect the revised logics:

"The Intermediate Range Neutron Flux, P-6 interlock is actuated when the respective NIS intermediate range channel increases to approximately one decade above the channel lower range limit. Above the setpoint, the P-6 interlock allows manual block of the source range neutron flux reactor trip. Below the setpoint, the P-6 interlock automatically energizes the source range detectors and unblocks the source range neutron flux reactor trip. As intermediate range flux decreases from above the setpoint to below the setpoint, the P-6 interlock automatically resets the flux doubling block function ensuring ~~unblocks~~ the source range neutron flux doubling function is enabled~~, permitting the block of boron dilution~~. Normally, the source range neutron flux doubling f~~ this F~~unction is blocked by the main control room operator during reactor startup. This Function is required to be OPERABLE in MODE 2."

On Page B3.3.2-42, the discussion of the new Function 18.d is added as follows to reflect the revised logics:

"The P-8 interlock is provided to permit a manual block of or to reset a manual block of the automatic Source Range Neutron Flux Doubling actuation of the Boron Dilution Block (Function 15.a).

The automatic Source Range Neutron Flux Doubling actuation of the Boron Dilution Block Function may be manually blocked (disabled) to permit plant startup and normal power operation when above the P-8 reactor coolant average temperature setpoint.

The manual block to disable the automatic Source Range Neutron Flux Doubling actuation of the Boron Dilution Block Function is automatically reset upon decreasing reactor coolant average temperature to below the P-8 setpoint.

Once reactor coolant average temperature is below the P-8 setpoint, the Source Range Neutron Flux Doubling actuation of the Boron Dilution Block Function may also be manually blocked to prevent inadvertent actuation during refueling operations and post-refueling control rod testing.

When the Source Range Neutron Flux Doubling actuation of the Boron Dilution Block is manually blocked below P-8 during shutdown conditions, the CVS demineralized water system isolation valves will automatically close to prevent inadvertent boron dilution.

The P-8 interlock is required to be OPERABLE in MODES 2, 3, 4 and 5. This Function is not applicable in MODES 3, 4 and 5, if the demineralized water makeup flow path is isolated. In MODE 6 a dilution event is precluded by the requirement in LCO 3.9.2 to close, lock and secure at least one valve in each unborated water source flow path."

- *Applicable Safety Analyses, LCOs, and Applicability (ASA) Section of TS Bases B3.3.1 (Reactor Trip System (RTS) Instrumentation)*

In addition, unrelated to the revised logics in the ESFAS, on Page B3.3.1-23, in the discussion of the permissive P-6, Item a(3) is revised as follows (with deleted text lined out and added text underlined) to reflect relevant information regarding the permissive P-6:

"(3) on decreasing ~~increasing~~ power, the P-6 interlock automatically resets the flux doubling block control ensuring ~~provides a backup block signal to~~ the source range neutron flux doubling circuit is enabled. Normally, the source range neutron flux doubling circuit ~~this Function~~ is manually blocked by the main control room operator during the reactor startup."

- *Actions Section of TS Bases B3.3.2 (ESFAS Instrumentation)*

On Page B3.3.2-57, in the discussion of Actions J.1 and J.2, the first paragraph is revised to read, in part, "[C]ondition J applies to P-6, P-8, P-11, P-12, and P-19 interlocks ..." to reflect the addition of the permissive P-8.

The staff finds the above proposed changes to TS LCO 3.3.2 and its associated bases acceptable because they reflect the revised logic for the source range neutron flux doubling function of the AP1000 ESFAS as described in DCD Section 7.3.

Based on the above evaluation, the staff finds the proposed TS and Bases revisions meet the requirements of 10 CFR 50.36.

B.4 Risk Results and Insights

This design departure does not affect the description of AP1000 design features that reduce the risk of boron dilution events. It does not modify the plant-specific probabilistic risk assessment model used for licensing. Consequently, there is no change to the risk profile described in the COL application or the risk insights concerning boron dilution in AP1000 DCD Revision 19, Table 19.59-18 (Item 9). Instead, the change improves confidence in the validity of the reported risk results and insights. Consistent with DC/COL-ISG 003, "PRA Information to

Support Design Certification and Combined License Applications," the plant-specific probabilistic risk assessment remains acceptable to the staff.

21.5.5 Post Combined License Activities

There are no post-COL activities related to this section.

21.5.6 Conclusion

The staff reviewed the application for proposed departure number WLS DEP 7.3-1 and checked the referenced DCD. The staff's review confirmed that the applicant addressed the required information relating to the departures and there is no outstanding information expected to be addressed in the WLS COL FSAR and TS related to this departure.

In addition, the staff concludes that the relevant information presented in the WLS COL FSAR TS is acceptable and meets the regulatory requirements and guidance discussed in Section 21.4.3 of this SER. The staff based its conclusion on the following:

Based on the evaluation discussed above, the staff concludes that the changes to the PMS design and the RAI responses for bypassing the source range neutron flux doubling logic input to the boron dilution block comply with 10 CFR 50.55a(h)(3) because they meet the criteria in Clauses 5.1 and 6.6 of IEEE Std. 603-1991. The staff therefore finds the design changes proposed by the applicant acceptable.

22.0 CONCLUSIONS

In accordance with Subpart C, “Combined Licenses,” of Title 10 of the *Code of Federal Regulations* (10 CFR), Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants,” the staff of the U.S. Nuclear Regulatory Commission reviewed the combined license (COL) application submitted by Duke Energy Carolinas LLC., for the William States Lee III Units 1 and 2. Based on the staff’s evaluation documented in this final safety evaluation report (FSER), the staff finds the following with respect to the safety aspects[1] of the COL application:

1) The applicable standards and requirements of the Atomic Energy Act and Commission’s regulations have been met,

2) Required notifications to other agencies or bodies have been duly made,

3) There is reasonable assurance that the facility will be constructed and will operate in conformity with the license, the provisions of the Atomic Energy Act, and the Commission’s regulations,

4) The applicant is technically and financially qualified to engage in the activities authorized, and,

5) Issuance of the license will not be inimical to the common defense and security or to the health and safety of the public.

[1] An environmental review was also performed of the COL application and its evaluation and conclusions are documented in NUREG-2111, “Final Environmental Impact Statement for Combined Licenses for William States Lee III Nuclear Station Units 1 and 2,” dated December 20, 2013.

Appendix A

Post COL Activities:
License Conditions; Inspections, Tests, Analyses, and Acceptance Criteria; and Final Safety Analysis Report Commitments

A.1 License Conditions

The Nuclear Regulatory Commission's (NRC's) regulations at Title 10 of the *Code of Federal Regulations* (10 CFR) 52.97, "Issuance of combined licenses," requires a combined license (COL) to specify any terms and conditions of the COL the Commission deems appropriate. A license condition is not needed when an existing NRC regulation requires a future regulatory review of a matter to ensure adequate safety during design, construction, inspection activities or operation for a new plant. The staff is proposing that the Commission include the following license conditions, which are set forth below, to control various safety matters.

Proposed License Condition in FSER	FSER Section	License Condition Description
1-1	1.5.1	Primary and secondary financial protection per 10 CFR 140.11(a)(4) and 10 CFR 50.54(w)
1-2	1.5.1	Financial assurance – deferred reporting of 10 CFR 140.21 for guarantee of payment
1-3	1.5.5	10 CFR Parts 30, 40, and 70 licenses governing the possession and use of applicable source, byproduct and special nuclear materials
1-4	1.5.5	Implementation schedule submission requirements for Special Nuclear Material Control and Accounting Program
1-5*	1.5.5	Implementation schedule submission requirements for Non-Licensed Plant Staff Training Program
1-6	1.5.5	Implementation of Special Nuclear Material Physical Protection Program
2-1	2.4.12.5	Removal of potential preferential groundwater pathways (legacy Cherokee stormwater drain line)
2-2	2.5.3.5	Geologic mapping
3-1	3.6.5	As-designed pipe rupture hazards analysis
3-2	3.7.2.5	Seismic interaction analysis update to reflect as-built information

Proposed License Condition in FSER	FSER Section	License Condition Description
3-3	3.7.2.5	Seismic analyses reconciliation to account for detailed design changes
3-4	3.8.5.5	Implementation schedule submission requirements for construction and inspection procedures for steel concrete composite construction activities for seismic Category I nuclear island modules
3-5	3.9.6.5	Preservice Testing Operational Program and the Motor-Operated Valve Testing Operational Program
3-6	3.9.6.5	Implementation schedule submission requirements for Inservice Testing program (including preservice and motor-operated valve testing)
3-7	3.9.6.5	Squib valve surveillance and maintenance
3-8	3.11.5	Implementation of Environmental Qualification Program
3-9	3.11.5	Implementation schedule submission requirements for Environmental Qualification Program
3-10	3.12.5	As-designed individual piping segments and reporting requirements
4-1	4.5	Instrument uncertainty for measuring departure from nucleate boiling ratio values
5-1**	5.2.4.5	Implementation schedule submission requirements of operational programs in FSAR Table 13.4-201 (Preservice Inspection and Inservice Inspection Programs)
5-2	5.3.2.5	Implementation of Reactor Vessel Material Surveillance Program
5-3	5.3.2.5	Implementation schedule submission requirements for Reactor Vessel Material Surveillance program
5-4	5.3.3.5	Updating the pressure-temperature limits using the approved pressure-temperature limits report methodologies for reactor vessel material properties
5-5	5.3.4.5	Plant-specific belt line material properties
5-6**	5.4.5	Implementation schedule submission requirements for Preservice Inspection and Inservice Inspection Programs
6-1	6.2.5	Implementation of containment leakage rate testing program
6-2	6.2.5	Implementation schedule submission requirements for containment leakage rate testing program
6-3**	6.6.5	Implementation schedule submission requirements for Preservice Inspection and Inservice Inspection Programs
9-1	9.1.2.5	Implementation of and implementation schedule submission requirements for spent fuel rack Metamic Coupon Monitoring Program
9-2	9.5.1.5	Implementation of Fire Protection Program
9-3	9.5.1.5	Implementation schedule submission requirements for Fire Protection Program

Proposed License Condition in FSER	FSER Section	License Condition Description
10-1	10.1.5	Implementation of and implementation schedule submission requirements for flow accelerated corrosion program
10-2	10.2.5	Implementation of and implementation schedule submission requirements for turbine maintenance and inspection program
11-1	11.2.5	Radionuclide inventory of unpackaged wastes
11-2	11.4.5	Implementation of operational program for process and effluent monitoring and sampling (including process control program)
11-3	11.4.5	Implementation schedule submission requirements for operational program for process and effluent monitoring and sampling (including process control program)
11-4	11.5.5	Implementation of operational program for process and effluent monitoring and sampling, including (1) Radiological Effluent Technical Specifications/Standard Radiological Effluent Controls, (2) Offsite Dose Calculation Manual, and (3) Radiological Environmental Monitoring Program
11-5	11.5.5	Implementation schedule submission requirements for operational program for process and effluent monitoring and sampling, including (1) Radiological Effluent Technical Specifications/Standard Radiological Effluent Controls, (2) Offsite Dose Calculation Manual, and (3) Radiological Environmental Monitoring Program
12-1	12.5.5	Implementation of Radiation Protection Program (including the as low as reasonably achievable (ALARA) principle)
12-2	12.5.5	Implementation schedule submission requirements for Radiation Protection Program (including the ALARA principle)
13-1	13.2.5	Implementation of Reactor Operator Training Program
13-2*	13.2.5	Implementation schedule submission requirements for Non-Licensed Plant Staff Training Program, Reactor Operator Training Program, and Reactor Operation Requalification Program
13-3	13.3.5	Schedule submission requirements for a fully developed set of site-specific emergency action levels
13-4	13.3.5	Identify specific locations of the reception centers and relocation sites and obtain of letters of agreement for locations not under Duke Energy’s control
13-5	13.3.5	NEI 10-05, detailed staffing analysis
13-6	13.3.5	10 CFR Part 50 Appendix E letters of agreement with emergency organizations

Proposed License Condition in FSER	FSER Section	License Condition Description
13-7	13.3.5	Demonstrate the integrated capability and functionality of the emergency operation facility
13-8	13.3.5	Implementation schedule submission requirements for operational programs in FSAR Table 13.4-201, including emergency plan implementing procedures
13-9	13.3.5	Implementation schedule submission requirements for operational programs in FSAR Table 13.4-201, including emergency response data system implementation program plan
13-10***	13.3.5	NEI 12-01, Staffing assessment
13-11****	13.3.5	NEI 12-01, Communications capability assessment
13-12	13.6.5	Implementation schedule submission requirements for physical security programs
13-13	13.7.5	Implementation schedule submission requirements for Fitness for Duty operational program
13-14	13.8.5	Implementation schedule submission requirements for Cyber Security program implementation
14-1	14.2.3.5	Implementation schedule submission requirements for implementation of preoperational and startup procedures
14-2	14.2.3.5	Initial startup test program changes
14-3	14.2.5.5	First-plant-only and first-three-plant-only testing
14-4	14.2.8.5	Implementation milestones for initial test program
14-5	14.2.8.5	Implementation schedule submission requirements for initial test program
14-6	14.2.8.5	Pre-operational, pre-critical, initial criticality, low-power, and power ascension testing
15-1	15.0.5	Schedule submission requirements for calculations for power calorimetric uncertainty instrumentation and administrative controls
17-1	17.6.5	Implementation schedule submission requirements for Maintenance Rule program
19-1	19.59.5	AP1000 seismic margin analysis
19-2	19.59.5	AP1000 probabilistic risk assessment
19-3	19.59.5	AP1000 internal fire and internal flood analysis
19-4	19.59.5	Implementation schedule submission requirements for site-specific severe accident management guidelines
19-5	19.59.5	Thermal lag assessment
19-6	Appendix 19F	Malevolent aircraft impact FSAR revisions

Proposed License Condition in FSER	FSER Section	License Condition Description
19.A-1	19.A.5	Implementation schedule submission requirements for operational and programmatic elements of mitigative strategies for responding to a loss of large areas event
20-1	20.1.5	Mitigation strategies for beyond-design-basis external events
20-2	20.2.5	Reliable spent fuel pool instrumentation
20-3***	20.3.5	NEI 12-01, Staffing assessment
20-4****	20.3.5	NEI 12-01, Communications capability assessment

* License Conditions 1-5 and 13-2 represent the same reporting requirements for the Non-Licensed Plant Staff Training Program.
** License Conditions 5-1, 5-6, and 6-3 represent the same reporting requirements for the Preservice Inspection Program and Inservice Inspection Program.
*** License Conditions 13-10 and 20-3 represent the same requirement to perform the staffing assessment in accordance with NEI 12-01, Revision 0.
**** License Conditions 13-11 and 20-4 represent the same requirement to perform the communications capability assessment in accordance with NEI 12-01, Revision 0.

Appendix A

License Conditions, Inspections, Tests, Analyses, and Acceptance Criteria, and Final Safety Analysis Commitments

A.2 Inspections, Tests, Analyses, and Acceptance Criteria

The staff has identified the certain Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) that it will recommend the United States Nuclear Regulatory Commission impose with respect to a COL issued to the applicant. The following is a list of those ITAAC. In addition to the ITAAC contained in this list, the ITAAC found in the AP1000 DCD Revision 19 Tier 1 material will also be incorporated into the COL should a COL be issued to the applicant.

Listing of Lee Site-Specific ITAAC

ITAAC Number from Draft License	ITAAC Description	SER Section
C.2.5.04.04a–C.2.5.04.04c	Feedwater Flow Measurement	15.0
C.2.6.09.01–C.2.6.09.09	Physical Security	13.6.A
C.2.6.12.01–C.2.6.12.07	Offsite Power System	8.2.A
C.3.8.01.01.01–C.3.8.01.10.01	Emergency Planning	13.3
C.3.8.02.01	Waterproof Membrane	3.8.5
C.3.8.05.01	Pipe Rupture Hazards Analysis	3.6
C.2.8.01.01	Piping Design	3.12
C.2.2.05.07e	Main Control Room Emergency Habitability System	21.2
C.2.3.09.03.iii	Containment Hydrogen Control System[1]	21.4

[1] The Main Control Room Emergency Habitability System ITAAC and the Containment Hydrogen Control System ITAAC appear in the AP1000 DCD Revision 19 and were revised in the WLS COL application.

Appendix A

License Conditions,
Inspections, Tests, Analyses, and Acceptance Criteria, and
Final Safety Analysis Report Commitments

A.3 Final Safety Analysis Report (FSAR) Commitments

The following FSAR commitments are identified as the responsibility of the licensee:

SER Section	Description
1.4	A site-specific construction plan and startup schedule will be provided after issuance of the COL.
1.4	The licensee will update the FSAR to identify additional participants, principal consultants, outside service organizations, or contractors for the design, construction, and operation of WLS. The licensee will also delineate the division of responsibility among the certified plant designer, architect-engineer, constructor, and plant operator as appropriate.
5.2.5	Prior to initial fuel load, the operating procedures that include identifying, monitoring, trending, and managing the prolonged low-level reactor coolant system leakage will be developed.
6.4	FSAR Commitment 6.4-1. The licensee's CR operator training program shall address the following: • Regulatory Position C.5, "Emergency Planning," of RG 1.78 • Regulatory Position 2.5, "Hazardous Chemicals," of RG 1.196 • Regulatory Position 2.2.1, "Comparison of System Design, Configuration, and Operation with Licensing Basis," of RG 1.196 Regulatory Position 2.7.1, "Periodic Evaluations and Maintenance," of RG 1.196
9.1.4	The light load handling program, including system inspections, will be implemented prior to receipt of fuel onsite.
9.1.5	The overhead heavy-load handling program, including system inspections, will be implemented prior to receipt of fuel onsite.

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Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
2/8/2008	ML080460359	Duke Energy WSL III Units 1 & 2 COLA (Emergency Plan), Rev. 1	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
12/9/2008	ML083460112	William States Lee III Nuclear Station, Units, 1 & 2, AP1000 Combined License Application for the Partial Response to Request for Additional Information (RAI No. 50) Ltr # WLG2008.12-01.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
12/9/2008	ML083460113	William States Lee III Nuclear Station, Units 1 and 2 - AP1000 Combined License Application for the Response to Request for Additional Information Ltr# WLG2008.12-10.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
12/10/2008	ML083450023	2008/12/10 Lee (Duke) COL Hearing - RAI Letter No. 057 Related SRP Section 9.5.1 for Lee Units 1 and 2 - CORRECTION	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
12/11/2008	ML083510881	William States Lee III, Units 1 and 2 - Response to Request for Additional Information Ltr# WLG2008.12-09 Re: AP1000 Combined License Application.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC Duke Energy Corp	NRC/Document Control Desk NRC/NRO	05200018 05200019

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Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
12/11/2008	ML083510884	William States Lee III Nuclear Station, Units 1 and 2 - Response to Request for Additional Information re: Socioeconomics.	Graphics incl Charts and Tables Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC Duke Energy Corp	NRC/Document Control Desk NRC/NRO	05200018 05200019
12/11/2008	ML083510885	William States Lee III, Units 1 and 2, AP1000 Combined License Application, Response to Request for Additional Information Ltr # WLG2008.12-13.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
12/11/2008	ML083520210	William States Lee III, Units 1 and 2 - AP1000 Combined License Application, Response to Request for Additional Information Ltr# WLG2008.12-14 Concerning Terrestrial Ecology.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
12/11/2008	ML083520211	William States Lee III Nuclear Station, Units 1 and 2 - Response to Request for Additional Information Nos. 701, 702, 703, and 704 re: SRP Section 11.02 for the AP1000 Combined License Application.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC Duke Energy Corp	NRC/Document Control Desk NRC/NRO	05200018 05200019

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Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
12/11/2008	ML083520336	William States Lee III, Units 1 and 2 - AP1000 Combined License Application, Partial Response to Request for Additional Information (RAI No. 826), Ltr # WLG2008.12-06.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
12/12/2008	ML083510882	William States Lee III, Units 1 and 2, AP1000 Combined License Application Contract for Disposal of High-Level Radioactive Waste Ltr # WLG2008.12-21.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
12/12/2008	ML083510883	William States Lee III, Units 1 and 2, AP1000 Combined License Application, Response to Request for Additional Information Ltr# WLG2008.12-11.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
12/12/2008	ML083520212	William States Lee III Units 1 & 2 - AP1000 Combined License Application, Response to Request for Additional Information Ltr# WLG2008.12-17.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC Duke Energy Corp	NRC/Document Control Desk NRC/NRO	05200018 05200019
12/17/2008	ML083540415	William States Lee III Nuclear Station Units 1 and 2 AP1000 Combined License Application, Response to Request for Additional Information (RAI No. 1705) Ltr. #WLG2008.12-26.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
12/17/2008	ML083540416	William States Lee III, Units 1 & 2 AP1000 Combined License Application, Partial Response to Request for Additional Information No. 50, Ltr. # WLG2008.12-20.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
12/17/2008	ML083570396	William States Lee III Nuclear Station, Units 1 and 2 - Response to Request for Additional Information, (RAI Nos. 1003 and 1004), Ltr# WLG2008.12-25.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC Duke Energy Corp	NRC/Document Control Desk NRC/NRO	05200018 05200019
12/17/2008	ML083570591	William States Lee III, Units 1 and 2, AP1000 Combined License Application, Extension Request for Response to Additional Information on Phase I Schedule, WLG2008.12-27.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
12/17/2008	ML083590244	William States Lee III Nuclear Station, Units 1 and 2 - Response to Request for Additional Information (RAI No. 448) Ltr# WLG2008.12-18.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
12/17/2008	ML083650408	William States Lee III Nuclear Station, Units 1 and 2 - AP1000 Combined License Application, Response to Request for Additional Information (RAI No. 451) Ltr# WLG2008.12-22.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
12/22/2008	ML080601189	William States Lee III Nuclear Station Request for Withholding Information from Public Disclosure	Letter Proprietary Information Review	NRC/NRO/DNRL/AP1000B1	Duke Energy Carolinas, LLC	05200018 05200019
12/23/2008	ML083580106	2008/12/23 Lee RAI for SER - LEE-RAI-LTR-050 RELATED TO SRP 02.05.02 FOR THE LEE UNITS 1 AND 2 COL	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
12/23/2008	ML083580208	2008/12/23 Lee RAI for SER - Request for Additional Information Letter No. 50 SRP Section 02.05.02 - Vibratory Ground Motion.	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
12/23/2008	ML083640466	William States Lee III, Units 1 and 2 AP1000 Combined License Application, Response to Request for Additional Information (RAI No. 837), Ltr# WLG2008.12-15.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
12/23/2008	ML083659363	William States Lee III Nuclear Station, Units 1 and 2, AP1000 Combined License Application, Response to Request for Additional Information (RAI No. 1473), Ltr# WLG2008.12-28.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
12/23/2008	ML083660090	William States Lee III Nuclear Station, Units 1 and 2, Combined License Application, to Response to Request for Additional Information Letter Ltr # WLG2008.12-29.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
12/23/2008	ML083660210	William States Lee III Nuclear Station, AP1000 Combined License Application, Non-Public Partial Response to Request for Additional Information (RAI No. 837), Ltr.# WLG2008.12-16.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
12/23/2008	ML083660272	William States Lee III Nuclear Station, Units 1 and 2 - Partial Response to Request for Additional Information (RAI No. 50) Letter No. 025 Related to SRP Section 13.3 for the AP1000 Combined License Application.	Drawing Graphics incl Charts and Tables Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC Duke Energy Corp	NRC/Document Control Desk NRC/NRO	05200018 05200019
12/23/2008	ML090020175	William States Lee III Nuclear Station, Units 1 and 2, Combined License Application, Response Request for Additional Information Letter Ltr # WLG2008.12-30.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
12/29/2008	ML083640079	2008/12/29 Lee RAI for SER - RAI LETTER NO. 050A RELATED TO SRP 02.05.02 for the LEE COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
12/31/2008	ML083530210	1/15/2009 -Notice of Meeting with AP1000 DCWG to Discuss Various Topics of Interest (TAC Q00118).	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL	NRC/NRO/DNRL	05200014 05200015 05200018 05200019 05200022 05200023 05200025 05200026 05200027 05200028 05200029 05200030 05200040 05200041 PROJ0763

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Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
12/31/2008	ML090060910	01/15/2009 Revised Meeting Notice, Meeting With AP1000 Design-Centered Working Group.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/NWE1	NRC/NRO/DNRL/NWE1	05200014 05200015 05200018 05200019 05200022 05200023 05200025 05200026 05200027 05200028 05200029 05200030 05200040 05200041 PROJ0763

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1/5/2009	ML090060684	E-Mail - Summary - Applicability of GDC to OPS December 19, 2008.	E-Mail	NRC/NRO/DNRL Nuclear Energy Institute (NEI)	NRC/NRO/DNRL	05200001 05200002 05200003 05200004 05200006 05200007 05200008 05200009 05200010 05200011 05200012 05200013 05200014 05200015 05200016 05200018 05200019

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						05200020 05200021 05200022 05200023 05200024 05200025 05200026 05200027 05200028 05200029 05200030 05200031 05200032 05200033 05200034 05200035 05200036

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1/8/2009	ML090080767	2009/01/08 Lee RAI for SER - RAI LETTER NO. 059 RELATED to SRP 02.05.01 FOR THE LEE UNITS 1 AND 2 COL	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
1/9/2009	ML090090150	2009/01/09 Lee RAI for SER - RAI LETTER NO. 060 RELATED TO SRP 02.05.04 FOR THE W S LEE UNITS 1 AND 2 COL	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
1/12/2009	ML090120621	2009/01/12 Lee RAI for SER - RAI 1874 Letter # 061 SRP 02.05.04 FOR THE WS LEE III UNITS 1 AND 2 COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
1/14/2009	ML090140072	2009/01/14 Lee RAI for SER - RAI LETTER NO. 062 RELATED TO SRP 13.03 FOR W. S. LEE UNITS 1 AND 2 COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
1/14/2009	ML090140208	2009/01/14 Lee RAI for SER - RAI LETTER NO. 051 RELATED TO SRP SECTION: 02.05.02 - VIBRATORY GROUND MOTION FOR W.S. LEE UNITS 1 AND 2 COL	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
1/14/2009	ML090140271	2009/01/14 Lee RAI for SER - RAI LETTER NO. 063 RELATED TO SRP 02.05.02 FOR THE W. S. LEE UNITS 1 AND 2 COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019

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1/21/2009	ML083120589	Request for Additional Information Regarding the Environmental Review of the Combined License Application for William States Lee III Nuclear Station, Units 1 and 2.	Letter	NRC/NRO/DSER/RAP2	Duke Power Co	05200018 05200019
1/21/2009	ML090070370	Lee RAI's Second Round Table PNNL, Second Requests for Additional Information (RAIs).	Request for Additional Information (RAI)	NRC/NRO/DSER	Duke Power Co	05200018 05200019
1/21/2009	ML090210239	2009/01/21 Lee RAI for SER - RAI letter no. 58 RELATED TO SRP SECTION PART 1 GENERAL AND FINANCIAL INFORMATION FOR THE WS LEE III UNITS 1 AND 2 COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
1/21/2009	ML090210325	2009/01/21 Lee RAI for SER - RAI LTR NO. 58 RELATED TO SRP SECTION PART 1 GENERAL AND FINANCIAL INFORMATION FOR THE WS LEE III UNITS 1 AND 2 COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019

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1/23/2009	ML090260039	NRC Response to GDC 2, 4, and 5 One-Pager.	E-Mail	NRC/NRO/DNRL	Nuclear Energy Institute (NEI)	05200001 05200002 05200003 05200004 05200006 05200007 05200008 05200009 05200010 05200011 05200012 05200013 05200014 05200015 05200016 05200017 05200018

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						05200019 05200020 05200021 05200022 05200023 05200024 05200025 05200026 05200027 05200028 05200029 05200030 05200031 05200032 05200033 05200034 05200035

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1/27/2009	ML090260304	Letter to Mr. Butch Smith Regarding the William States Lee COL.	Letter	NRC/NRO/DSER/RHEB	Cleveland County Sanitary District, NC	05200018 05200019
1/28/2009	ML090280416	2009/01/28 Lee RAI for SER - Request for Additional Information Letter No. 064 Related to SRP Section 9.2.1 for the William States Lee III Units 1 and 2 Combined License Application	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
1/28/2009	ML090280454	2009/01/28 Lee (Duke) COL Hearing - RAI Letter No. 064 Related SRP Section 9.2.1 for Lee Units 1 and 2	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
2/3/2009	ML112800226	USDA 2009 2007 Census of Agriculture.	Report, Miscellaneous	US Dept of Agriculture, National Agricultural Statistics Service	NRC/NRO	05200018 05200019
2/5/2009	ML090400619	William States Lee III, Units 1 & 2, AP1000 Combined License Application, Summary Identification of Concurrence with Standard Content in Response to Requests for Additional Information.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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2/13/2009	ML090260352	02/13/2009 Summary of Meeting with the AP1000 Design-centered Working Group to Discuss the Status of Items of Interest.	Meeting Agenda Meeting Summary Memoranda	NRC/NRO/DNRL/NWE1	NRC/NRO/DNRL/NWE1	05200014 05200015 05200018 05200019 05200022 05200023 05200025 05200026 05200027 05200028 05200029 05200030 05200040 05200041 PROJ0763

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2/16/2009	ML090490675	William States Lee III Nuclear Station, Units 1 and 2, AP1000 Combined License Application, Response to Request for Additional Information on Ltr# WLG2009.2-06.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC Duke Energy Corp	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/16/2009	ML090490676	William States Lee III Nuclear Station, Units 1 and 2, AP1000 Combined License Application - Response to Request for Additional Information Ltr# WLG2009.2-05.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC Duke Energy Corp	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/16/2009	ML090490677	William States Lee III Nuclear Station Units 1 and 2, Combined License Application, Response to Request for Additional Information on Emergency Planning, Ltr# WLG2009.02-03.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/16/2009	ML090490679	William States Lee III Nuclear Station Units 1 and 2, Combined License Application, Response to Request for Additional Information on Environmental Review, Ltr# WLG2009.2-04.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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2/19/2009	ML090540474	AP1000 Combined License Application for the William States Lee III Nuclear Station Units 1 and 2 - Response to Request for Additional Information.	Letter Response to Request for Additional Information (RAI)	Duke Energy Generation Services	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/19/2009	ML090540808	William States Lee III, Units 1 and 2 - Response to Request for Additional Information Ltr# WLG2009.02-08 Regarding Environmental Review of the Combined License Application, Dated January 21, 2009.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/20/2009	ML090560373	AP1000 Combined License Application for the William States Lee III Nuclear Station Units 1 and 2 - Response to Request for Additional Information (RAI No. 1760) Ltr# WLG2009.2-07.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC Duke Energy Corp	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/25/2009	ML090620574	PNNL SER Input for Lee NPS COLA.	Emergency Preparedness-Emergency Plan	NRC/NSIR		05200018 05200019
2/26/2009	ML090420471	Letter to Duke Addressing: Change in Schedule of William State Lee III Nuclear Station, Units 1 and 2 Combined License Application Environmental Review.	Letter	NRC/NRO/DSER/RAP3	Duke Power Co	05200018 05200019

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3/26/2009	ML112940519	General Assembly of North Carolina, Senate Bill 907 - Water Resource Policy Act of 2009.	- No Document Type Applies	State of NC, General Assembly	NRC/NRO	05200018 05200019
4/2/2009	ML091060213	WS Lee ETE Matrix to Accompany ETE Analysis.	- No Document Type Applies	- No Known Affiliation	NRC/NSIR	05200018 05200019
4/14/2009	ML091060497	William States Lee III Nuclear Station Units 1 and 2, Combined License Application, Response to Request for Additional Information on Environmental Review Ltr# WLG2009.04-01.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
4/14/2009	ML091060499	William States Lee III Nuclear Station Units 1 and 2, Combined License Application Response to Request for Additional Information on (RAI No. 2002) Ltr# WLG2009.04-04.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
4/14/2009	ML091060500	William States Lee III Nuclear Station, Units 1 and 2, AP1000 Combined License Application, Review Guide for Part 3, Environmental Report, Revision 1, and Part 9, Withheld Information, Revision 2, LTR# WLG2009.04-02.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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4/15/2009	ML091050662	2009/04/15 Lee RAI for SER - RAI LETTER NO. 068 RELATED TO SRP SECTION: 03.08.05 - FOUNDATIONS FOR THE WILLIAM STATES LEE III UNITS 1 AND 2 COL	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
4/17/2009	ML091080048	2009/04/17-Reply to Answers of Duke Energy and NRC Staff Regarding New Contention Eleven.	Legal-Motion	Blue Ridge Environmental Defense League	NRC/ASLBP	05200018 05200019
4/17/2009	ML091110030	William States Lee III, Units 1 and 2 - AP1000 Combined License Application, Response to Request for Additional Information No. 1487, Revision 1, Ltr#WLG2009.04-03.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
4/20/2009	ML091060210	Staff Analysis of Evacuation Time Estimate (ETE) Study WS Lee COLA.	- No Document Type Applies	- No Known Affiliation	NRC/NSIR	05200018 05200019
4/28/2009	ML091200383	William States Lee III, Units 1 and 2 - AP1000 Combined License Application - Response to Request for Additional Information, RAI 109, Radiological Health.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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4/28/2009	ML091200402	William States Lee III Nuclear Station - AP1000 Combined License Application, Response to Request for Additional Information (RAI No. 1589) Ltr. #WLG2009.04-07.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC Duke Energy Corp	NRC/Document Control Desk NRC/NRO	05200018 05200019
4/29/2009	ML091190338	2009/04/29-Memorandum and Order (Regarding BREDL's New Contention Eleven).	Legal-Order	NRC/ASLBP	Blue Ridge Environmental Defense League	05200018 05200019
4/29/2009	ML091200570	William States Lee III, AP1000 Combined License Application, Response to Request for Additional Information Ltr#WLG2009.04-06.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
4/30/2009	ML090990348	Duke Energy WSL III Units 1 & 2 COLA (Environmental Report), Rev. 1	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
5/5/2009	ML091280032	William States Lee III, Units 1 and 2 - Combined License Application, Environmental Report (Part 3) Thermal Discharge Modeling, Ltr # WLG2009.05-01.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
5/12/2009	ML091340410	William States Lee III, Units 1 and 2, Supplemental Response to Request for Additional Information (RAI No. 826), Ltr# WLG2009.05-07.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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5/12/2009	ML091340476	William States Lee III, Units 1 and 2, Response to Request for Additional Information, Ltr# WLG2009.05-02.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
5/12/2009	ML091350201	William States Lee III, Units 1 & 2 - Submittal of Revised Safeguards/Physical Security Plan, Ltr # WLG2009.05-03.	Letter Security Plan	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO NRC/NSIR	05200018 05200019
5/15/2009	ML091400205	William States Lee III AP1000 Combined License Application, Response to Request for Additional Information (RAI No. 1874) Ltr. #WLG2009.05-06.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
5/15/2009	ML091400206	William States Lee III. Units 1 & 2, AP1000 Combined License Application, Response to Request for Additional Information (RAI No. 1881) Ltr# WLG2009.05-05.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
5/15/2009	ML091400207	William States Lee III Nuclear Station, Units 1 & 2, AP1000 Combined License Application, Request for Additional Information (RAI No. 1922), Ltr# WLG2009.05-08.	Letter	Duke Energy Carolinas, LLC Duke Energy Corp	NRC/Document Control Desk NRC/NRO	05200018 05200019

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5/21/2009	ML091100248	April 9, 2009 Summary of Category 1 Public Meeting with the AP1000 Design-Centered Working Group to Discuss the Status of Items of Interest.	Meeting Agenda Meeting Summary Memoranda	NRC/NRO/DNRL/NWE1	NRC/NRO/DNRL/NWE1	05200014 05200015 05200018 05200019 05200022 05200023 05200025 05200026 05200027 05200028 05200029 05200030 05200040 05200041 PROJ0763

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5/21/2009	ML091480603	William States Lee III, Units 1 and 2 - AP1000 Combined License Application for the Response to Request for Additional Information (RAI No. 1657), Ltr# WLG2009.05-04.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
5/22/2009	ML091470055	William States Lee III, Units 1 and 2 - Summary Identification of Concurrence with Standard Content RAIs.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
5/27/2009	ML091470226	2009/05/27 Lee RAI for SER - RAI LETTER NO. 069 RELATED TO SRP SECTION: 02.04.03 - PROBABLE MAXIMUM FLOOD (PMF) FOR THE W.S. LEE UNITS 1 AND 2 COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
6/1/2009	ML112650823	Brockington 2009a Cultural Resources Survey of the Lee Nuclear Station Utilities Project.	Report, Miscellaneous	Brockington & Associates, Inc	NRC/NRO	05200018 05200019
6/2/2009	ML091560104	William States Lee III, Units 1 and 2 - Response to Request for Additional Information (RAI No. 2098) Re: AP1000 Combined License Application.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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6/2/2009	ML112280643	USACE Cleveland County Letters.	Letter	US Dept of the Army, Corps of Engineers, Wilmington District	Cleveland County Sanitary District, NC NRC/NRO	05200018 05200019
6/3/2009	ML091560105	William States Lee III, Units 1 and 2 - Development of Horizontal and Vertical Site Specific Hazard Consistent Uniform Hazard Response Spectra at the Lee Nuclear Station Unit 1, Revision 1.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
6/10/2009	ML112650389	FBI 2006.	Graphics incl Charts and Tables	US Dept of Justice, Federal Bureau of Investigation (FBI)	NRC/NRO	05200018 05200019
6/11/2009	ML091660230	William States Lee III Nuclear Station, Units 1 and 2 - Combined License Application Response to Request for Additional Information on Letter (RAI No. 2563) Ltr# WLG2009.06-04.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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6/12/2009	ML091670375	William States Lee, Units 1 and 2, Changes to the Fitness for Duty Program Information, Physical Security During Construction, and Physical Security Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC).	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
6/12/2009	ML091670459	William States Lee III, Units 1 & 2 AP1 000 Combined License Application, Supplemental Response to Request for Additional Information No. 1826, Ltr# WLG2009.06-03.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
6/17/2009	ML091680085	2009/06/16-Notice of Withdrawal of Notice of Appearance for Kathryn M. Sutton, Paul M. Bessette, Jonathan M. Rund.	Legal-Correspondence/Miscellaneous	Duke Energy Carolinas, LLC Morgan, Lewis & Bockius, LLP	NRC/OCM	05200018 05200019
6/18/2009	ML091690038	2009/06/18 Lee RAI for SER - FW: LEE-RAI-LTR-070 RELATED TO SRP SECTION: 02.04.12 - GROUNDWATER FOR THE W.S. LEE COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019

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6/19/2009	ML091700401	2009/06/19-Notice of Appearance of David R. Lewis on Behalf of Duke Energy Carolinas, LLC.	Legal-Notice of Appearance Legal-Pleading	Duke Energy Carolinas, LLC Pillsbury, Winthrop, Shaw, Pittman, LLP	NRC/OCM	05200018 05200019
6/19/2009	ML091750090	William States Lee III Nuclear Station - AP1000 Combined License Application, Response to Request for Additional Information (RAI No. 2680) Ltr# WLG2009.06-06.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC Duke Energy Corp	NRC/Document Control Desk NRC/NRO	05200018 05200019
6/22/2009	ML091730493	2009/06/22 Lee RAI for SER - LEE-RAI-LTR-072.doc	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019

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6/25/2009	ML091760494	Presentation by David Matthews at the June 4, 2009 Public Meeting with New Plant Working Group.	Meeting Briefing Package/Handouts Slides and Viewgraphs	NRC/NRO/DNRL		

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7/15/2009	ML091960539	2009/07/15 Lee RAI for SER - REQUEST FOR ADDITIONAL INFORMATION LETTER NO. 071 RELATED TO SRP SECTION 19 FOR THE WILLIAM STATES LEE III, UNITS 1 and 2 COMBINED LICENSE APPLICATION	Request for Additional Information (RAI)	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
7/16/2009	ML091970441	2009/07/16 Lee RAI for SER - LEE-RAI-LTR-073.doc	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
7/17/2009	ML092030108	William States Lee III Nuclear Station, Units 1 and 2 - Combined License Application, Transmittal of Post-Demolition Field and Laboratory Basemat Test Results, Lee Nuclear Station Unit 1 Concrete Basemat Evaluation on Ltr# WL12009.07-03.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
7/17/2009	ML092030129	William States Lee III, Units 1 & 2 AP1000 Combined License Application, Response to Request for Additional Information No. 2679, Ltr# WLG2009.07-04.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
7/20/2009	ML092010286	2009/07/20 Lee (Duke) COL Hearing - Draft RAI 3337 Related SRP Section 8.2 for Lee Units 1 and 2	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019

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7/21/2009	ML092020642	2009/07/21 Lee RAI for SER - Request for Additional Information Letter No. 074 Related to SRP Section 08.02 for the William States Lee III Units 1 and 2 Combined License Application	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
7/21/2009	ML092020670	2009/07/21 Lee (Duke) COL Hearing - RAI Letter No. 074 Related SRP Section 8.2 for Lee Units 1 and 2	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
7/22/2009	ML092050070	William States Lee III, Units 1 and 2 AP1000 Combined License Application, Response to Request for Additional Information No. 538, Ltr # WLG2009.07-06.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
7/25/2009	ML090640862	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report), Rev. 1	FSAR	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
7/25/2009	ML090640875	Letter from Bryan Dolan re: Annual Update for William States Lee III Nuclear Station Units 1 and 2 Combined License Application.	Letter	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
7/25/2009	ML090640497	Duke Energy WSL III Units 1 & 2 COLA (Generic DCD Departures Report), Rev. 1 - Part 7, Departures and Exemptions Requests	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019

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7/25/2009	ML090640499	Duke Energy WSL III Units 1 & 2 COLA (Quality Assurance Program), Rev. 1 - Part 11, Enclosures, Cover	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
7/25/2009	ML090640500	Duke Energy WSL III Units 1 & 2 COLA (Quality Assurance Program), Rev. 1 - Part 11, Nuclear Plant Development Quality Assurance Program Description	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
7/25/2009	ML080460359	Duke Energy WSL III Units 1 & 2 COLA (Emergency Plan), Rev. 1	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
7/25/2009	ML073510876	Duke Energy WSL III Units 1 & 2 COLA (Environmental Report), Rev. 0	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
7/25/2009	ML090640863	Duke Energy WSL III Units 1 & 2 COLA (General and Admin Information), Rev. 1 - Part 1, General and Financial Information	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
7/25/2009	ML090640865	Duke Energy WSL III Units 1 & 2 COLA (ITAAC), Rev. 1 - Part 10, Conditions and ITAAC	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
7/25/2009	ML073511305	Duke Energy WSL III Units 1 & 2 COLA (LWA Request), Rev. 0 - LWA Request	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019

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7/25/2009	ML090640868	Duke Energy WSL III Units 1 & 2 COLA (Technical Specifications), Rev. 1 - Part 4, Williams Lee III Nuclear Station Technical Specifications	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
7/25/2009	ML090990081	William States Lee III Nuclear Station, Units 1 and 2 - Revision 1 to the Environmental Report (Part 3) and Revision 2 to Withheld Information (Part 9) for Combined License Application.	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
7/25/2009	ML090640497	Duke Energy WSL III Units 1 & 2 COLA (Generic DCD Departures Report), Rev. 1 - Part 7, Departures and Exemptions Requests	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
7/27/2009	ML092080055	2009/07/27 Lee (Duke) COL Hearing - Draft RAI 3345 Related SRP Section 8.1 for Lee Units 1 and 2	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
7/31/2009	ML092150223	2009/07/31 Lee RAI for SER - RAI LETTER NO. 75 RELATED TO THE SRP SECTION 01-INTRODUCTION AND INTERFACES FOR THE WILLIAMS STATES LEE III UNITS 1 AND 2 COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019

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7/31/2009	ML092150224	2009/07/31 Lee RAI for SER - RAI LETTER NO. 75 RELATED TO THE SRP SECTION 01-INTRODUCTION AND INTERFACES FOR THE WILLIAMS STATES LEE III UNITS 1 AND 2 COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
7/31/2009	ML092170268	William States Lee III, AP1000 Combined License Application, Supplemental Information Addressing Potential for Reservoir-Induced Seismicity Associated with Off-Site Water Storage Ltr. #WLG2009.07-05.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
7/31/2009	ML092170378	William States Lee III Nuclear Station, Units 1 and 2 - Response to Request for Additional Information (RAI No. 2685) Ltr# WLG2009.07-09.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC Duke Energy Corp	NRC/Document Control Desk NRC/NRO	05200018 05200019
7/31/2009	ML092170642	William States Lee III, Units 1 and 2 - AP1000 Combined License Application, Response to Request for Additional Information Ltr# WLG2009.08-01, RAI 119, Terrestrial Ecology.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC Duke Energy Corp	NRC/Document Control Desk NRC/NRO	05200018 05200019

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7/31/2009	ML092230151	William States Lee III Nuclear Station Units 1 and 2, Supplemental Information Addressing Hydrology Associated with Off-Site Water Storage.	Letter	Duke Energy Corp	NRC/Document Control Desk NRC/NRO	05200018 05200019
7/31/2009	ML092710472	William S. Lee III - 230kV and 525kV Transmission Line Ecological Survey Report, Introduction through Appendix A (Part 1 of 2).	Environmental Report	HDR/DTA	Duke Energy Carolinas, LLC NRC/NRO	05200018 05200019
7/31/2009	ML092710473	William S. Lee III - 230kV and 525kV Transmission Line Ecological Survey Report, Appendix A (Part 2 of 2) through Appendix B (Part 1 of 2).	Environmental Report	HDR/DTA	Duke Energy Carolinas, LLC NRC/NRO	05200018 05200019
7/31/2009	ML092710474	William S. Lee III - 230kV and 525kV Transmission Line Ecological Survey Report, Appendix B(Part 2 of 2).	Environmental Report	HDR/DTA	Duke Energy Carolinas, LLC NRC/NRO	05200018 05200019
7/31/2009	ML092730481	William States Lee III, Response to Request for Additional Information, Attachment 63S-2, Computational Fluid Dynamics Thermal Modeling Lee Nuclear Station.	Response to Request for Additional Information (RAI)	GeoSyntec Consultants MMI Engineering, LTD	NRC/NRO	05200018 05200019

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8/5/2009	ML092170258	2009/08/05 Lee RAI for SER - RAI LETTER NO. 75 RELATED TO THE SRP SECTION 01-INTRODUCTION AND INTERFACES FOR THE WILLIAMS STATES LEE III UNITS 1 AND 2 COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
8/5/2009	ML092170363	2009/08/05 Lee RAI for SER - RAI LETTER NO. 75 RELATED TO THE SRP SECTION 01-INTRODUCTION AND INTERFACES FOR THE WILLIAMS STATES LEE III UNITS 1 AND 2 COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
8/6/2009	ML092220179	William States Lee III, Loss of Large Areas of the Plant Due to Explosions of Fire (LOLA) Mitigative Strategies Description and Plans - Reviewer's Guide Ltr. # WLG2009.08-02.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
8/14/2009	ML092310276	William States Lee III Nuclear Station, AP1000 Combined License Application, Response to Request for Additional Information Ltr. #WLG2009.08-06.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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8/17/2009	ML092310486	William States Lee III Nuclear Station Units 1 & 2 AP1000 Combined License Application re Partial Response to Request for Additional Information (RAI No. 2744) Ltr. #WLG2009.08-03.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC Duke Energy Corp	NRC/Document Control Desk NRC/NRO	05200018 05200019
8/18/2009	ML092240458	Environmental Project Manager Change for the Combined License Environmental Review for William States Lee III Nuclear Station, Units 1 and 2.	Letter	NRC/NRO/DSER/RAP3	Duke Power Co	05200018 05200019
8/20/2009	ML092360176	William States Lee III, Units 1 & 2 Combined License Application, Response to Request for Additional Information No. 3337 Ltr# WLG2009.08-05.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
8/21/2009	ML092390268	FEMA, Request an Extension to the Schedule for Interim Finding, Report for Open Items for the William States Lee III Nuclear Plant Combined License Application.	Letter	US Dept of Homeland Security US Federal Emergency Mgmt Agency (FEMA)	NRC/NSIR/DPR/DDIR/OB	05200018 05200019

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8/21/2009	ML092610615	2009/08/21 Lee (Duke) COL Hearing - W. S. Lee Units 1 and 2 Endorsement of Bellefonte RAI 8.01-02 response (Letter 25 Supplement 1)	E-Mail	- No Known Affiliation	NRC/NRO/DNRL/NWE1	05200018 05200019
8/24/2009	ML092380163	William States Lee III AP1000 Combined License Application, Departures Report Update, Ltr. #WLG2009.08-07.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
8/24/2009	ML093140390	"Herpetological Survey of the W.S. Lee III Nuclear Station, South Carolina."	Environmental Report	Davidson College	NRC/NRO	05200018 05200019
8/24/2009	ML093140391	"Herpetological Surveys of the Railroad Corridor Between Gaffney & the W.S. Lee III Nuclear Station, Cherokee County, SC."	Environmental Report	Davidson College	NRC/NRO	05200018 05200019
8/24/2009	ML093491112	Attachment 89SB-1, "Herpetological Survey of London Creek, Cherokee County, South Carolina & Its Vicinity," Enclosure 1.	Environmental Report	Davidson College	NRC/NRO	05200018 05200019
8/31/2009	ML093130453	"An Avian Survey of the Railroad Corridor Between Gaffney & W.S. Lee III Nuclear Station, Cherokee County, SC."	Environmental Report	HDR/DTA	Duke Energy Carolinas, LLC NRC/NRO	05200018 05200019
8/31/2009	ML093140392	"Avian Survey of the William S. Lee III Nuclear Station; Cherokee County, SC."	Environmental Report	HDR/DTA	Duke Energy Carolinas, LLC NRC/NRO	05200018 05200019

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9/3/2009	ML093270025	09/03/09, Slides, Meeting Summary, DCWG Re: Implementation of DC/COL-ISG-08 "Necessary Content of Plant-Specific Technical Specifications."	Meeting Summary Slides and Viewgraphs	NRC/NRO/DNRL/NWE1		05200006 05200014 05200015 05200018 05200019 05200022 05200023 05200025 05200026 05200027 05200028 05200030 05200040 05200041 PROJ0763
9/4/2009	ML092170267	Update on the William States Lee III Nuclear Station Units 1 and 2 Combined License	Letter	NRC/NRO/DSER/RAP3	Duke Energy Corp Duke Power Co	05200018 05200019

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		Application Environmental Review.				
9/14/2009	ML092580474	William States Lee III, Units 1 & 2 AP1000 Combined License Application, Response to Request for Additional Information Ltr# WLG2009.09-04.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
9/14/2009	ML092580475	William States Lee III Nuclear Station - AP1000 Combined License Application, Response to Request for Additional Information Ltr# WLG2009.09-03.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
9/14/2009	ML092590318	William States Lee III Nuclear Station, Units 1 & 2 - 2009 Integrated Resource Plan, Ltr # WLG2009.09-02.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
9/23/2009	ML092710039	Transmittal of William States Lee III Nuclear Station, Units 1 and 2 - Combined License Application for the Response to Request for Additional Information on Environmental Review Ltr# WLG2009.09-07.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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9/23/2009	ML092710471	William States Lee III, Units 1 and 2 - Response to Request for Additional Information RAI 90, Ecology Re: AP1000 Combined License Application.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Processing Center NRC/NRO	05200018 05200019
9/24/2009	ML092710228	William States Lee III Nuclear Station, Units 1 and 2, AP1000 Combined License Application, Response to Request for Additional Information Ltr# WLG2009.09-06, Regarding Ecology.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
9/24/2009	ML092710230	William States Lee III Nuclear Station, Units 1 and 2 - AP1000 Combined License Application for the Response to Request for Additional Information (RAI No. 3208).	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC Duke Energy Corp	NRC/Document Control Desk NRC/NRO	05200018 05200019
9/24/2009	ML092730480	William States Lee III, Units 1 and 2, AP1000 Combined License Application, Response to Request for Additional Information Ltr# WLG2009.09-10.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
9/24/2009	ML092810254	William States Lee III Nuclear Station - Appendix B (Supplement) Agency Correspondence, Supplemental Water Source.	Letter Map	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019

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9/24/2009	ML092810255	William States Lee III - AP1000 Combined License Application, Response to Request for Additional Information, Ltr. #WLG2009.09-05.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
9/24/2009	ML092810257	Supplement to Revision 1 of the William States Lee III Nuclear Station COL Application, Part 3 Applicant's Environmental Report, Construction and Operation of Make-Up Pond C.	Report, Miscellaneous	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
9/28/2009	ML091380400	William States Lee III Nuclear Station Request for Withholding Information From Public Disclosure.	Letter Proprietary Information Review	NRC/NRO/DNRL/NWE1	Duke Energy Carolinas, LLC	05200018 05200019
9/29/2009	ML092730446	William States Lee III, Units 1 and 2 - Supplemental Information Regarding Lee Nuclear Station Unit 1 Northwest Corner Foundation Rock.	Letter	Duke Energy Carolinas, LLC Duke Energy Corp	NRC/Document Control Desk NRC/NRO	05200018 05200019
10/1/2009	ML092780249	William States Lee III Nuclear Station, Units 1 and 2 - Combined License Application Response to Request for Additional Information on Ltr. # WLG2009.09-12.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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10/1/2009	ML112650824	Brockington 2009b Cultural Resources Survey of the Proposed London Creek Reservoir (Make-Up Pond C) and Water Pipeline.	Report, Miscellaneous	Brockington & Associates, Inc	Duke Energy Carolinas, LLC NRC/NRO	05200018 05200019
10/6/2009	ML092790323	2009/10/06 Lee RAI for SER - RAI Letter 078 SRP 13.6 for W. S. Lee Units 1 and 2	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
10/6/2009	ML092790325	2009/10/06 Lee RAI for SER - RAI Letter 079 SRP 13.6 for W.S. Lee units 1 and 2 Cola	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
10/16/2009	ML092930116	William States Lee III Nuclear Station Units 1 and 2, Combined License Application for the Response to Request for Additional Information on Environmental Report Ltr# WLG2009.10-01.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
10/21/2009	ML092940581	2009/10/21 Lee RAI for SER - LEE-RAI-LTR-079 RELATED TO SRP SECTION: 13.6 PHYSICAL SECURITY FOR THE W.S. LEE COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
10/22/2009	ML092950483	2009/10/22 Lee RAI for SER - LEE-RAI-LTR-078(1) RELATED TO SRP SECTION 13.6 PHYSICAL SECURITY FOR THE W.S. LEE 1 & 2 COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019

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10/27/2009	ML093000092	2009/10/27 Lee (Duke) COL Hearing - Conference Call Summary - October 26, 2009 - William States Lee Nuclear Station Units 3 and 4 COLA - RAIs related to SRP Chapter 13	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
10/30/2009	ML093080101	William States Lee III, Units 1 and 2, Supplemental Response to Request for Additional Information (RAI Nos. 1874, 1881, and 2098), Ltr# WLG2009.10-02.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
11/1/2009	ML093050007	2009/11/01 Lee RAI for SER - LEE-RAI-LTR-080.doc	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
11/1/2009	ML093050008	2009/11/01 Lee RAI for SER - LEE-RAI-LTR-080 Related to SRP 02.03.04, 05 for the W.S. Lee Unit 1 and 2 COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
11/1/2009	ML093050010	2009/11/01 Lee RAI for SER - LEE-RAI-LTR-081 RELATED TO SRP 02.03.01, 02 FOR THE W.S. LEE Units 1 and 2 COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
11/1/2009	ML093050011	2009/11/01 Lee RAI for SER - LEE-RAI-LTR-080 RELATED TO SRP 02.03.04,05 For the W.S. LEE Units 1 AND 2 COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019

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11/1/2009	ML093050020	2009/11/01 Lee RAI for SER - RAI Letter No. 82 Related to SRP 02.03.02 for the W.S. Lee units 1& 2 COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
11/2/2009	ML093060265	2009/11/02 Lee RAI for SER - REQUEST FOR ADDITIONAL INFORMATION LETTER NO.083 RELATED TO SRP SECTION 13.3 FOR THE WILLIAM STATES LEE III UNITS 1 AND 2 COMBINED LICENSE APPLICATION	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
11/2/2009	ML093060285	2009/11/02 Lee (Duke) COL Hearing - REQUEST FOR ADDITIONAL INFORMATION LETTER NO.083 RELATED TO SRP SECTION 13.3 FOR THE WILLIAM STATES LEE III UNITS 1 AND 2 COMBINED LICENSE APPLICATION	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
11/2/2009	ML093130451	William States Lee III Nuclear Station, Response to Request for Additional Information Ltr# WLG2009.11-01 re RAI 89, Ecology.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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11/3/2009	ML093070196	SRM-M091103A - Affirmation Session: I. SECY-09-0033 - Duke Energy Carolinas, LLC (William States Lee III Nuclear Station, Unit 1 and 2); TVA (Bellefonte Nuclear Plant, Units 3 and 4)-- Referred Rulings on Contention Admissibility.	Commission Staff Requirements Memo (SRM)	NRC/SECY	NRC/OCAA	05000438 05000439 05200018 05200019
11/3/2009	ML093070568	M091103A-Affirmation Session: I - SECY-09-0033 - Duke Energy Carolinas, LLC (William States Lee III Nuclear Station, Unit 1 and 2); Tennessee Valley Authority (Bellefonte Nuclear Plant, Units 3 and 4)- Referred Rulings on Contention Admissibility.	Commission Meeting Transcript/Exhibit	NRC/OCM		05200014 05200015 05200018 05200019
11/3/2009	ML093070689	2009/11/03-CLI-09-21, Memorandum and Order, Commission Two Contention Admissibility Rulings Concerning Consideration in COL Applications of Certain Environmental Impacts Relevant to Greenhouse Gas Emissions.	Legal-Memorandum and Order	NRC/SECY	NRC/Chairman NRC/OCM	05200018 05200019

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11/3/2009	ML093070753	2009/11/03 Lee RAI for SER - LEE-RAI-LTR-076 related to SRP Section 02.05.02 Vibratory Ground Motion for the W.S. Lee Unit 1 & 2 COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
11/4/2009	ML093080146	2009/11/04 Lee RAI for SER - LEE-RAI-LTR-084.doc	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
11/4/2009	ML093080312	2009/11/04 Lee RAI for SER - LEE-RAI-LTR-085 RELATED TO SRP SECTION: 02.05.03 - SURFACE FAULTING FOR THE W.S.LEE UNIT 1 & 2 COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
11/5/2009	ML093090013	2009/11/05 Lee RAI for SER - LEE-RAI-LTR-086 RELATED TO SRP SECTION: 02.05.01 - BASIC GEOLOGICAL FOR THE W.S. LEE UNITS 2 AND 3 COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
11/11/2009	ML093170198	William States Lee III Nuclear Station, Units 1 and 2 - Combined License Application for the Response to Request for Additional Information Ltr# WLG2009.11-03.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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11/12/2009	ML093210477	William States Lee III, Units 1 and 2, AP1000 Combined License Application for the Response to Request for Additional Information (RAI No. 2686) Ltr# WLG2009.11-05.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
11/12/2009	ML093220237	William States Lee, Units 1 and 2, AP1000 Combined License Application for Non-Public Response to Request for Additional Information (RAI Nos. 3529, 3530, 3531, 3534, 3535, 3528, 3536, 3537, 3539, 3540, 3542, 3542, and 3545), Ltr# WLG2009.11-02.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
11/18/2009	ML093220974	2009/11/18 Lee RAI for SER - LEE-RAI-LTR-078 RELATED TO SRP SECTION 13.6 FOR THE W.S. LEE UNits 1 & 2 COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
11/18/2009	ML093220976	2009/11/18 Lee RAI for SER - LEE-RAI-LTR-078(1) RELATED TO SRP 13.6 PHYSICAL SECURITY FOR THE W.S. LEE UNITS 2 & 3 COLA (Corrected TYPO to change Q 13.06.2 to 13.06.23)	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019

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11/18/2009	ML093491113	Attachment 89SB-2, "The Fish Community of London Creek; Cherokee County, SC, IN 2008-2009," Enclosure 1	Environmental Report	Duke Energy Corp	NRC/NRO	05200018 05200019
11/19/2009	ML093280308	William States Lee III Nuclear Station - Revised Response to Request for Additional Information (RAI Nos. 1922) Ltr # WLG2009.11-04.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
11/24/2009	ML093280269	2009/11/24 Lee RAI for SER - LEE-RAI-LTR-087 RELATED TO SRP SECTION: 03.07.01-SEISMIC DESIGN PARAMETERS FOR THE W.S. LEE UNITS 1 & 2 COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
11/30/2009	ML093490248	Enclosure 1 - Lee Nuclear Station, Transportation Assessment Executive Summary, Attachment 28S.	Report, Miscellaneous	Kimley-Horn & Associates, Inc	Duke Energy Corp NRC/NRO	05200018 05200019
12/3/2009	ML093380647	AP1000 Combined License Application for the William States Lee III Nuclear Station Units 1 and 2 Response to Request for Additional Information Ltr# WLG2009.12-01.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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12/3/2009	ML093420405	William States Lee III Nuclear Station, Units 1 and 2 - Combined License Application for the Response to Request for Additional Information on Environmental Review Ltr# WLG2009.12-04.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
12/11/2009	ML093490247	William States Lee III Nuclear Station, Units 1 and 2 - Response to Request for Additional Information, Ltr# WLG2009.12-05.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
12/11/2009	ML093491111	William States Lee III Nuclear Station, Units 1 and 2 - Response to Request for Additional Information Ltr# WLG2009.12-07.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC Duke Energy Corp	NRC/Document Control Desk NRC/NRO	05200018 05200019
12/11/2009	ML093491114	Attachment 89SB-3, "A Biological Survey for Breeding & Migratory Avian Species Associated With London Creek, Cherokee County, SC," Enclosure 1.	Environmental Report	Devine, Tarbell & Associates, Inc	Duke Energy Carolinas, LLC NRC/NRO	05200018 05200019
12/11/2009	ML093491115	Attachment 89S-4, "Macroinvertebrate Surveys on London Creek, Cherokee Co, South Carolina," Enclosure 1.	Environmental Report	Duke Energy Corp	NRC/NRO	05200018 05200019

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12/11/2009	ML093491117	Attachment 89SB-5, "Mammals of the Make-Up Pond C Project Area, Cherokee County, South Carolina," Enclosure 1.	Environmental Report	Univ of North Carolina - Wilmington	Duke Energy Corp NRC/NRO	05200018 05200019
12/11/2009	ML093491118	Attachment 89SB-6, "A Botanical Inventory of Make-Up Pond C Study Area, Cherokee County, SC," Enclosure 1.	Environmental Report	Terra Incognita	Duke Energy Corp NRC/NRO	05200018 05200019
12/15/2009	ML093491055	12/15/09 Letter to Honorable Thomas R. Carper Responding to his 10/15/09 Letter Regarding the U.S. NRC's Method for Reviewing New Nuclear Reactor Applications.	Letter Congressional Correspondence	NRC/OCA	US SEN, Comm on Environment & Public Works	05200011 05200014 05200015 05200017 05200018 05200019 05200020 05200021 05200037
12/16/2009	ML093500394	2009/12/16 Lee (Duke) COL Hearing - Telcon Summary - Telcon with Lee 12/15/09	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019

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12/18/2009	ML093570280	William States Lee III, Units 1 and 2 - AP1000 Combined License Application, Summary Identification of Concurrence with Standard Content RAIs and Safety Evaluation Report Open Items.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
12/18/2009	ML112800522	FERC 2009 Upper Pacolet Federal Register Notice.	Report, Miscellaneous	US Environmental Protection Agency (EPA)	NRC/NRO	05200018 05200019
12/18/2009	ML112800523	FERC 2009 Upper Pacolet Scoping Document.	- No Document Type Applies	US Federal Energy Regulatory Commission	NRC/NRO	05200018 05200019
12/31/2009	ML112650819	ACC 2009 Cultural Resources Survey of Proposed William States Lee III Nuclear Station 230 kV and 525 kV Transmission Lines.	Environmental Monitoring Report	Archaelogical Consultants of the Carolinas, Inc (ACC, Inc) Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
1/1/2010	ML112800270	USFWS 2010 Sunflower.	Environmental Monitoring Report	US Dept of Interior, Fish & Wildlife Service	NRC/NRO	05200018 05200019

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1/1/2010	ML113260223	NPCC 2010 Sixth Power.	Technical Paper	Northwest Power & Conservation Council	NRC/NRO	05200018 05200019
1/5/2010	ML092660080	Duke Energy Carolinas, LLC, William States Lee III Nuclear Station Units 1 and 2 Combined Application License Review Schedule.	Letter	NRC/NRO/DNRL	Duke Energy Carolinas, LLC	05200018 05200019
1/8/2010	ML100120287	William States Lee III Nuclear Station, Combined License Application for the William States Lee III Nuclear Station Units 1 and 2, Response to Request for Additional Information on Environmental Review, Ltr# WLG2010.01-01.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
1/11/2010	ML092880776	09/03/2009 Summary of Category II Public Meeting with AP1000 Design-Centered Working Group to Discuss The Implementation of DC/COL-ISG-8, "Necessary Content of Plant-Specific Technical Specifications.	Meeting Agenda Meeting Summary Memoranda	NRC/NRO/DNRL/NWE1	NRC/NRO/DNRL/NWE1	05200014 05200015 05200018 05200019 05200022 05200023 05200025 05200026

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						05200027 05200028 05200029 05200030 05200040 05200041
1/14/2010	ML112800400	EPA 2010 Greenhouse Gas.	Report, Miscellaneous	US Environmental Protection Agency (EPA)	NRC/NRO	05200018 05200019
2/4/2010	ML100350591	2010/02/04 Lee (Duke) COL Hearing - Telcon Summary - Telcon with Lee 8/27/2009	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
2/19/2010	ML112800524	NCDEH Broad River.	Environmental Report	State of NC, Div of Environmental Health	NRC/NRO	05200018 05200019
2/19/2010	ML112800526	NCDEH 2010 Forest City.	Environmental Report	State of NC, Div of Environmental Health	NRC/NRO	05200018 05200019

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2/19/2010	ML113260222	NCDEH 2010 Kings Mountain.	Environmental Monitoring Report	State of NC, Dept of Environment & Natural Resources, Div of Water Quality	NRC/NRO	05200018 05200019
2/22/2010	ML100550192	William States Lee III, Units 1 & 2, Summary Identification of Concurrence With Standard Content RAIs Letter #WLG2010.02-05.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/22/2010	ML100550350	William States Lee III Nuclear Station, Units 1 & 2 - Transmittal of Unit 1 Foundation Input Response Spectra (FIRS) Horizontal and Vertical Component Analysis, Ltr# WLG2010.02-01.	Letter Report, Miscellaneous	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/25/2010	ML100620211	Update for William States Lee III Nuclear Station Units 1 and 2 Combined License Application. Includes the Annual update of the Docketed Final Safety Analysis Report and the Semi-Annual Update of the Departures Report (COL Application, Part 2 & Part 7).	Letter License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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2/25/2010	ML100620317	Duke Energy WSL III Units 1 & 2 COLA (Generic DCD Departures Report), Rev. 2 - Part 7, Departures and Exemptions Requests	Generic DCD Departures Report License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/25/2010	ML100620322	Duke Energy WSL III Units 1 & 2 COLA (Quality Assurance Program), Rev. 2 - Part 11, Enclosures, Cover	License-Application for Combined License (COLA) Quality Assurance Program	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/25/2010	ML100620324	Duke Energy WSL III Units 1 & 2 COLA (Quality Assurance Program), Rev. 2 - Part 11, Nuclear Plant Development Quality Assurance Program Description	License-Application for Combined License (COLA) Quality Assurance Program	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/25/2010	ML100620327	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report), Rev. 2 - FSAR Master Table of Contents	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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2/25/2010	ML100620328	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report), Rev. 2 - FSAR Chapter 01 Introduction and General Description of the Plant - Sections 01.01 - 01.10, Appendices 1A, 1AA, 1B	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/25/2010	ML100620329	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report), Rev. 2 - FSAR Chapter 01 Figure 1.1-201	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/25/2010	ML100620330	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report), Rev. 2 - FSAR Chapter 01 Figure 1.1-202	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/25/2010	ML100620331	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report), Rev. 2 - FSAR Chapter 01 Figure 1.2-201	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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2/25/2010	ML100620332	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report), Rev. 2 - FSAR Appendix 2AA, Attachments	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/25/2010	ML100620334	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report), Rev. 2 - FSAR Appendix 2BB, Attachments	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/25/2010	ML100620336	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report), Rev. 2 - FSAR Appendix 2CC, Attachments	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/25/2010	ML100620337	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report), Rev. 2 - FSAR Chapter 02 Site Characteristics - Sections 02.00 - 02.05, Appendices 2AA - 2CC	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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2/25/2010	ML100620338	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report), Rev. 2 - FSAR Chapter 02 Figure 2.1-201	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/25/2010	ML100620339	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report), Rev. 2 - FSAR Chapter 02 Figure 2.1-202	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/25/2010	ML100620340	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report), Rev. 2 - FSAR Chapter 02 Figure 2.1-203	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/25/2010	ML100620341	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report), Rev. 2 - FSAR Chapter 02 Figure 2.1-204	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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2/25/2010	ML100620342	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report), Rev. 2 - FSAR Chapter 02 Figure 2.1-205	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/25/2010	ML100620343	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report), Rev. 2 - FSAR Chapter 02 Figure 2.1-206	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/25/2010	ML100620748	Duke Energy WSL III Units 1 & 2 COLA (ITAAC), Rev. 2 - Part 10, Conditions and ITAAC	Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/25/2010	ML100620750	Duke Energy WSL III Units 1 & 2 COLA (Technical Specifications), Rev. 2 - Part 4, Williams Lee III Nuclear Station Technical Specifications	License-Application for Combined License (COLA) Technical Specifications	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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2/25/2010	ML100620745	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report), Rev. 2	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
2/25/2010	ML100620746	Duke Energy WSL III Units 1 & 2 COLA (General and Admin Information), Rev. 2 - Part 1, General and Financial Information	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
2/25/2010	ML100620748	Duke Energy WSL III Units 1 & 2 COLA (ITAAC), Rev. 2 - Part 10, Conditions and ITAAC	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
2/25/2010	ML100620750	Duke Energy WSL III Units 1 & 2 COLA (Technical Specifications), Rev. 2 - Part 4, Williams Lee III Nuclear Station Technical Specifications	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
3/3/2010	ML112800408	EPA 2010 Global Greenhouse.	Report, Miscellaneous Graphics incl Charts and Tables	US Environmental Protection Agency (EPA)	NRC/NRO	05200018 05200019
3/4/2010	ML100640642	William States Lee III, Units 1 and 2 - Response to NRC Regulatory Issue Summary (RIS) 2010-01, Ltr # WLG2010.03-01.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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3/4/2010	ML100690444	William States Lee III, Units 1 and 2 - AP1000 Combined License Application, Supplemental Response to Request for Additional Information No. 50, Ltr# WLG2010.03-02.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
3/11/2010	ML100710539	Task Order No. 091 Under Delivery Order No. NRC-42-07-036.	ACQ-Contract Task Order	NRC/ADM/DC	Information Systems Labs, Inc ISL, Inc	05200018 05200019
3/12/2010	ML100760097	William States Lee III, Units 1 and 2 Combined License Application, Response to Request for Additional Information No. 3621, Ltr. #WL12010.03-04.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
3/12/2010	ML100850377	William States Lee III, Units 1 and 2, Submittal of Update Roadmap Ltr# WLG2010.03-05.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
3/22/2010	ML100810316	2010/03/22 Lee (Duke) COL Hearing - Draft RAI 4507 Related SRP Section 9.3.3 for Lee Units 1 and 2	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019

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3/24/2010	ML100890526	Oconee, Independent Spent Fuel Storage Installation, McGuire, Catawba, and William States Lee III, Submittal of Service/Distribution Listing Update.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NMSS NRC/NRO NRC/NRR	05000269 05000270 05000287 05000369 05000370 05000413 05000414 05200018 05200019 07200004
3/29/2010	ML100880061	2010/03/29 Lee RAI for SER - Request for Additional Information Letter No. 089 Related to SRP Section 09.03.03 for the William States Lee III Units 1 and 2 Combined License Application	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
3/29/2010	ML100880080	2010/03/29 Lee (Duke) COL Hearing - RAI Letter No. 089 Related SRP Section 9.3.3 for Lee Units 1 and 2	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019

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3/31/2010	ML100920024	William States Lee III Nuclear Station - AP1000 Combined License Application, Editorial Text Changes to the Environmental Report.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
3/31/2010	ML112800518	SCDHEC 2010 Annual Report on Reportable Conditions.	Report, Miscellaneous	State of SC, Dept of Health & Environmental Control	NRC/NRO	05200018 05200019

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4/5/2010	ML100910533	U.S. Nuclear Regulatory Commission Staff Review of the Department of Homeland Security Consultation Reports.	Memoranda	NRC/NSIR/DSP/DDRSR/RSPLB	NRC/NSIR/DSP/DDRSR/RSPLB	05000390 05000391 05200012 05200013 05200014 05200015 05200016 05200017 05200018 05200019 05200022 05200023 05200024 05200025 05200026 05200027 05200028

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4/6/2010	ML100640699	Trip Report regarding 2010 The International Conference on Simulation Technology for Power Plants February 22-25, 2010 Handout - Regulatory Guide 1.149 Update.	Meeting Briefing Package/Handouts Slides and Viewgraphs Trip Report	NRC/NRO/DCIP		05200001 05200006 05200010 05200011 05200012 05200014 05200015 05200016 05200017 05200018 05200019 05200020 05200022 05200023 05200024 05200025 05200026

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4/6/2010	ML101060138	William States Lee III Nuclear Station, Units 1 and 2 - AP1000 Combined License Application, Response to Request for Additional Information (RAI No. 3726 and 3727) Ltr. WLG2010.04-01.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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4/9/2010	ML100960417	New Reactor Information Slides.	Meeting Briefing Package/Handouts Slides and Viewgraphs	NRC/NRO/DCIP/CHPB		05200001 05200006 05200010 05200011 05200012 05200014 05200015 05200016 05200017 05200018 05200019 05200020 05200022 05200023 05200024 05200025 05200026

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4/14/2010	ML101050300	Modification No. 001 to Task Order No. 007 Under Contract No. NRC-42-07-036.	ACQ-Contract Task Order Modification	NRC/ADM/DC	Information Systems Labs, Inc	05200018 05200019
4/14/2010	ML101090072	William States Lee III Nuclear Station, AP1000 Combined License Application and Response to Request for Additional Information (RAI No. 3798 and 3799).	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
4/14/2010	ML101090314	William States Lee III, Units 1 and 2, AP1000 Combined License Application, 2009 Integrated Resource Plan, Revision 1, Ltr# WLG2010.04-03.	Annual Operating Report Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
5/4/2010	ML101260120	Combined License Application for the William States Lee III Nuclear Station Units 1 and 2 10 CFR 50.46 Annual Report.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
5/4/2010	ML101260121	William States Lee III Nuclear Station, Units 1 and 2 - Response to Request for Additional Information re Letter 089.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
5/18/2010	ML093420654	Letter to Duke Energy Regarding the Federal Register Notice of Intent to Conduct Scoping for the Supplement to the William States Lee III ER.	Federal Register Notice Letter	NRC/NRO/DSER	Duke Energy Nuclear, LLC	05200018 05200019

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5/18/2010	ML093430226	Federal Register Notice of Intent to Conduct Scoping for the Supplement to the Lee ER.	Federal Register Notice	NRC/NRO/DSER		05200018 05200019
5/24/2010	ML093480445	Environmental Scoping Letter to South Carolina State Historic Preservation Office Regarding Make-Up Pond C for William States Lee Nuclear Station, Units 1 and 2.	Letter Map	NRC/NRO/DSER/RAP3	State of SC, Dept of Archives and History	05200018 05200019
5/24/2010	ML093560024	Supplemental Scoping Letter to ACHP Regarding the Addition of Make-Up Pond C to William States Lee III Nuclear Station, Units 1 and 2 Combined License Application.	Letter Map	NRC/NRO/DSER/RAP3	US Advisory Council On Historic Preservation	05200018 05200019
5/24/2010	ML093570175	Request for Participation in a Supplemental Scoping Process Regarding the Addition of a Third Cooling Water Reservoir for the William States Lee III Nuclear Station, Units 1 & 2 Combined License Application (DNR Project 0742).	Letter Map	NRC/NRO/DSER/RAP3	State of SC, Dept of Natural Resources	05200018 05200019 PROJ0742
5/24/2010	ML093580019	Supplement Scoping Letter to FWS Regarding Make-Up Pond C for the William States Lee III Nuclear Station, Units 1 and 2	Letter Map	NRC/NRO/DSER/RAP3	US Dept of Interior, Fish & Wildlife Service	05200018 05200019

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		Combined License Application.				
5/24/2010	ML101190491	Letter to Linville - NC Wildlife Resources Commission Request for Participation in Supplemental Scoping Regarding Additional Cooling Water Reservoir for William States Lee Nuclear Station, Units 1 and 2.	Letter Map	NRC/NRO/DSER/RAP3	State of NC, Wildlife Resources Commission	05200018 05200019
5/24/2010	ML101190500	Request for Participation in a Supplemental Scoping Process Regarding the Addition of a Third Cooling Water Reservoir for the William States Lee III Nuclear Station, Units 1 and 2 Combined License Application.	Letter Map	NRC/NRO/DSER/RAP3	State of SC, Dept of Radiological Health & Environmental Control	05200018 05200019
5/24/2010	ML101200120	05/24/10 Letter to R. McConney - U.S. EPA Region 4 NEPA Program Office, Request for Participation in Supplemental Scoping Regarding Additional Cooling Water Reservoir for William States Lee Nuclear Station, Units 1 and 2.	Letter Map	NRC/NRO/DSER/RAP3	US Environmental Protection Agency (EPA)	05200018 05200019

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5/24/2010	ML101200150	Request for Participation in a Supplemental Scoping Process Regarding the Addition of a Third Cooling Water Reservoir for William States Lee III Nuclear Station, Units 1 and 2 Combined License Application.	Letter Map	NRC/NRO/DSER/RAP3	Catawba Indian Nation	05200018 05200019
5/24/2010	ML101200368	Letter Seminole - Request for Participation in Supplemental Scoping Process Regarding Addition of Third Cooling Water Reservoir for William States Lee III Nuclear Station, Units 1 & 2, Combined License Application.	Letter Map	NRC/NRO/DSER/RAP3	Seminole Tribe of Florida	05200018 05200019
5/24/2010	ML101200371	Request for Participation in a Supplemental Scoping Process Regarding the Addition of a Third Cooling Water Reservoir for the William Lee III Nuclear Station, Units 1 and 2 Combined License Application.	Letter Map	NRC/NRO/DSER/RAP3	Eastern Band of Cherokee Indians	05200018 05200019
5/24/2010	ML101200375	Letter Eastern Shawnee Tribe - Supplemental Scoping Process, William Lee III Nuclear Station, Units 1 and 2.	Letter Map	NRC/NRO/DSER/RAP3	Eastern Shawnee Tribe of Oklahoma	05200018 05200019

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5/24/2010	ML101200416	Letter - Carolina Indian Heritage Association - Supplemental Scoping Process, William States Lee III, Units 1 and 2.	Letter Map	NRC/NRO/D SER/RAP3	Carolina Indian Heritage Association	05200018 05200019
5/24/2010	ML101200435	Letter United South and Eastern Federation of Tribes - Supplemental Scoping Process, William States Lee III, Units 1 and 2.	Letter Map	NRC/NRO/D SER/RAP3	United South & Eastern Federation of Tribes	05200018 05200019
5/24/2010	ML101200443	Request for Participation in a Supplemental Scoping Process Regarding the Addition of a Third Cooling Water Reservoir for the William States Lee III, Units 1 and 2 Combined License Application.	Letter Map	NRC/NRO/D SER/RAP3	Lower Eastern Cherokee Nation of South Carolina Piedmont American Indian Association	05200018 05200019
5/24/2010	ML101200452	Request for Participation in a Supplemental Scoping Process Regarding the Addition of a Third Cooling Water Reservoir for the William States Lee III, Units 1 and 2 Combined License Application.	Letter Map	NRC/NRO/D SER/RAP3	Pine Hill Indian Community	05200018 05200019
5/25/2010	ML101440498	NRC Web Address Correction to the May 18, 2010 Federal Register Notice for William States Lee III Nuclear Station, Units	Federal Register Notice Letter	NRC/NRO/D SER	Duke Power Co	05200018 05200019

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		1 and 2 Supplemental Scoping Process.				
5/25/2010	ML101450180	Duke Energy Carolinas, LLC; William States Lee III Combined License Application; Notice of Intent to Conduct a Supplemental Scoping Process for the Supplement to the Environmental Report.	Federal Register Notice	NRC/NRO/DSER		05200018 05200019
5/26/2010	ML101460482	Press Release-10-094: NRC Seeking Additional Environmental Scoping Comments Regarding Lee New Reactor Application, Meeting June 17.	Press Release	NRC/OPA		05200018 05200019
5/27/2010	ML093580157	06/17/10 Notice of Meeting to Discuss the Scoping Process for the Supplemental Environmental Report Regarding Make-Up Pond C for the William States Lee III Nuclear Station, Units 1 and 2, Combined License Application.	Meeting Notice	NRC/NRO/DSER/RAP3	NRC/NRO/DSER/RAP3	05200018 05200019
5/27/2010	ML101330578	Environmental Project Manger Change for the Combined License Environmental Review for	Letter	NRC/NRO/DSER/RAP3	Duke Power Co	05200018 05200019

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		William States Lee III, Units 1 & 2.				
5/27/2010	ML101450387	06/17/2010 Agenda for the Public Meeting to Obtain Public Scoping Comments Relating to the Supplement to the Environmental Report for the William States Lee III Nuclear Station, Units 1 and 2 Combined Licenses Application.	Meeting Agenda	NRC/NRO/DSER/RAP3	NRC/NRO/DSER/RAP3	05200018 05200019
5/27/2010	ML101950208	Lee Nuclear Transmission Line Visual Survey, Cherokee County, SC SHPO #: 09CW0247.	Letter	State of SC, Dept of Archives and History State of SC, State Historic Preservation Office	NRC/NRO	05200018 05200019
6/1/2010	ML112650825	Brockington 2010 Cultural Resources Survey of the Proposed London Creek Reservoir, Water Pipeline, Railroad Corridor, Transmission Line, SC 329 Realignment, Railroad Culvert, Water Pipeline Additions, Spoils Areas and Road Widening's.	Environmental Report	Brockington & Associates, Inc	Duke Energy Carolinas, LLC NRC/NRO/DSER/RAP3	05200018 05200019

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6/8/2010	ML101650109	William States Lee III Units 1 & 2 - AP1000 Combined License Application Updated Information Addressing Potential for Reservoir-Induced Seismicity Associated with Off-Site Water Storage, Ltr#WLG2010.06-01.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
6/11/2010	ML101650529	William States Lee III, Units 1 & 2 AP1000 Combined License Application, Corrected Information Addressing Existing Land Use in York County, South Carolina.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
6/11/2010	ML101650706	William States Lee III Nuclear Station Units 1 and 2, Combined License Application Response to Request for Additional Information on Environmental Review (ER RAI 119, Supplement E) Ltr# WLG2010.06-02.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
6/11/2010	ML101660701	William States Lee III - AP1000 Combined License Application Concurrence with Standard Content CI Number 04.04-1 LTR#WLG2010.06-04.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
6/11/2010	ML101660738	Submittal of Ddraft Pond C RAIs.	E-Mail	NRC/NRO/DSER/RAP3	Duke Energy Carolinas, LLC	05200018 05200019

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6/16/2010	ML101720639	E-mail Comment from Kendall Hale, Regarding Lee Nuclear Station, Supplemental Scoping Process.	E-Mail	- No Known Affiliation	NRC/NRO/DSER/RAP3	05200018 05200019
6/16/2010	ML101740331	Comment from Steve Moss, South Carolina House of Representative, Submitted at the William States Lee Nuclear Station Pond C Supplemental Scoping Public Meeting.	Letter	State of SC, House of Representative	NRC/ADM/DAS/RDEB	05200018 05200019
6/16/2010	ML101740333	Comment from Dennis Moss, South Carolina House of Representatives, Submitted at the William States Lee Nuclear Station Pond C Supplemental Scoping Public Meeting.	Letter	State of SC, House of Representative	NRC/ADM/DAS/RDEB	05200018 05200019
6/17/2010	ML101660730	06/17/2010 William States Lee III Nuclear Station, Units 1 and 2, Supplemental Scoping Meeting on Pond C Meeting Slides.	Meeting Briefing Package/Handouts Slides and Viewgraphs	NRC/NRO/DSER/RAP3 US Dept of the Army, Corps of Engineers		05200018 05200019
6/17/2010	ML101740332	Comment from Lanny F. Littlejohn, South Carolina House of Representatives, Submitted at the William States Lee Nuclear Station Pond C Supplemental Scoping Public Meeting.	Letter	State of SC, House of Representative	NRC/ADM/DAS/RDEB	05200018 05200019

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6/17/2010	ML101740334	Comment from Anne Craig, Submitted at the William States Lee Nuclear Station Pond C Supplemental Scoping Public Meeting.	Note	- No Known Affiliation	NRC/NRO	05200018 05200019
6/17/2010	ML101740335	Comment from Rachael Bliss, Submitted at the William States Lee Nuclear Station Pond C Supplemental Scoping Public Meeting.	- No Document Type Applies	- No Known Affiliation	NRC/NRO	05200018 05200019
6/17/2010	ML101740336	Comment from Mandy Hancock, High Risk Energy Organizer Southern Alliance for Clean Energy, Submitted at the William States Lee Nuclear Station Pond C Supplemental Scoping. Public Meeting.	- No Document Type Applies	Southern Alliance for Clean Energy	NRC/NRO	05200018 05200019
6/17/2010	ML101740337	Comment from Robert F. Howarth, Submitted at the William States Lee Nuclear Station Pond C Supplemental Scoping Public Meeting.	- No Document Type Applies	- No Known Affiliation	NRC/NRR	05200018 05200019
6/17/2010	ML101740338	Comment from Deb Arnason, Submitted at the William States Lee Nuclear Station Pond C Supplemental Scoping Public Meeting.	- No Document Type Applies	- No Known Affiliation	NRC/NRO/DSER	05200018 05200019

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6/17/2010	ML101740339	Comment from Ellen Thomas, Submitted at the William States Lee Nuclear Station Pond C Supplemental Scoping Public Meeting.	- No Document Type Applies	- No Known Affiliation	NRC/NRO	05200018 05200019
6/17/2010	ML101740340	Comment from Mary Olson on Behalf of Nuclear Information & Resource Service Southeast, Submitted at the William States Lee Nuclear Station Pond C Supplemental Scoping Public Meeting.	- No Document Type Applies	Nuclear Information & Resource Service (NIRS)	NRC/NRO	05200018 05200019
6/17/2010	ML101740341	Comment from Dr. Don Richardson on Behalf of Himself Western North Carolina Physicians for Social Responsibility, Submitted at the William States Lee Nuclear Station Pond C Supplemental Scoping Public Meeting.	- No Document Type Applies	Western North Carolina Physicians for Social Responsibility	NRC/NRO	05200018 05200019
6/17/2010	ML101740342	Comment from Valerie LeVander on Behalf of Global Warming Task Force of Henderson County, Submitted at the William States Lee Nuclear Station Pond C Supplemental Scoping Public Meeting.	- No Document Type Applies	Global Warming Task Force of Henderson County, NC	NRC/NRO	05200018 05200019

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6/17/2010	ML101740343	Comment from Katie Hicks on Behalf of Clean Water for North Carolina, Submitted at the William States Lee Nuclear Station Pond C Supplemental Scoping Public Meeting.	- No Document Type Applies	Clean Water for North Carolina	NRC/NRO	05200018 05200019
6/17/2010	ML101740344	Comment from Harvey S. Peeler, Jr., South Carolina Senator, Submitted at the William States Lee Nuclear Station Pond C Supplemental Scoping Public Meeting.	Letter	State of SC, Senate	NRC/ADM/DAS/RDEB	05200018 05200019
6/17/2010	ML101760446	Transcript of William States Lee III Nuclear Station, Units 1 and 2, Supplemental Scoping Public Meeting, June 17, 2010, Gaffney, SC, Pages 1-139.	Transcript	NRC/NRO/DSER/RAP3		05200018 05200019
6/17/2010	ML102030057	Comment (6) of Judith Hallock on William States Lee III, COL Application Unsatisfied due to Severe Drought and Associate Water Evaporation to Service the Proposed Reactor.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
6/18/2010	ML101720649	E-mail Comment from Pat McCall, Regarding Lee Nuclear Station, Supplemental Scoping Period.	E-Mail	- No Known Affiliation	NRC/NRO	05200018 05200019

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6/21/2010	ML101720644	E-mail comment from K. N. Mominee, Regarding Lee Nuclear Station, Supplemental Scoping Process.	E-Mail	- No Known Affiliation	NRC/NRO	05200018 05200019
6/21/2010	ML101720651	E-mail re: Comment from Caroline Dover Wilson, South Carolina State Historic Preservation Office, Regarding Lee Nuclear Station, Supplemental Scoping for Make-Up Pond C, Cherokee County, SC.	E-Mail	State of SC, Dept of Archives and History	NRC/NRO	05200018 05200019
6/22/2010	ML101370415	Pond C RAIs Cover Letter.	Letter	NRC/NRO/DSER/RAP3	Duke Power Co	05200018 05200019
6/22/2010	ML101370419	Pond C RAIs.	Request for Additional Information (RAI)	NRC/NRO/DSER/RAP3	Duke Power Co	05200018 05200019
6/22/2010	ML101750036	Comment (1) of Harvey S. Peeler on Behalf of South Carolina Senate Supporting the Proposed Lee Nuclear Station Make-Up Pond C.	General FR Notice Comment Letter Letter	State of SC, Senate	NRC/ADM/DAS/RDEB	05200018 05200019
6/23/2010	ML101740613	E-mail from Michael Mixon, Supplemental Scoping Comment on William States Lee Nuclear Station.	E-Mail	Shaw Group, Inc	NRC/NRO	05200018 05200019
6/23/2010	ML101740616	E-mail Comment from John Cross Regarding the William States Lee Nuclear Station, Supplemental Scoping.	E-Mail	URS Corp	NRC/NRO	05200018 05200019

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6/23/2010	ML101740618	E-mail Comment from Deborah Thrift, Regarding Lee Nuclear Station, Supplemental Scoping.	E-Mail	Shaw Power Group	NRC/NRO	05200018 05200019
6/23/2010	ML101750766	Judy & Glenn Ledford's Support for Gaffney Nuclear Plant.	E-Mail	- No Known Affiliation	NRC/NRO	05200018 05200019
6/23/2010	ML101800213	William States Lee III, Units 1 and 2 - AP1000 Combined License Application - Response to Request for Additional Information on Environmental Review Ltr# WLG2010.06-05.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
6/24/2010	ML101750764	E-mail from Barbara Barnett Regarding Lee Nuclear Station, Supplemental Scoping.	E-Mail	Four Seasons Sierra Committee of Henderson County, NC League of Women Voters of Henderson County, NC	NRC/NRO/DSER/RAP3	05200018 05200019
6/24/2010	ML101750767	E-mail from Brian Smith Regarding Lee Nuclear Station, Supplemental Scoping.	E-Mail	- No Known Affiliation	NRC/NRO	05200018 05200019
6/24/2010	ML101760352	E-mail from JW Drake, Lee Nuclear Station Supplemental Scoping Comment.	E-Mail	- No Known Affiliation	NRC/NRO	05200018 05200019

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6/25/2010	ML101810147	William States Lee III, Units 1 and 2, Combined License Application, Response to Request for Additional Information Ltr# WLG2010.06.06.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
6/25/2010	ML13247A044	Chem-Nuclear Systems. 2010. 2010 Interim Site Stabilization and Closure Plan for the Barnwell Disposal Facility.	Report, Technical	Chem-Nuclear Systems, LLC EnergySolutions, LLC	State of SC, Dept of Health & Environmental Control NRC/NRO	05200018 05200019
6/28/2010	ML101810248	Comment (2) of Bill Thomas, on Behalf of the Pisgah Group of NC Chapter of Sierra Club,on NRC-2008-0170-0002, William States Lee Combined License Application; Notice of Intent to Conduct a Supplemental Scoping Process for Supplement to Environmental...	General FR Notice Comment Letter	Sierra Club, North Carolina Chapter Sierra Club, Pisgah Group	NRC/ADM/DAS/RDEB	05200018 05200019 05200022 05200023
6/30/2010	ML101900426	Comment (5) of Sara Barczak & Mandy Hancock on behalf of Southern Alliance for Clean Energy Opposing Supplement 1 to Revision 1 of the Enviromnental Report of Duke Energy's W.H. Lee Combined Operating License Application.	General FR Notice Comment Letter	Southern Alliance for Clean Energy	NRC/ADM/DAS/RDEB	05200018 05200019

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7/1/2010	ML101820355	E-mail re: Comment from M. Hancock, Southern Alliance for Clean Energy, Regarding William States Lee Supplemental Scoping, Addition of Pond C.	E-Mail	Southern Alliance for Clean Energy	NRC/NRO	05200018 05200019
7/1/2010	ML101820646	E-mail Comment from Ben Gregg, SC Wildlife Federation, Regarding Lee Nuclear Station, Supplemental Scoping.	E-Mail	South Carolina Wildlife Federation	NRC/NRO	05200018 05200019
7/1/2010	ML101880072	AP1000 Combined License Application for the William States Lee III Units 1 & 2 Response to Request for Additional Information, Ltr# WLG2010.07-01.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
7/1/2010	ML101890471	Oconee, Units 1, 2 & 3, & ISFSI, McGuire, Units 1 & 2, Catawba, Units 1 and 2, and William States Lee, Units 1 and 2, Change in Legal Counsel.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NMSS NRC/NRR	05000269 05000270 05000287 05000369 05000370 05000413 05000414 05200018 05200019

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						07200004
7/1/2010	ML102290307	Comment from S. Breckheimer Regarding Response from Comment on NRC Documents.	E-Mail	- No Known Affiliation	NRC/NRO/DSER	05200018 05200019
7/1/2010	ML112800491	SCDHEC 2010 Network.	Report, Miscellaneous	State of SC, Dept of Health & Environmental Control	NRC/NRO	05200018 05200019
7/2/2010	ML101800423	Summary of Supplemental Environmental Scoping Meeting Conducted Related to Combined License Application Review of William States Lee III, Units 1 & 2.	Meeting Summary Memoranda	NRC/NRO/DSER/RAP3	NRC/NRO/DSER/RAP3	05200018 05200019
7/2/2010	ML102030058	Comment (7) of Lee Pennington on William States III COL Application.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019

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7/6/2010	ML101890550	Comment (4) of Ben Gregg on Behalf of South Carolina Wildlife Federation on Duke Energy Lee Nuclear Station, Make-Up Pond C, Ref: William States Lee III Nuclear Station, Units 1 & 2, COLA.	General FR Notice Comment Letter	South Carolina Wildlife Federation	NRC/ADM/DAS/RDEB	05200018 05200019
7/7/2010	ML101870564	June 15, 2010 Summary of Teleconference with Duke Energy Carolinas, LLC., Concerning Requests for Additional Information Regarding Make-Up Pond C for William States Lee III Nuclear Station, Units 1 & 2.	Conference/Symposium/Workshop Paper Meeting Summary Memoranda	NRC/NRO/DSER/RAP3	NRC/NRO/DSER/RAP3	05200018 05200019
7/7/2010	ML102070103	Comment (8) of Clyde E. (Butch) Smith on Behalf of Cleveland County Water Supporting on Proposed Make-Up Pond "C" for William States Lee III in Gaffney South Carolina.	General FR Notice Comment Letter	Cleveland County, NC	NRC/ADM/DAS/RDEB	05200018 05200019
7/9/2010	ML101940026	AP1000 Combined License Application for the William States Lee III Nuclear Station Units 1 and 2, Conforming Changes to Environmental Report Based on Supplemental Response to Request for Additional Information Ltr # WLG2010.07-04.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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7/9/2010	ML101950207	William States Lee III, Units 1 and 2 - AP1000 Combined License Application - Response to Request for Additional Information, Ltr# WLG2010.07-03.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
7/9/2010	ML101950209	Map - Proposed Williams States Lee III Nuclear Cooler Water Make Ponds.	Map	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
7/16/2010	ML102020479	William States Lee III, Units 1 & 2 AP1000 Combined License Application, Submittal of Response to Request for Additional Information Ltr# WLG2010.07-07.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
7/16/2010	ML102100214	William States Lee III, Units 1 and 2 - AP1000 Combined License Application, Response to Request for Additional Information.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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7/19/2010	ML102000359	08/12/10 - Meeting Notice for Design-Centered Working Group (DCWG), Re: Discuss Guidance Associated with Complying with 10 CFR 52.79(a)(31) Regarding Construction Impacts.	Meeting Agenda Meeting Summary	NRC/NRO/DNRL/NWE1	NRC/NRO/DNRL/NWE1	05200014 05200015 05200018 05200019 05200022 05200023 05200025 05200026 05200027 05200028 05200029 05200030 05200040 05200041

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7/19/2010	ML102210385	08/26/2010 - Revised Meeting Notice - Notice of Forthcoming Public Meeting with the AP1000 Design-Centered Working Group to Discuss Guidance Associated with Complying with 10 CFR 52.79(a)(31) Regarding Construction Impacts.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/NWE1	NRC/NRO/DNRL/NWE1	05200014 05200015 05200018 05200019 05200022 05200023 05200025 05200026 05200027 05200028 05200029 05200030 05200040 05200041
7/21/2010	ML102040037	William States Lee III, Units 1 & 2, AP1000 Combined License Application, Editorial Corrections to Revision 2, Ltr. #WLG2010-07-05.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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7/22/2010	ML102070357	William States Lee III Nuclear Station, Units 1 and 2 - Combined License Application for the Response to Request for Additional Information on Environmental Report, Ltr# WLG2010.07-08.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
7/22/2010	ML102090223	William States Lee III, Units 1 and 2, AP1000 Combined License Application, Response to Request for Additional Information, RAI 137 and RAI 140, Cultural Resources.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
7/22/2010	ML102110494	Letter from W.G. Haire, Catawba Indian Nation to S. Flanders, NRC, Regarding William States Lee Nuclear Station COL Application.	Letter	Catawba Indian Nation	NRC/NRO/DSER	05200018 05200019
7/27/2010	ML102160393	Comment (9) of Vivianne Vejdani on Behalf of South Carolina Dept. of Natural Resources on William States Lee III Combined License Application, Notice of Intent to Conduct a Supplemental Scoping Process for Supplement to the Environmental Report.	General FR Notice Comment Letter	State of SC, Dept of Natural Resources	NRC/ADM/DAS/RDEB	05200018 05200019
7/30/2010	ML102110501	E-mail to R. Wiley, Duke Energy, Regarding Lee Pond C Site Audit Information Needs.	E-Mail	NRC/NRO/DSER/RAP3	Duke Energy Corp	05200018 05200019

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
8/3/2010	ML112760816	USGS 2010 Ann Water.	Environmental Report Graphics incl Charts and Tables	US Dept of Interior, Geological Survey (USGS)	NRC/NRO	05200018 05200019
8/4/2010	ML102180175	William States Lee III, Units 1 and 2 Combined License (COL) Application, Submittal of 30-Day Report in Accordance with 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors."	Letter Licensee 30-Day Written Event Report	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
8/10/2010	ML102220109	08/12/2010 - Notice of Cancellation of Forthcoming Public Meeting with AP1000 Design-Centered Working Group to Discuss Guidance Associated with Complying with 10 CFR 52.79(a)(31) Regarding Construction Impacts.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/NWE1	NRC/NRO/DNRL/NWE1	05200014 05200015 05200018 05200019 05200022 05200023 05200025 05200026 05200027 05200028

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
						05200029 05200030 05200040 05200041
8/10/2010	ML102220269	08/26/2010 Notice of Forthcoming Meeting With the AP1000 Design-Centered Working Group to Discuss Guidance Associated With Complying With 10 CFR 52.79(a)(31) Regarding Construction Impacts.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/NWE1	NRC/NRO/DNRL/NWE1	05200014 05200015 05200018 05200019 05200022 05200023 05200025 05200026 05200027

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						05200028 05200029 05200030 05200040 05200041
8/12/2010	ML102240279	2010/08/12 Lee RAI for SER - RAI-LTR-090 RELATED TO SRP SECTION 2.3.5 FOR THE W.S. LEE UNITS 1 AND 2 COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
8/16/2010	ML102290314	Electronic Comment submitted by E. Thomas Regarding Make-Up Pond C, William States Lee Nuclear Station, Supplemental Scoping.	E-Mail	Proposition One Committee	NRC/NRO	05200018 05200019
8/24/2010	ML102360006	2010/08/24 Lee RAI for SER - LEE-RAI-LTR-091 RELATED TO SRP 2.3.4,2.3.3, 2.3.5 DISPERSION ESTIMATES FOR ACCIDENTS FOR	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019

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		THE W S LEE UNITS 1 AND 2 COLA				
8/24/2010	ML102360015	2010/08/24 Lee RAI for SER - LEE-RAI-LTR-091 RELATED TO SRP 2.3.4,2.3.3, 2.3.5 DISPERSION ESTIMATES FOR ACCIDENTS FOR THE W S LEE UNITS 1 AND 2 COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
8/24/2010	ML102380039	William States Lee III, Units 1 and 2, AP1000 Combined License Application, Departure Report Update.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
8/24/2010	ML102380042	William States Lee III, Units 1 and 2 Combined License Application, Response to Request for Additional Information No. 2350.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
8/25/2010	ML102371060	2010/08/25 Lee RAI for SER - LEE-RAI-LTR-091 RELATED TO SRP 02.04.12 GROUND WATER FOR THE W.S. LEE UNITS 1 AND 2 COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019

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8/25/2010	ML112710653	SC Rare, Threatened & Endangered Species Inventory, SCDNR 2010.	- No Document Type Applies	State of SC, Dept of Natural Resources	NRC/NRO	05200018 05200019
8/25/2010	ML112800429	USGS 2010 Browns Sand Dredge.	Report, Miscellaneous	US Dept of Interior, Geological Survey (USGS)	NRC/NRO	05200018 05200019
8/25/2010	ML112800431	USGS 2010 Red Clay-Higgins.	Report, Miscellaneous	US Dept of Interior, Geological Survey (USGS)	NRC/NRO	05200018 05200019
8/25/2010	ML112800433	USGS 2010 Thomas Sand Co - Blacksburg Plant.	Report, Miscellaneous	US Dept of Interior, Geological Survey (USGS)	NRC/NRO	05200018 05200019
9/1/2010	ML102440857	E-mail to R. Wiley, Duke Energy Carolinas, Transmitting Draft Make-Up Pond C Follow-Up RAIs.	E-Mail Request for Additional Information (RAI)	NRC/NRO/DSER/RAP3	Duke Energy Carolinas, LLC	05200018 05200019
9/1/2010	ML102980279	William States Lee III, Units 1 and 2, Integrated Resource Plan (Annual Report).	Annual Report	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019

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9/7/2010	ML102440473	08/26/2010 Summary of Public Meeting With AP1000 Design-Centered Working Group to Discuss Guidance Associated With Complying With 10 CFR 52.79(a)(31) Regarding Construction Impacts.	Memoranda	NRC/NRO/DNRL/NWE1	NRC/NRO/DNRL/NWE1	05200014 05200015 05200018 05200019 05200022 05200023 05200025 05200026 05200027 05200028 05200029 05200030 05200040 05200041

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9/7/2010	ML102450020	Slides - Summary of the August 26, 2010, Public Meeting With AP1000 Design-Centered Working Group to Discuss Guidance Associated With Complying With 10 CFR 52.79(a)(31) Regarding Construction Impacts.	Meeting Summary Slides and Viewgraphs	NRC/NRO/DNRL/NRGA		05200014 05200015 05200018 05200019 05200022 05200023 05200025 05200026 05200027 05200028 05200029 05200030 05200040 05200041
9/7/2010	ML102530391	William States Lee III, Units 1 & 2, AP1000 Combined License Application, Information Omitted from Response to Environmental Report RAI 192.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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9/14/2010	ML102371173	William Lee III Nuclear Station, Units 1 and 2, Follow-Up Requests for Additional Information (RAIs) Regarding Make-Up Pond C (Cover Letter).	Letter	NRC/NRO/DSER/RAP3	Duke Energy Carolinas, LLC	05200018 05200019
9/14/2010	ML102371189	William States Lee III, Units 1 and 2, Make-up Pond C Audit Follow-up RAIs.	Request for Additional Information (RAI)	NRC/NRO/DSER/RAP3	Duke Energy Carolinas, LLC	05200018 05200019
9/15/2010	ML102600559	William States Lee III Nuclear Station Units 1 and 2, AP1000 Combined License Application for the Supplemental Response to RAI No. 02.04.13-013, Ltr # WLG2010.09-04.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
9/16/2010	ML102640040	William States Lee III Nuclear Station, Units 1 & 2 - AP1000 Combined License Application for the Response to Request for Additional Information (RAI 4959 and 4960) Ltr# WLG2010.09-02.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
9/21/2010	ML102730224	William States Lee III AP1000 Combined License Application, Supplemental Response to RAI No. 02.04.13-013, Ltr #WLG2010.09-06.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
9/27/2010	ML112860659	EPA Westpoint Stevens Clemson - Envirofacts Report.	Environmental Report	US Environmental Protection Agency (EPA)	NRC/NRO/DSER/RAP3	05200018 05200019

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
9/28/2010	ML102740218	William States Lee III Nuclear Station, Units 1 & 2 - Combined License Application for the Response to Request for Additional Information (RAI No. 4961) Ltr# WLG2010.09-05.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
9/28/2010	ML102740485	William States Lee III, Units 1 and 2 - AP1000 Combined License Application, Response to Request for Additional Information (ER RAI 207, ER RAI 208, ER RAI 212, ER RAI 214, ER RAI 215, and ER RAI 219).	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
9/30/2010	ML102770372	William States Lee III, Units 1 and 2, AP1000 Combined License Application, Response to Request for Additional Information No. 4870.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
9/30/2010	ML102780268	William States Lee III, Units 1 and 2 AP1000 Combined License Application, Response to Request for Additional Information on Supplement to Environmental Report.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
10/6/2010	ML102810637	William States Lee III - AP1000 Combined License Application Response to Request for Additional Information, Ltr. #WLG2010.10-01.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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10/6/2010	ML102850208	William States Lee III, Units 1 & 2 - AP1000 Combined License Application, Responses to Request for Additional Information 209, Ecology - Aquatic, 213; Ecology - Terrestrial and 220, Cultural Resources.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
10/12/2010	ML112760701	EPA 2010 Carolina Quarries.	Database File	US Environmental Protection Agency (EPA)	NRC/NRO	05200018 05200019
10/12/2010	ML112800414	EPA 2010 Vulcan Materials.	Report, Miscellaneous	US Environmental Protection Agency (EPA)	NRC/NRO	05200018 05200019
10/12/2010	ML112800517	EPA 2010 Carroll, Water Discharge Permits.	- No Document Type Applies	US Environmental Protection Agency (EPA)	NRC/NRO	05200018 05200019
10/12/2010	ML112800519	EPA 2010 Industrial Minerals Webpage.	FACT Sheet	US Environmental Protection Agency (EPA)	NRC/NRO	05200018 05200019
10/14/2010	ML102920172	William States Lee III, Units 1 and 2, AP1000 Combined License Application, Response to Request for Additional Information Ltr#WLG2010.10-05.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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10/14/2010	ML102940206	William States Lee III, Units 1 and 2, AP1000 Combined License Application for the Hydrological Input/Output Files for HMR52, HEC-RAS, and HEC-HMS Ltr# WLG2010.10-03.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
10/14/2010	ML102980199	William States Lee III, Units 1 and 2, 2010 Integrated Resource Plan.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
10/14/2010	ML103360418	Enclosure 1, Lee Nuclear Station Response to Request for Additional Information, RAI Letter Dated: September 14, 2010, ER-RAI 206, Alternatives.	Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
10/14/2010	ML103360419	William States Lee III Nuclear Station - AP1000 Combined License Application, Response to Request for Additional Information Ltr# WLG2010.10-04.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
10/14/2010	ML103360420	Appendix A: Study Plan/Scope.	- No Document Type Applies	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
10/14/2010	ML112800416	EPA 2010 Green Book 1 Hour.	Report, Miscellaneous	US Environmental Protection Agency (EPA)	NRC/NRO	05200018 05200019

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10/20/2010	ML112800529	NRC 2011 VC Summer.	- No Document Type Applies	NRC/NRO/DSER/RAP3	NRC/NRO	05000395 05200018 05200019
10/25/2010	ML112700862	Big Creek East WWTP, Recipent Project Summary 2010,	Database File	US Recovery Accountability & Transparency Board	NRC/NRO	05200018 05200019
10/25/2010	ML112760710	USFWS 2010 Species.	Database File	US Dept of Interior, Fish & Wildlife Service	NRC/NRO	05200018 05200019
10/25/2010	ML112800281	FWS 2010 York.	- No Document Type Applies	US Dept of Interior, Fish & Wildlife Service	NRC/NRO	05200018 05200019
10/26/2010	ML112760696	EPA 2010 Arteva.	Graphics incl Charts and Tables Database File FACT Sheet	US Environmental Protection Agency (EPA)	NRC/NRO	05200018 05200019
10/26/2010	ML112760707	2010 Carolina Sand.	Database File	US Environmental Protection Agency (EPA)	NRC/NRO	05200018 05200019

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10/29/2010	ML103070311	William States Lee III, Units 1 and 2 - AP1000 Combined License Application, Response to Request for Additional Information RAI 128 and RAI 216.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
10/29/2010	ML112650443	SCBCB 2006 2000-2035 Population.	Report, Miscellaneous Graphics incl Charts and Tables	US Dept of Commerce, Bureau of Census	NRC/NRO/DSER/RAP3	05200018 05200019
10/29/2010	ML112710648	Bureau of Labor Statistic 2011.	Report, Miscellaneous	US Dept of Labor, Bureau of Labor Statistics	NRC/NRO	05200018 05200019
10/29/2010	ML112800219	USCB 2009 NC City Populations.	Report, Miscellaneous	US Dept of Commerce, Bureau of Census	NRC/NRO	05200018 05200019
11/1/2010	ML113260096	GDNR 2010a Title V.	Database File	State of GA, Environmental Protection Div	NRC/NRO	05200018 05200019
11/1/2010	ML113260100	GDNR 2010b.	Database File	State of GA, Environmental Protection Div	NRC/NRO	05200018 05200019
11/3/2010	ML112800521	EPA 2010 Industrial Minerals 2.	FACT Sheet	US Environmental Protection Agency (EPA)	NRC/NRO	05200018 05200019

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11/4/2010	ML103070537	11/17/2010 Notice of Forthcoming Public Teleconference With Duke Energy Carolinas, LLC, to Discuss Request for Additional Information Response for the William States Lee III Nuclear Station, Units 1 & 2 Combined License Application Environmental Review.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DSER/RAP3	NRC/NRO/DSER/RAP3	05200018 05200019
11/4/2010	ML103130134	William States Lee III, Units 1 & 2 - AP1000 Combined License Application, Summary Identification of Concurrence with Standard Content RAIs and Safety Evaluation Report Open Items Ltr# WLG2010.11-01.	Legal-Affidavit Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
11/4/2010	ML112710640	North Carolina Department of Environment and Natural Resoures 2010.	Database File	State of NC, Dept of Environment & Natural Resources, Div of Water Quality	NRC/NRO	05200018 05200019
11/4/2010	ML112760682	NCDOT 2010 Beltway.	Database File	State of NC, Dept of Transportation	NRC/NRO	05200018 05200019
11/4/2010	ML112760686	NCDPR 2010 Map.	Map	State of NC, Div of Parks & Recreation	NRC/NRO	05200018 05200019

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11/5/2010	ML112710636	City of Winston-Salem Landfill Winston-Salem, NC - 4350 kw 2010.	- No Document Type Applies	Landfill Engineering Solutions, Inc	NRC/NRO	05200018 05200019
11/10/2010	ML103160424	Oconee, McGuire, Catawba, Oconee, Independent Spent Fuel Storage Installation and William States Lee, III, Notification of Service/Distribution Listing Update.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NMSS NRC/NRR	05000269 05000270 05000287 05000369 05000370 05000413 05000414 05200018 05200019 07200004
11/10/2010	ML112800402	EPA 2010 Major Air Sources.	Map	US Environmental Protection Agency (EPA)	NRC/NRO	05200018 05200019
11/12/2010	ML103210413	William States Lee III, Units 1 and 2 AP1000 Combined License Application, Response to Request for Additional Information 210	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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		Supplement, Ecology - Aquatic.				
11/17/2010	ML103260471	E-mail to R. Wylle Regarding Lee Nuclear Station Public Teleconference on Hybrid Cooling RAI.	E-Mail	NRC/NRO/DSER/RAP3	Duke Energy Corp	05200018 05200019
11/18/2010	ML103120606	Inspection of Duke Energy Carolinas, LLC Quality Assurance Program Implementation for William States Lee III Nuclear Station Units 1 and 2 Combined License Application.	Letter	NRC/NRO/DCIP/CQVP	Duke Energy Carolinas, LLC	05200018 05200019
11/19/2010	ML103150012	Response Letter to Sandra Threatt Regarding Environmental Monitoring Around the Proposed William States Lee III Nuclear Station (WSL) Units 1 and 2, Located in Cherokee County, South Carolina.	Letter	NRC/NRO/DNRL/NWE1	State of SC, Dept of Health & Environmental Control	05200018 05200019
12/15/2010	ML103510030	William States Lee III Nuclear Station Units 1 and 2 - Response to NRC Regulatory Issue Summary (RIS) 2010-10.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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12/17/2010	ML103550032	William States Lee, Units 1 and 2, AP1000 Combined License Application, Responses to Request for Additional Information Ltr# WLG2010.12-01.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
12/17/2010	ML110030205	William States Lee III, Units 1 and 2, AP1000 Combined License Application, Update License Application.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
12/17/2010	ML110030639	Duke Energy WSL III Units 1 & 2 COLA (General and Financial Information), Rev. 3 - Part 1, General and Financial Information	License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
12/17/2010	ML110030641	Duke Energy WSL III Units 1 & 2 COLA (ITAAC), Rev. 3 - Part 10, Conditions and ITAAC	Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
12/17/2010	ML110030643	Duke Energy WSL III Units 1 & 2 COLA (Technical Specifications), Rev. 3 - Part 4, Williams Lee III Nuclear Station Technical Specifications	License-Application for Combined License (COLA) Technical Specifications	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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12/17/2010	ML110030205	William States Lee III, Units 1 and 2, AP1000 Combined License Application, Update License Application.	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
12/17/2010	ML110030243	Duke Energy WSL III Units 1 & 2 COLA (Generic DCD Departures Report), Rev. 3 - Part 7, Departures and Exemptions Requests	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
12/17/2010	ML110030249	Duke Energy WSL III Units 1 & 2 COLA (Quality Assurance Program), Rev. 3	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
12/17/2010	ML110030248	Duke Energy WSL III Units 1 & 2 COLA (Quality Assurance Program), Rev. 3 - Part 11, Nuclear Plant Development Quality Assurance Program Description	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
12/17/2010	ML110030638	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report), Rev. 3	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
12/17/2010	ML110030639	Duke Energy WSL III Units 1 & 2 COLA (General and Financial Information), Rev. 3 - Part 1, General and Financial Information	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
12/17/2010	ML110030641	Duke Energy WSL III Units 1 & 2 COLA (ITAAC), Rev. 3 - Part 10, Conditions and ITAAC	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019

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12/17/2010	ML110030643	Duke Energy WSL III Units 1 & 2 COLA (Technical Specifications), Rev. 3 - Part 4, Williams Lee III Nuclear Station Technical Specifications	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
12/21/2010	ML102640559	William States Lee Nuclear Station Make-Up Pond C and Alternative Sites Tour Audit Summary.	Memoranda	NRC/NRO/DSER/RAP3	NRC/NRO/DSER/RAP3	05200018 05200019
12/22/2010	ML103220015	Supplemental Scoping Summary Report Regarding Make-Up Pond C for the William States Lee III Nuclear Station, Units 1 and 2, Environmental Review.	Memoranda Environmental Report	NRC/NRO/DSER/RAP3	NRC/NRO/DSER/RAP3	05200018 05200019
12/27/2010	ML112800444	EPA 2010 Accurate Plating Inc.	Report, Miscellaneous	US Environmental Protection Agency (EPA)	NRC/NRO	05200018 05200019
12/27/2010	ML112800446	EPA 2010 BIC Corporation.	Report, Miscellaneous	US Environmental Protection Agency (EPA)	NRC/NRO	05200018 05200019
12/27/2010	ML112800451	EPA 2010 Hanson Brick Blacksburg.	Report, Miscellaneous	US Environmental Protection Agency (EPA)	NRC/NRO	05200018 05200019

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12/27/2010	ML112800466	EPA 2010 Linpac.	Report, Miscellaneous	US Environmental Protection Agency (EPA)	NRC/NRO	05200018 05200019
12/27/2010	ML112800498	EPA 2010 Milliken Magnolia.	Report, Miscellaneous	US Environmental Protection Agency (EPA)	NRC/NRO	05200018 05200019
12/27/2010	ML112800501	EPA 2010 Bommer Industries, Facility Detail Report.	Report, Miscellaneous	US Environmental Protection Agency (EPA)	NRC/NRO	05200018 05200019
12/27/2010	ML112800505	EPA 2010 Broad River Energy Center.	Report, Miscellaneous	US Environmental Protection Agency, Office of Enforcement and Compliance Assurance	NRC/NRO	05200018 05200019
12/27/2010	ML112800506	EPA 2010 Grover Compressor Station.	Report, Miscellaneous	US Environmental Protection Agency (EPA)	NRC/NRO	05200018 05200019

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12/27/2010	ML112800510	EPA 2010 Cherokee County Co-Generation Plant.	Report, Miscellaneous	US Environmental Protection Agency, Office of Enforcement and Compliance Assurance	NRC/NRO	05200018 05200019
12/27/2010	ML112800514	EPA 2010 SC Pipeline.	Report, Miscellaneous	US Environmental Protection Agency (EPA)	NRC/NRO	05200018 05200019
12/28/2010	ML112800447	EPA 2010 CNA Holdings.	Report, Miscellaneous	US Environmental Protection Agency (EPA)	NRC/NRO	05200018 05200019
12/28/2010	ML112800500	EPA 2010 National Textiles.	Report, Miscellaneous	US Environmental Protection Agency (EPA)	NRC/NRO	05200018 05200019
12/28/2010	ML112800512	EPA 2010 Gaffney WWTF Peoples Creek.	Report, Miscellaneous	US Environmental Protection Agency, Office of Enforcement and Compliance Assurance	NRC/NRO	05200018 05200019

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12/28/2010	ML112800513	EPA 2010 SC Distributors.	Report, Miscellaneous	US Environmental Protection Agency (EPA)	NRC/NRO	05200018 05200019
12/28/2010	ML112800515	EPA 2010 Shelby WWTF.	Report, Miscellaneous	US Environmental Protection Agency (EPA)	NRC/NRO	05200018 05200019
12/28/2010	ML112800516	EPA 2010 Parr Steam Plant.	Map Report, Miscellaneous	US Environmental Protection Agency (EPA)	NRC/NRO	05200018 05200019
12/29/2010	ML112800504	EPA 2010 Cleveland Water Supply, Facility Detail Report.	Report, Miscellaneous	US Environmental Protection Agency (EPA)	NRC/NRO	05200018 05200019
12/29/2010	ML112800525	NCDEH 2010 Cleveland, Source Water Assessment Program Report.	- No Document Type Applies	Cleveland County Water, NC	NRC/NRO	05200018 05200019
12/29/2010	ML112800528	NRC 2011 HB Robinson.	Report, Miscellaneous	NRC/NRO/DSER/RAP3	NRC/NRO	05000261 05200018 05200019
12/30/2010	ML112800450	EPA 2010 Core Molding.	Report, Miscellaneous	US Environmental Protection Agency (EPA)	NRC/NRO	05200018 05200019

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1/1/2011	ML112710647	RIMS II Multipliers (2002/2007), Table 1.5 Total Multipliers for Output, Earnings, Employment, and Value Added by Detailed Industry Cherokee and York Counties, SC (Type II).	Graphics incl Charts and Tables	- No Known Affiliation	NRC/NRO	05200018 05200019
1/1/2011	ML112800285	USFWS 2011 Heartleaf.	Environmental Monitoring Report	US Dept of Interior, Fish & Wildlife Service	NRC/NRO	05200018 05200019
1/1/2011	ML112800311	USFWS 2011 Coneflower.	Environmental Monitoring Report	US Dept of Interior, Fish & Wildlife Service	NRC/NRO	05200018 05200019
1/6/2011	ML110110399	William States Lee III Nuclear Station, Units 1 and 2 - License for Update Roadmap Ltr# WLG2011.01-01.	Letter Report, Miscellaneous	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
1/7/2011	ML110140903	E-mail from Scott Harder, SC Dept. of Natural Resources, Regarding Lee Nuclear Project - Alternatives Modeling Analysis (Hybrid Cooling).	E-Mail	State of SC, Dept of Natural Resources	Battelle Memorial Institute, Pacific Northwest National Lab NRC/NRO	05200018 05200019
1/11/2011	ML103370325	William States Lee III Nuclear Station, Units 1 and 2 Combined License Application - Revised Review Schedule.	Letter Graphics incl Charts and Tables	NRC/NRO/DNRL	Duke Energy Carolinas, LLC	05200018 05200019

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1/19/2011	ML12033A157	Comment (4) of Melissa Lemoing Opposing the Construction of the William States Lee Nuclear Power Plant.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
1/19/2011	ML12033A158	Comment (5) of Patricia Allison Opposing the Draft Environmental Impact Statement for the William States Lee Nuclear Station.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
1/19/2011	ML12033A160	Comment (7) of Michaeljon Drouin, on William States Lee III Nuclear Station, Units 1 & 2, DEIS Public Meeting.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
1/23/2011	ML112860662	EPA BASF	Report, Miscellaneous	US Environmental Protection Agency (EPA)	NRC/NRO	05200018 05200019
1/25/2011	ML103630488	11/17/10 Summary of Teleconference with Duke Energy Carolinas to discuss Duke's response to a request for additional information (RAI) regarding an analysis of hybrid wet-dry cooling as a potential alternative to the proposed Make-Up Pond C.	Memoranda Conference/Symposium/Workshop Paper Meeting Summary	NRC/NRO/DSER/RAP3	NRC/NRO/DSER/RAP3	05200018 05200019

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1/26/2011	ML110310017	William States Lee III, Units 1 and 2, AP1000 Combined License Application for the Responses to Request for Additional Information RAI 128 Supplement Alternatives.	Legal-Affidavit Letter	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
1/28/2011	ML112800425	EPA 2011 Cherokee County Mines.	Report, Miscellaneous	US Environmental Protection Agency (EPA)	NRC/NRO	05200018 05200019
2/2/2011	ML110330576	2011/02/02 Lee (Duke) COL Hearing - DRAFT RAI RELATED TO SRP SECTION 13.3 FOR THE WILLIAM STATES LEE III UNITS 1 AND 2 COMBINED LICENSE APPLICATION	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
2/3/2011	ML110140852	William States Lee III Nuclear Station, Units 1 and 2, Request for Additional Information Regarding Environmental Review.	Letter Request for Additional Information (RAI)	NRC/NRO/DSER/RAP3	Duke Energy Carolinas, LLC	05200018 05200019
2/9/2011	ML112800418	EPA 2011 Sole Source Aquifers in the Southeast.	Report, Miscellaneous	US Environmental Protection Agency (EPA)	NRC/NRO	05200018 05200019
2/10/2011	ML112700859	NCUC 2011 - E-100, Subs 128 and 129.	Letter	State of NC, Utilities Commission	NRC/NRO	05200018 05200019

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2/11/2011	ML112700854	NCDENR_2011.	Database File	State of NC, Dept of Environment & Natural Resources	NRC/NRO	05200018 05200019
3/5/2011	ML12067A095	Comment (30) of Daniel Schmitt on behalf of self, Opposing Duke Energy Carolina, LLC, Combined License Application, Intent to Prepare Environmental Impact Statement.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DC	05200018 05200019
3/10/2011	ML112700863	USCB 2010 American Fact Finder.	Database File	US Dept of Commerce, Bureau of Census	NRC/NRO	05200018 05200019
3/14/2011	ML103000023	Letter to Catawba Indian Nation on Cultural Resources Information Related to the William States Lee III Nuclear Station, Units 1 and 2, Environmental Review.	Letter	NRC/NRO/DSER/RAP3	Catawba Indian Nation	05200018 05200019
3/31/2011	ML112800305	USFWS 2011 Cougar.	Environmental Monitoring Report	US Dept of Interior, Fish & Wildlife Service	NRC/NRO	05200018 05200019
5/4/2011	ML11129A054	William States Lee III, Units 1 and 2 - AP1000 Combined License Application for the Supplemental Response to Request for Additional Information (ER RAI 23).	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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5/10/2011	ML111390647	05/24/2011 Revised Notice of Meeting With AP1000 Design-Centered Working Group (DCWG).	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/NWE1	NRC/NRO/DNRL/NWE1	05200014 05200015 05200018 05200019 05200022 05200023 05200025 05200026 05200027 05200028 05200029 05200030 05200040 05200041
5/12/2011	ML111320652	Petitioners' Motion for Modification of the Commission's April 19, 2011, Order to Permit a Consolidated Reply.	Legal-Motion	Blue Ridge Environmental Defense League	NRC/OCM	05200018 05200019

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5/12/2011	ML111320661	Petitioners' Reply to Responses to Emergency Petition to Suspend All Pending Reactor Licensing Decisions and Related Rulemaking Decisions Pending Investigation of Lessons Learned From Fukushima Daiichi Nuclear Power Station Accident.	Legal-Pleading	Blue Ridge Environmental Defense League	NRC/OCM	05200018 05200019
5/16/2011	ML111360327	NRC Staff's Answer to Petitioners' Motion for Modification of the Commission's April 19, 2011, Order to Permit A Consolidated Reply.	Legal-Pleading	NRC/OGC	NRC/OCM	05200018 05200019
5/16/2011	ML11140A030	Request to Deny Duke Energy the License to build and Operate a New Nuclear Plant in Cherokee, Carolina.	Letter	- No Known Affiliation	NRC/NRO	05200018 05200019
5/17/2011	ML112901424	Associated Press 2011 - Texas House OKs Taking in More Radioactive Waste.	News Article	Associated Press Bloomberg Businessweek	NRC/NRO	05200018 05200019
5/18/2011	ML11139A408	William States Lee III, Units 1 and 2 - AP1000 Combined License Application, Response to Request for Additional Information (RAI No. 5507) Ltr. # WLG2011.05-02.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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5/18/2011	ML11139A409	William States Lee III, Units 1 and 2, AP1000 Combined License Application - Concurrence with Standard Content Regarding In-Transit Requirements for New Fuel Shipments Ltr. # WLG2011.05-03.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
5/20/2011	ML111400413	William States Lee COL Environmental Review - E-mail to R. Wiley Transmitting Alternatives Audit Information Needs.	E-Mail Request for Additional Information (RAI)	NRC/NRO/DSER/RAP3	Duke Energy Carolinas, LLC	05200018 05200019
5/23/2011	ML111430644	Duke Energy's Answer Opposing Motion To Allow Unauthorized Reply.	Legal-Pleading	Duke Energy Corp Pillsbury, Winthrop, Shaw, Pittman, LLP	NRC/OCM	05200018 05200019

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5/24/2011	ML111460082	5/24/11 - AP1000 DCWG Meeting to Discuss Piping DAC and Initial Test Program License Conditions.	Slides and Viewgraphs Meeting Briefing Package/Handouts	NRC/NRO/DNRL/NWE1		05200014 05200015 05200018 05200019 05200022 05200023 05200025 05200026 05200027 05200028 05200029 05200030 05200040 05200041

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5/24/2011	ML111460084	Meeting Handouts for 5/24/11 - AP1000 DCWG Meeting to Discuss Piping DAC and Initial Test Program License Conditions.	Slides and Viewgraphs Meeting Briefing Package/Handouts	Southern Nuclear Operating Co, Inc Southern Co	NRC/NRO	05200014 05200015 05200018 05200019 05200022 05200023 05200025 05200026 05200027 05200028 05200029 05200030 05200040 05200041

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5/24/2011	ML111460093	AP1000 DCWG Meeting to Discuss Piping DAC and Initial Test Program License Conditions.	Meeting Briefing Package/Handouts	NRC/NRO/DNRL/NWE1		05200014 05200015 05200018 05200019 05200022 05200023 05200025 05200026 05200027 05200028 05200029 05200030 05200040 05200041

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
5/24/2011	ML111460096	Meeting Handouts for 5/24/11 - AP1000 DCWG Meeting to Discuss Piping DAC and Initial test Program License Conditions - Staff Handouts Draft Inspection Procedure 65001.20.	Meeting Briefing Package/Handouts	NRC/NRO/DNRL/NWE1		05200014 05200015 05200018 05200019 05200022 05200023 05200025 05200026 05200027 05200028 05200029 05200030 05200040 05200041
5/25/2011	ML111470774	William States Lee - Letter to J. Holling, SCDNR, Request for Species Occurrences.	Letter	Battelle Memorial Institute, Pacific Northwest National Lab	NRC/NRO State of SC, Dept of Natural Resources	05200018 05200019

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5/25/2011	ML111470794	William States Lee - Letter to H. LeGrand, NCDENR, Request for Species Occurrences at the Perkins Site.	Letter	Battelle Memorial Institute, Pacific Northwest National Lab	State of SC, Dept of Natural Resources NRC/NRO	05200018 05200019
6/6/2011	ML11158A171	William States Lee Environmental Review - 5/26/2011 Phone Call Record With SC DHEC & Ninety-nine Islands Reference Material.	Note to File incl Telcon Record, Verbal Comm Meeting Summary	Battelle Memorial Institute, Pacific Northwest National Lab	NRC/NRO State of SC, Dept of Health & Environmental Control	05200018 05200019
6/7/2011	ML111400028	Summary of May 3, 2011, Teleconference Between NRC and Duke Energy Carolinas, LLC, Regarding the William States Lee III Nuclear Station, Units 1 and 2 Combined License Application Environmental Review.	Memoranda Meeting Summary	NRC/NRO/DSER/RAP3	NRC/NRO/DSER/RAP3	05200018 05200019
6/8/2011	ML111741378	SC Dept. of Natural Resources Natural Heritage Species for Lee Nuclear Station and Alternative Sites.	E-Mail	Battelle Memorial Institute, Pacific Northwest National Lab	NRC/NRO/DSER/RAP3	05200018 05200019

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6/9/2011	ML111470482	05/24/2011 Summary of Public Meeting with the AP1000 Design Centered Working Group (DCWG) To Discuss the Closure Plan for AP1000 Piping Design Acceptance Criteria (DAC) and Initial Test Program (ITP) License Conditions.	Meeting Summary Memoranda	NRC/NRO/DNRL/NWE1	NRC/NRO/DNRL/NWE1	05200014 05200015 05200018 05200019 05200022 05200023 05200026 05200027 05200028 05200029 05200030 05200040 05200041
6/9/2011	ML11160A103	2011/06/09 Lee RAI for SER - LTR NO. 097 RELATED TO SRP 14.03 AND 13.06 FOR THE WS LEE COLA	E-Mail Request for Additional Information (RAI)	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
6/9/2011	ML11161A135	William States Lee III Nuclear Station Units 1 and 2, Regarding Departure Report Update.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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				Duke Energy Corp		
6/13/2011	ML11171A303	Westinghouse AP1000 Design Control Document Rev. 19 - Introduction	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
6/16/2011	ML11172A288	AP1000 Combined License Application for the William States Lee III Nuclear Station Units 1 & 2, Responses to Request for Additional Information, RAI 190, Supplement, Site Layout & Plant Description and RAI 210, Supplement, Ecology, Aquatic.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
6/16/2011	ML11172A315	William States Lee III, Units 1 and 2, AP1000 Combined License Application - Supplemental Response to Request for Additional Information (ER RAI 63).	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
6/19/2011	ML112700892	Brockington 2010 Cultural Resources Survey of the Proposed London Creek Reservoir, Water Pipeline, Railroad Corridor, Transmission Line, SC 329 Realignment, Railroad Culvert, Water Pipeline Additions, Spoils Areas and	Environmental Report	Brockington & Associates, Inc	Duke Energy Carolinas, LLC NRC/NRO	05200018 05200019

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
		Road Widening's, Part 2 of 2.				
6/23/2011	ML111741383	NC Dept. of Environment and Natural Resources Natural Heritage Species List for Perkins Alternative Site.	E-Mail	NRC/OIP	State of NC, Dept of Environment & Natural Resources	05200018 05200019
6/23/2011	ML11179A079	William States Lee III, Units 1 and 2 - AP1000 Combined License Application, Responses to Request for Additional Information 216 Supplement, Alternatives.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
7/1/2011	ML112800437	USGS 2011 Monthly Stats.	Report, Miscellaneous	US Dept of Interior, Geological Survey (USGS)	NRC/NRO	05200018 05200019
7/1/2011	ML112800440	USGS 2011 Surface-Water Data for North Carolina.	Report, Miscellaneous	US Dept of Interior, Geological Survey (USGS)	NRC/NRO	05200018 05200019

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7/1/2011	ML112800442	USGS 2011 Surface-Water Data for South Carolina.	Report, Miscellaneous	US Dept of Interior, Geological Survey (USGS)	NRC/NRO	05200018 05200019
7/5/2011	ML11195A165	William States Lee III, AP1000 Combined License Application, Response to Request for Additional Information RAI 189 Supplement, Site Layout and Plant Description.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
7/5/2011	ML11195A171	William States Lee III, Units 1 and 2, Figure 3.1-6, Rev. 1.	Map	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
7/6/2011	ML11192A054	William States Lee III, Units 1 and 2 - AP1000 Combined License Application, Response to Request for Additional Information Nos. 4225, 4226, and 5672.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO NRC/NSIR	05200018 05200019
7/8/2011	ML11194A008	William States Lee III, Units 1 and 2 - AP1000 Combined License Application, Responses to Request for Additional Information.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
7/19/2011	ML11151A083	Cover Letter: William States Lee III Nuclear Station, Units 1 And 2 Combined License Application - Advanced Safety Evaluation Without Open Items For Chapter 16, "Technical Specification".	Letter	NRC/NRO/DNRL/NWE1	Duke Energy Carolinas, LLC	05200018 05200019

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7/19/2011	ML11151A092	William States Lee III Nuclear Station, Units 1 & 2, Combined License Application - Advanced Safety Evaluation Without Open Items for Chapter 16, "Technical Specification."	Memoranda	NRC/NRO/DNRL	NRC/ACRS	05200018 05200019
7/28/2011	ML11214A028	William States Lee III , Units 1 & 2, AP1 000 Combined License Application for Concurrence with Standard Content Regarding Supplemental Information in Support of a Special Nuclear Material License Application & Voluntary Revision to Part 10.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
7/28/2011	ML112440727	Lee Nuclear Station Environmental Review - Call Record with FERC Regarding Ninety-Nine Islands Hydro License.	Note to File incl Telcon Record, Verbal Comm	NRC/NRO/DSER/RAP3	US Federal Energy Regulatory Commission	05200018 05200019
7/29/2011	ML11229A563	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report) - FSAR Chapter 03 Figure 3.7-202	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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7/29/2011	ML11229A605	Duke Energy WSL III Units 1 & 2 COLA (ITAAC) - Part 10, Conditions and ITAAC	Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
7/29/2011	ML11229A607	Duke Energy WSL III Units 1 & 2 COLA (Technical Specifications) - Part 4, Williams Lee III Nuclear Station Technical Specifications	Technical Specifications License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
7/29/2011	ML11229A665	Update for William States Lee III Units 1 & 2 Combined License Application.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
7/29/2011	ML11229A665	Update for William States Lee III Units 1 & 2 Combined License Application.	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
7/29/2011	ML11229A202	Duke Energy WSL III Units 1 & 2 COLA (Generic DCD Departures Report) - Part 7, Departures and Exemptions Requests	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
7/29/2011	ML11229A602	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report) Revision 4	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019

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7/29/2011	ML11229A603	Duke Energy WSL III Units 1 & 2 COLA (General and Financial Information) - Part 1, General and Financial Information	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
7/29/2011	ML11229A605	Duke Energy WSL III Units 1 & 2 COLA (ITAAC) - Part 10, Conditions and ITAAC	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
8/4/2011	ML112220298	Attachment 216S-02, Hybrid Cooling System Performance Model Data.	Spreadsheet File	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
8/4/2011	ML11222A129	William States Lee III, Units 1 and 2, AP1000 Combined License Application - Response to Request for Additional Information 216 Supplement, Alternatives.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
8/11/2011	ML11223A486	Motion to Admit New Contention Regarding the Safety and Environmental Implications of the Nuclear Regulatory Commission Task Force Report on the Fukushima Dai-Ichi Accident.	Legal-Petition To Intervene/Request for Hearing	Blue Ridge Environmental Defense League	NRC/ASLBP	05200018 05200019
8/17/2011	ML11216A256	Trip Report - July 11-15, 2011, Geologic Site Visit In Support Of The William States Lee Iii Combined License Application.	Trip Report Memoranda	NRC/NRO/DNRL/NWE1	NRC/NRO/DNRL/NWE1	05200018 05200019

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8/18/2011	ML11230B325	Order Referring Motions To Reopen the Record and/or Admit a New Contention before the Atomic Safety and Licensing Boards that Earlier Conducted the Contested Portion of the Captioned Proceedings.	Legal-Order	NRC/SECY	NRC/OCM	05200018 05200019
8/18/2011	ML11234A242	William States Lee III, Units 1 and 2 Combined License Application - Updated Roadmap of Changes.	Letter Report, Miscellaneous	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
8/22/2011	ML11234A007	2011/08/22 Lee RAI for SER - LTR NO. 098 RELATED TO SRP 13.07 FOR THE WS LEE COLA	E-Mail Request for Additional Information (RAI)	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
8/22/2011	ML112710384	Lee Nuclear Station Preliminary JD Calculations				05200018 05200019
8/23/2011	ML112710643	National Park Service, National Register of Historic Places 2011.	Database File	US Dept of Interior, National Park Service	NRC/NRO	05200018 05200019

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8/25/2011	ML11241A149	Oconee, McGuire, Catawba, Oconee Independent Spent Fuel Storage Installation and William States Lee III - Submittal of Service/Distribution Listings for Routine Information, Official Use Only and Safeguards Information.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NMSS NRC/NRO NRC/NRR	05000269 05000270 05000287 05000369 05000370 05000413 05200018 05200019 07200004 05000414

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8/26/2011	ML112360228	09/21/2011 Notice of Forthcoming Meeting with AP1000 Design-Centered Working Group (DCWG) to Discuss Piping Systems Design Acceptance Criteria (DAC).	Meeting Notice Memoranda	NRC/NRO/DCIP/CIPB	NRC/NRO/DCIP/CIPB	05200006 05200014 05200015 05200018 05200019 05200022 05200023 05200026 05200027 05200028 05200029 05200030 05200040 05200041
8/30/2011	ML11242A057	Notice of Appearance for Kevin C. Roach, Office of the General Counsel.	Legal-Correspondence/Miscellaneous	NRC/OCM	NRC/ASLBP	05200018 05200019

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8/31/2011	ML112450498	William States Lee III Nuclear Station Units 1 and 2 National Pollutant Discharge Elimination System Permit Application August 2011.	Environmental Report Report, Technical	Duke Energy Carolinas, LLC	NRC/NRO State of SC, Dept of Health & Environmental Control	05200018 05200019
9/6/2011	ML11249A140	NRC Staff Answer to Intervenors' Motion to Admit New Contention Regarding the Safety and Environmental Implications of the NRC Task Force Report on the Fukushima Dai-Ichi Accident.	Legal-Pleading	NRC/OGC	NRC/ASLBP	05200018 05200019
9/6/2011	ML11249A141	Duke Energy's Opposition to BREDL's New Contention.	Legal-Pleading	Duke Energy Carolinas, LLC Duke Energy Corp Pillsbury, Winthrop, Shaw, Pittman, LLP	NRC/ASLBP	05200018 05200019
9/6/2011	ML11249A231	Establishment of Atomic Safety and Licensing Board.	Legal-Order	NRC/ASLBP		05200025 05200018 05200019 05200026

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
9/9/2011	ML112521039	Commission Memorandum and Order (CLI-11-05) regarding PR 52 AP1000 Design Certification Amendment.	Legal-Order Rulemaking-Comment	NRC/SECY		05000247 05000275 05000286 05000293 05000323 05000346 05000397 05000443 05200016 05200018 05200019 05200022 05200029 05200033 05200034 05200035 05200037

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						05200039 05200040 05200041 05000391 05200012 05200013 05200014 05200015 05200017 05200023 05200025 05200026 05200027 05200028 05200030

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
9/9/2011	ML112521106	Commission Memorandum and Order (CLI-11-05) regarding PR 52 ESBWR Design Certification Amendment.	Legal-Memorandum and Order Rulemaking-Comment	NRC/SECY		05000247 05000275 05000286 05000293 05000323 05000346 05000391 05000397 05000443 05200012 05200013 05200014 05200015 05200016 05200017 05200018 05200019

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						05200022 05200023 05200025 05200026 05200027 05200028 05200029 05200030 05200033 05200034 05200035 05200037 05200039 05200040 05200041

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9/9/2011	ML11252B035	Commission Memorandum and Order CLI-11-05, Denying Suspension Petitions, Addressing Additional Requests for Relief, and Granting a Request for a Safety Analysis.	Legal-Order	NRC/SECY	NRC/Chairman NRC/OCM	05200018 05200019
9/13/2011	ML112570445	Eastern Band of Cherokee Indians Comment on Proposed Duke Energy William States Lee Nuclear Station.	Letter	Eastern Band of Cherokee Indians	NRC/NRO/DSER/RAP3	05200018 05200019
9/13/2011	ML11262A000	Intervenor's Memorandum in Reply to Oppositions to Admission of New Contention.	Legal-Pleading	Blue Ridge Environmental Defense League	NRC/ASLBP	05200018 05200019
9/14/2011	ML11259A044	William States Lee III Nuclear Station, Units 1 and 2 - AP1000 Combined License Application for the Supplemental Response to Request for Additional Information, Ltr # WLG2011.09-05.	Legal-Affidavit Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
9/14/2011	ML11259A045	William States Lee III Nuclear Station, Units 1 and 2 - Supplemental Response to RAI Nos. 13.06-16 (Letter 078) and 13.06-32 (Letter 079) Ltr # WLG2011.09-01.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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9/15/2011	ML11262A205	William States Lee III, Units 1 and 2, AP1000 Combined License Application, 2011 Integrated Resource Plan.	Annual Operating Report Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
9/19/2011	ML112710656	Savannah River Ecology Laboratory Herpetology Program 2011.	- No Document Type Applies	Savannah River Lab	NRC/NRO	05200018 05200019
9/19/2011	ML112760804	LeGrand et al. 2010.	Brochure	North Carolina Natural Heritage Program	NRC/NRO	05200018 05200019
9/19/2011	ML112760805	Sauer et al. 2007.	Database File Report, Miscellaneous	US Dept of Interior, Geological Survey (USGS)	NRC/NRO	05200018 05200019
9/19/2011	ML112760806	SCDNR 2010 Rare.	Database File	State of SC, Dept of Natural Resources	NRC/NRO	05200018 05200019
9/20/2011	ML112760813	SCDNR 2010 Breeding Bird.	Report, Miscellaneous	State of SC, Dept of Natural Resources	NRC/NRO	05200018 05200019
9/20/2011	ML112800260	USFWS 2003 Red Cockaded Woodpecker.	Environmental Report	US Dept of Interior, Fish & Wildlife Service	NRC/NRO	05200018 05200019

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9/21/2011	ML11266A048	William States Lee III, Units 1 and 2, AP1000 Combined License Application, Response to No. 13.07-1 (Letter No. 098).	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
9/22/2011	ML11265A262	Order Granting Leave to File Supplement to Answer.	Legal-Order	NRC/ASLBP	NRC/ASLBP	05200026 05200025 05200019 05200018
9/28/2011	ML11271A169	Duke Energy's Motion to Strike.	Legal-Pleading	Duke Energy Carolinas, LLC Duke Energy Corp Pillsbury, Winthrop, Shaw, Pittman, LLP	NRC/ASLBP	05200019 05200018
10/3/2011	ML11224A216	William States Lee III Nuclear Station, Units 1 And 2 Combined License Application Review Schedule Revision, 08/12/2011.	Letter	NRC/NRO/DNRL	Duke Energy Nuclear, LLC	05200018 05200019
10/4/2011	ML112790295	FERC Participating Agency Invitation for the William States Lee Nuclear Station Environmental Review.	E-Mail	NRC/NRO/DSER/RAP3	Federal Eastern Corp	05200018 05200019

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10/5/2011	ML112780203	10/25/11 Notice of Category 2 Public Meeting With AP1000 Design-Centered Working Group (DCWG) to Discuss Digital Instrumentation and Control (DI&C) Systems Design Acceptance Criteria (DAC).	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DCIP/CIPB	NRC/NRO/DCIP/CIPB	05200014 05200015 05200018 05200019 05200022 05200023 05200025 05200026 05200027 05200028 05200029 05200030 05200040 05200041
10/5/2011	ML112790296	FERC Participating Agency Acceptance Letter for William States Lee III Nuclear Station, Units 1 and 2, Environmental Review.	Letter	US Federal Energy Regulatory Commission	NRC/NRO/DSER/RAP3	05200018 05200019

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10/10/2011	ML11285A420	Intervenor's Reply to Duke Energy Motion to Strike.	Legal-Pleading Legal-Motion	Blue Ridge Environmental Defense League	NRC/ASLBP	05200018 05200019
10/11/2011	ML11152A251	Letter - William States Lee III Nuclear Station, Units 1 And 2 Combined License Application - Advanced Safety Evaluation Without Open Items For Chapter 7, "Instrumentation And Controls".	Letter	NRC/NRO/DNRL/NWE1	Duke Energy Carolinas, LLC	05200018 05200019
10/11/2011	ML112800542	09/21/11 Summary of Meeting with AP1000 Design-Centered Working Group (DWCG) re: AP1000 Piping Design Acceptance Criteria memo.	Meeting Summary	NRC/NRO/DCIP/CIPB	NRC/NRO/DCIP/CIPB	05200006 05200014 05200015 05200018 05200019 05200022 05200023 05200025 05200026 05200027 05200028

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						05200029 05200030 05200040 05200041
10/13/2011	ML11152A254	William States Lee III Nuclear Station, Units 1 And 2 Combined License Application - Advanced Safety Evaluation Without Open Items For Chapter 7 "Instrumentation And Controls".	Memoranda	NRC/NRO/DNRL	NRC/ACRS	05200018 05200019
10/13/2011	ML113260104	GDNR 2011 Rare Species Location.	- No Document Type Applies	State of GA, Wildlife Federation	NRC/NRO	05200018 05200019
10/14/2011	ML113260221	NCDENR 2010 Natural.	Database File	State of NC, Dept of Environment & Natural Resources	NRC/NRO	05200018 05200019

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10/18/2011	ML112760826	Summary of William States Lee III Nuclear Station, Units 1 and 2, Cooling Systems and Energy Alternatives Audit.	Memoranda Audit Report	NRC/NRO/DSER/RAP3	NRC/NRO/DSER/RAP3	05200018 05200019
10/18/2011	ML11291A129	Memorandum and Order (Denying Motions To Reopen Closed Proceedings and Intervention Petition / Hearing Request as Premature).	Legal-Order	NRC/ASLBP	Blue Ridge Environmental Defense League	05200026 05200025 05200019 05200018
10/19/2011	ML110340095	ASE - Lee ASE Ch 17 Clean Master.	NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/NWE1		05200018 05200019
10/19/2011	ML11171A053	Lee ASE Ch 17 Cover Letter.	Letter	NRC/NRO/DNRL/NWE1	Duke Energy Carolinas, LLC	05200018 05200019
10/20/2011	ML11293A070	Memorandum (Corrections Regarding LBP-11-27).	Legal-Order	NRC/ASLBP	Duke Energy Carolinas, LLC Energy Northwest Luminant Generation Co, LLC PPL Bell Bend, LLC Southern	05200026 05200025 05200019 05200018

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					Nuclear Operating Co, Inc	
10/20/2011	ML112940761	10/25/2011 Public Meeting - DAC Inspection Process Flowchart (Draft).	Meeting Briefing Package/Handouts	NRC/NRO/DCIP/CIPB		05200014 05200015 05200018 05200019 05200022 05200023 05200025 05200026 05200027 05200028 05200029 05200030 05200040

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						05200041
10/24/2011	ML11171A055	Lee ASE Ch 17 ACRS Memo.	Memoranda	NRC/NRO/DNRL	NRC/ACRS	05200018 05200019
10/26/2011	ML11299A120	2011/10/26 Lee RAI for SER - LTR NO. 099 RELATED TO SRP SECTION 13.07 FOR THE WS LEE COLA	E-Mail Request for Additional Information (RAI)	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
10/28/2011	ML11301A363	Motion to Reinstate and Supplement the Basis for Fukushima Task Force Report Contention.	Legal-Motion	Blue Ridge Environmental Defense League	NRC/ASLBP	05200018 05200019
10/28/2011	ML11301A364	Motion to Reinstate and Supplement the Basis for Fukushima Task Force Report Contention.	Legal-Motion	Blue Ridge Environmental Defense League	NRC/ASLBP	05200019 05200018

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11/1/2011	ML11182B036	Lee ASE Ch 14 Cover Letter.	Letter	NRC/NRO/DNRL/NWE1	Duke Energy Carolinas, LLC	05200018 05200019
11/2/2011	ML11182B046	Lee ASE Ch 14 ACRS Memo.	Memoranda	NRC/NRO/DNRL	NRC/ACRS	05200018 05200019
11/2/2011	ML11306A434	Petition for Review of LBP-11-27.	Legal-Petition To Intervene/Request for Hearing	Blue Ridge Environmental Defense League Kauffman & Eye Northwest Environmental Advocates Turner Environmental Law Clinic	NRC/OCM	05200026 05200025 05200019 05200018
11/7/2011	ML11311A288	Duke Energy's Opposition to BREDL'S Motion to Reinstate and Supplement the Basis for Its Fukushima-Related Contention.	Legal-Pleading	Pillsbury, Winthrop, Shaw, Pittman, LLP Duke Energy Carolinas, LLC	NRC/ASLBP	05200019 05200018
11/7/2011	ML11311A304	NRC Staff's Answer to Motion to Reinstate and Supplement the Basis for Fukushima Task Force Report Contention.	Legal-Pleading	NRC/OGC	NRC/ASLBP	05200026 05200025 05200019

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
						05200018
11/9/2011	ML11313A170	2011/11/09 Lee RAI for SER - RAI LTR 100 related to SRP Section 3.7 Seismic Analysi for the W.S. Lee Unit 1 & 2 COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
11/9/2011	ML11313A213	2011/11/09 Lee RAI for SER - RAI-LTR-100 RELATED TO SRP SECTION 3.7 SEISMIC SYSTEMS ANALYSISFOR THE W.S. LEE COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
11/14/2011	ML11236A221	Federal Register Notification of the Availability of the Combined License Application for William States Lee III Nuclear Station Units 1 and 2.	Letter	NRC/NRO/DNRL/NWE1	Duke Energy Nuclear, LLC	05200018 05200019
11/14/2011	ML11318A086	Duke Energy's Answer to Petition for Review of LBP-11-27.	Legal-Pleading	Duke Energy Corp Pillsbury, Winthrop, Shaw, Pittman, LLP	NRC/OCM	05200019 05200018

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11/14/2011	ML11318A108	Notice of Appearance for Laura R. Goldin.	Legal-Correspondence/Miscellaneous	NRC/OGC	NRC/OCM	05200035 05200026 05200025 05200034 05200018 05200019
11/14/2011	ML11318A112	NRC Staff Answer to Petition for Review of LBP-11-27.	Legal-Pleading	NRC/OGC	NRC/OCM	05200035 05200026 05200025 05200034 05200018 05200019
11/14/2011	ML11318A118	Notice of Appearance For Laura R. Goldin on Behalf of the U.S. Nuclear Regulatory Commission in the Matter of Comanche Peak, Units 3 and 4, Vogtle, Units 3 and 4 and William States Lee, Units 1 and 2.	Legal-Correspondence/Miscellaneous	NRC/OGC	NRC/OCM	05200026 05200025 05200019 05200018 05200034 05200035

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11/14/2011	ML11318A120	NRC Staff's Answer to Petition for Review of LBP-11-27.	Legal-Pleading	NRC/OGC	NRC/OCM	05200026 05200025 05200019 05200018 05200034 05200035
11/15/2011	ML12173A295	FERC Order Amending Article 402 of Ninety-Nine Islands Hydroelectric Project License.	Order	US Federal Energy Regulatory Commission	NRC/NRO	05200018 05200019
11/17/2011	ML113250035	William States Lee III, Units 1 and 2, AP1000 Combined License Application for the Safeguards/Security Plans.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
11/17/2011	ML113250037	William States Lee III Nuclear Station - AP1000 Combined License Application for the Safeguards/Security Plans - Reviewer's Guide Ltr # WLG2011.11-02.	Letter	Duke Energy Corp Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
11/18/2011	ML11145A016	Lee ASE Ch 15 Cover Letter.	Letter NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/NWE1	Duke Energy Carolinas, LLC	05200018 05200019

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11/18/2011	ML11145A018	Lee ASE Ch 15 ACRS Memo-William States Lee III Nuclear Station, Units 1 And 2 Combined License Application - Advanced Safety Evaluation Without Open Items For Chapter 15, "Accident Analysis".	Memoranda	NRC/NRO/DNRL	NRC/ACRS	05200018 05200019
11/21/2011	ML11236A136	Second 50.43a FRN for Lee-Harris-Levy-Turkey Point.	Federal Register Notice	NRC/NRO/DNRL/NWE1		05200022 05200023 05200018 05200019 05200029 05200030 05200040 05200041
11/21/2011	ML12065A124	DUK-001-PR-01, Rev. 2, "Cherokee Nuclear Station Final Foundation Geologic Map Record", Figure 7 Through Figure 10.	Environmental Report	Fugro	NRC/NRO	05200018 05200019
11/21/2011	ML12065A125	DUK-001-PR-01, Rev. 2, "Cherokee Nuclear Station Final Foundation Geologic Map Record", Figure 11 Through Figure 22.	Environmental Report	Fugro	NRC/NRO	05200018 05200019

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11/21/2011	ML12065A126	DUK-001-PR-01, Rev. 2, "Cherokee Nuclear Station Final Foundation Geologic Map Record", Table of Contents Through Figure 6.	Environmental Report	Fugro	NRC/NRO	05200018 05200019
11/21/2011	ML12065A129	DUK-001-PR-01, Rev. 2, "Cherokee Nuclear Station Final Foundation Geologic Map Record", Plate 1.	Environmental Report	Fugro	NRC/NRO	05200018 05200019
11/21/2011	ML12065A130	DUK-001-PR-01, Rev. 2, "Cherokee Nuclear Station Final Foundation Geologic Map Record", Appendix B.	Environmental Report	Fugro	NRC/NRO	05200018 05200019
11/21/2011	ML12065A132	DUK-001-PR-01, Rev. 2, "Cherokee Nuclear Station Final Foundation Geologic Map Record", Appendix C.	Environmental Report	Fugro	NRC/NRO	05200018 05200019
11/21/2011	ML12065A133	DUK-001-PR-01, Rev. 2, "Cherokee Nuclear Station Final Foundation Geologic Map Record", Appendix A.	Environmental Report	Fugro	NRC/NRO	05200018 05200019
11/21/2011	ML12065A134	DUK-001-PR-01, Rev. 2, "Cherokee Nuclear Station Final Foundation Geologic Map Record", Appendix D.	Environmental Report	Fugro	NRC/NRO	05200018 05200019
11/21/2011	ML12065A136	DUK-001-PR-01, Rev. 2, "Cherokee Nuclear Station Final Foundation Geologic Map Record", Appendix E.	Environmental Report	Fugro	NRC/NRO	05200018 05200019

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11/21/2011	ML12065A150	DUK-001-PR-01, Rev. 2, "Cherokee Nuclear Station Final Foundation Geologic Map Record", Appendix H, Pages H1 of H414 through H138 of H414.	Environmental Report	Fugro	NRC/NRO	05200018 05200019
11/21/2011	ML12065A151	DUK-001-PR-01, Rev. 2, "Cherokee Nuclear Station Final Foundation Geologic Map Record", Appendix H, Pages H139 of H414 through H276 of H414.	Environmental Report	Fugro	NRC/NRO	05200018 05200019
11/21/2011	ML12065A152	DUK-001-PR-01, Rev. 2, "Cherokee Nuclear Station Final Foundation Geologic Map Record", Appendix G, Sheet 5.	Environmental Report	Fugro	NRC/NRO	05200018 05200019
11/21/2011	ML12065A153	DUK-001-PR-01, Rev. 2, "Cherokee Nuclear Station Final Foundation Geologic Map Record", Appendix G, Sheet 6.	Environmental Report	Fugro	NRC/NRO	05200018 05200019
11/21/2011	ML12065A154	DUK-001-PR-01, Rev. 2, "Cherokee Nuclear Station Final Foundation Geologic Map Record", Appendix G, Sheet 7.	Environmental Report	Fugro	NRC/NRO	05200018 05200019
11/21/2011	ML12065A155	DUK-001-PR-01, Rev. 2, "Cherokee Nuclear Station Final Foundation Geologic Map Record", Appendix H, Pages H277 of H414 through H414 of H414.	Environmental Report	Fugro	NRC/NRO	05200018 05200019

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11/21/2011	ML12065A156	DUK-001-PR-01, Rev. 2, "Cherokee Nuclear Station Final Foundation Geologic Map Record", Appendix I through End.	Environmental Report	Fugro	NRC/NRO	05200018 05200019
11/21/2011	ML12065A160	DUK-001-PR-01, Rev. 2, "Cherokee Nuclear Station Final Foundation Geologic Map Record", Appendix G, Sheet 2.	Environmental Report	Fugro	NRC/NRO	05200018 05200019
11/21/2011	ML12065A161	DUK-001-PR-01, Rev. 2, "Cherokee Nuclear Station Final Foundation Geologic Map Record", Appendix G, Sheet 3.	Environmental Report	Fugro	NRC/NRO	05200018 05200019
11/21/2011	ML12065A162	DUK-001-PR-01, Rev. 2, "Cherokee Nuclear Station Final Foundation Geologic Map Record", Appendix F, Table of Contents through Panel 49.	Environmental Report	Fugro	NRC/NRO	05200018 05200019
11/21/2011	ML12065A163	DUK-001-PR-01, Rev. 2, "Cherokee Nuclear Station Final Foundation Geologic Map Record", Appendix F, Panel 50 through Appendix F, Panel SB2.	Environmental Report	Fugro	NRC/NRO	05200018 05200019
11/21/2011	ML12065A164	DUK-001-PR-01, Rev. 2, "Cherokee Nuclear Station Final Foundation Geologic Map Record", Appendix G, Sheet 4.	Environmental Report	Fugro	NRC/NRO	05200018 05200019

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11/21/2011	ML12065A165	DUK-001-PR-01, Rev. 2, "Cherokee Nuclear Station Final Foundation Geologic Map Record", Appendix G, Table of Contents.	Environmental Report	Fugro	NRC/NRO	05200018 05200019
11/21/2011	ML12065A166	DUK-001-PR-01, Rev. 2, "Cherokee Nuclear Station Final Foundation Geologic Map Record", Appendix G, Sheet 1.	Environmental Report	Fugro	NRC/NRO	05200018 05200019
11/22/2011	ML113260080	GBPW 2010 About Us.	Database File	Gaffney, SC, Board of Public Works	NRC/NRO	05200018 05200019
11/28/2011	ML11236A137	Third 50.43a FRN for Lee-Harris-Levy-Turkey Point.	Federal Register Notice	NRC/NRO/DNRL/NWE1		05200022 05200023 05200018 05200019 05200029 05200030 05200040 05200041

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12/5/2011	ML11236A139	Fourth 50.43a FRN for Lee-Harris-Levy-Turkey Point.	Federal Register Notice	NRC/NRO/DNRL/NWE1		05200022 05200023 05200018 05200019 05200029 05200030 05200040 05200041
12/8/2011	ML12125A313	The 2006-2010 ACS 5-Year Summary File Technical Documentation				05200018 05200019
12/12/2011	ML112940233	Letter to Environmental Protection Agency on Submittal of the Draft Environmental Impact Statement for the Williams States Lee III Nuclear Station Units 1 and 2 Combined Licenses Application Review.	Letter	NRC/NRO/DNRL	US Environmental Protection Agency (EPA)	05200018 05200019
12/12/2011	ML112940260	Letter to R. Jones, Duke Energy Carolinas, Notice of Availability of Draft Environmental Impact Statement for Williams States Lee III Nuclear Station Units 1 and 2.	Letter	NRC/NRO/DNRL/EPB1	Duke Energy Carolinas, LLC	05200018 05200019

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12/12/2011	ML112940305	Federal Register Notice-William States Lee III Nuclear Station Draft Environmental Impact Statement.	Federal Register Notice	NRC/NRO/D NRL		05200018 05200019
12/12/2011	ML11313A167	Letter to Dept. of Health and Environmental Control Regarding the Draft Environmental Impact Statement for the Williams States Lee Combined Licenses Application.	Letter	NRC/NRO/D NRL	State of SC, Dept of Health & Environmental Control	05200018 05200019
12/12/2011	ML11314A229	Letter to R. Perry, South Carolina Dept. of Natural Resources Regarding the William States Lee Nuclear Station Draft Environmental Impact Statement.	Letter	NRC/NRO/D SER/RAP1	State of SC, Dept of Natural Resources	05200018 05200019
12/12/2011	ML11319A017	Letter to the North Carolina Wildlife Resources Commission, Regarding the Draft Environmental Impact Statement for William States Lee III Nuclear Station, Units 1 and 2.	Letter	NRC/NRO/D NRL/EPB1	State of NC, Wildlife Resources Commission	05200018 05200019
12/12/2011	ML11319A023	Letter to R. McConney, EPA Region 4, Regarding the Draft Environmental Impact Statement for William States Lee III Nuclear Station, Units 1 and 2.	Letter	NRC/NRO/D NRL	US Environmental Protection Agency (EPA)	05200018 05200019
12/15/2011	ML112450007	Lee - Notice of Availability of COLA to Federal Energy Regulatory Commission.	Letter	NRC/NRO/D NRL/LB4	US Federal Energy	05200018 05200019

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					Regulatory Commission	
12/15/2011	ML112450014	Lee - Notice of Availability of COLA to Public Service Commission.	Letter	NRC/NRO/DNRL/NWE1	State of SC, Public Service Commission	05200018 05200019
12/15/2011	ML112450028	Lee - Notification of Availability of COLA to Utilities Commission.	Letter	NRC/NRO/DNRL/NWE1	State of NC, Utilities Commission	05200018 05200019
12/15/2011	ML12065A114	William States Lee III Nuclear Station Units 1 & 2, AP1000 Combined License Application - Cherokee Nuclear Station Final Foundation Geologic Map Record.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
12/16/2011	ML11146A004	Lee ASE Chapter 09 Cover Letter.	Letter NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/LB4	Duke Energy Carolinas, LLC	05200018 05200019
12/19/2011	ML11146A007	David Mathews Memo: Lee ASE Ch 09 ACRS Memo.	Memoranda	NRC/NRO/DNRL/NWE1	NRC/ACRS	05200018 05200019
12/21/2011	ML12033A159	Comment (6) of Kimchi Rylander, on Behalf of Earthaven Ecovillage, Opposing Approval of the William States Lee III Units 1 and 2.	General FR Notice Comment Letter	Earthaven Ecovillage	NRC/ADM/DAS/RDEB	05200018 05200019

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12/22/2011	ML12039A295	Duke Energy Carolinas, LLC, Response to Comments on NPDES Permit Application William States Lee III Nuclear Station, Attachment 4 through End.	- No Document Type Applies Report, Miscellaneous	Duke Energy Carolinas, LLC	NRC/NRO State of SC, Dept of Environmental Control, Bureau of Water	05200018 05200019
12/22/2011	ML12039A296	Duke Energy Responses to SCDHEC Technical Comments on Lee Nuclear Station NPDES Application.	Letter	Duke Energy Carolinas, LLC	NRC/NRO State of SC, Dept of Environmental Control, Bureau of Water	05200018 05200019
12/22/2011	ML12039A297	Duke Energy Carolinas, LLC, Response to Comments on NPDES Permit Application William States Lee III Nuclear Station, Cover through Attachment 3.	- No Document Type Applies Calculation Report, Miscellaneous	Duke Energy Carolinas, LLC	NRC/NRO State of SC, Dept of Environmental Control, Bureau of Water	05200018 05200019
12/27/2011	ML113610360	EPA Region 4 K-J Shell Comment on William States Lee III Nuclear Station Units 1 and 2 Draft Environmental Impact Statement.	E-Mail	US Environmental Protection Agency (EPA)	NRC/NRO/DNRL	05200018 05200019
1/3/2012	ML12023A052	Comment (2) of Leah R. Karpen Opposing the Draft Environmental Impact Statement for the William States Lee Nuclear Station.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019

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1/4/2012	ML101270154	ASE - Lee ASE Chapter 4 CLEAN Master.	NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/NWE1		05200018 05200019
1/5/2012	ML12151A384	Comment on Lee Nuclear Station Draft EIS - V. Boever.	Letter	- No Known Affiliation	NRC/NRO	05200018 05200019
1/6/2012	ML12048A662	Comment (9) of David Rittenberg Opposing the Construction of the William States Lee Nuclear Power Plant.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
1/8/2012	ML12060A279	Comment (24) of Ella Boyle Opposing Duke Energy's Combined License Application (COL) To Build William States Lee Nuclear Power Plant In Gaffney, SC.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
1/9/2012	ML11152A127	William States Lee III, Units 1 & 2 Combined License Application - Advanced Safety Evaluation Without Open Items for Chapter 19, "Probabilistic Risk Assessment (PRA)".	Letter	NRC/NRO/DNRL/NWE1	Duke Energy Carolinas, LLC	05200018 05200019
1/9/2012	ML11152A139	Lee ASE Ch 19 (PRA) ACRS Memo.	Memoranda	NRC/NRO/DNRL	NRC/ACRS	05200018 05200019
1/11/2012	ML12023A051	Comment (1) of Patricia Severin Opposing the Approval of the William States Lee III Units 1 and 2.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019

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1/15/2012	ML12033A156	Comment (3) of Susan Broadhead Opposing William States Lee Nuclear Plant.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
1/16/2012	ML12039A135	Written Comment (6) from Sky Conard Regarding Lee Nuclear Station Draft Environmental Impact Statement.	Environmental Impact Statement	Green River Watershed Alliance	NRC/NRO	05200018 05200019
1/17/2012	ML12018A388	William States Lee III Nuclear Station, Units 1 and 2, AP1000 Combined License Application for the Response to Request for Additional Information (RAI No. 6183) Ltr# WLG2012.01-01.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
1/18/2012	ML11147A083	Lee ASE Ch 05 Cover Letter, Advanced Safety Evaluation Without Open Items for Chapter 5, "Reactor Coolant System and Connected Systems."	Letter	NRC/NRO/DNRL/LB4	Duke Energy Carolinas, LLC	05200018 05200019
1/18/2012	ML11147A092	Lee ASE Ch 05 ACRS Memo.	Memoranda	NRC/NRO/DNRL	NRC/ACRS	05200018 05200019
1/19/2012	ML120170372	William States Lee Nuclear Station Draft Environmental Impact Statement Public Meeting Handouts.	Meeting Agenda Meeting Briefing Package/Handouts	NRC/NRO/DNRL/EPB1		05200018 05200019

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1/19/2012	ML120170375	William States Lee Nuclear Station Draft Environmental Impact Statement 1/19/2012 Public Meeting Presentation Slides.	Meeting Briefing Package/Handouts Slides and Viewgraphs	NRC/NRO/DNRL/EPB1		05200018 05200019
1/19/2012	ML120260611	1/19/2012 William States Lee Nuclear Station DEIS Public Meeting Transcript - Afternoon.	Meeting Transcript	NRC/NRO/DNRL/EPB1		05200018 05200019
1/19/2012	ML120260614	Transcripts of William States Lee III, Units 1 & 2, Draft Environmental Impact Statement Public Meetings: Evening Session, 1/19/2012, Pages 1-144.	Meeting Transcript	NRC/NRO/DNRL/EPB1		05200018 05200019
1/19/2012	ML12039A130	Written Comment (1) from Ellen Thomas Regarding Lee Nuclear Station Draft Environmental Impact Statement.	Environmental Impact Statement	Women's International League for Peace & Freedom	NRC/NRO	05200018 05200019
1/19/2012	ML12039A131	Written Comment (2) from Katie Hicks, Clean Water for North Carolina, Regarding Lee Nuclear Station Draft Environmental Impact Statement.	Environmental Impact Statement	Clean Water for North Carolina	NRC/NRO	05200018 05200019
1/19/2012	ML12039A132	Written Comment (3) from Cindy McFadden, Cherokee 2020 Volunteer, Regarding Lee Nuclear Station Draft Environmental Impact Statement.	Environmental Impact Statement	- No Known Affiliation	NRC/NRO	05200018 05200019

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1/19/2012	ML12039A133	Written Comment (4) from Irma M. Howarth, Regarding Lee Nuclear Station Draft Environmental Impact Statement.	Environmental Impact Statement	- No Known Affiliation	Blue Ribbon Commission on America's Nuclear Future NRC/NRO	05200018 05200019
1/19/2012	ML12039A134	Written Comment (5) from Robert F. Howarth, Regarding Lee Nuclear Station Draft Environmental Impact Statement.	Environmental Impact Statement	- No Known Affiliation	Blue Ribbon Commission on America's Nuclear Future NRC/NRO	05200018 05200019
1/19/2012	ML12039A136	Written Comment (7) from Anne Craig Regarding Lee Nuclear Station Draft Environmental Impact Statement.	Environmental Impact Statement	- No Known Affiliation	NRC/NRO	05200018 05200019
1/19/2012	ML12039A137	Written Comment (8) from Representative Dennis Carroll Moss Regarding Lee Nuclear Station Draft Environmental Impact Statement.	Environmental Impact Statement	State of SC, House of Representative	NRC/NRO	05200018 05200019
1/19/2012	ML12039A138	Written Comment (9) from Jean Larson Regarding Lee Nuclear Station Draft Environmental Impact Statement.	Environmental Impact Statement	- No Known Affiliation	NRC/ADM/DAS/RDEB NRC/NRO	05200018 05200019
1/19/2012	ML12039A139	Written Comment (10) from Susan Broadhead Regarding Lee Nuclear Station Draft Environmental Impact Statement.	Environmental Impact Statement	- No Known Affiliation	NRC/NRO	05200018 05200019

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1/19/2012	ML12039A140	Written Comment (11) from Lori Greenberg Regarding Lee Nuclear Station Draft Environmental Impact Statement.	Environmental Impact Statement	- No Known Affiliation	NRC/NRO	05200018 05200019
1/19/2012	ML12039A141	Written Comment (12) from Don Richardson Regarding Lee Nuclear Station Draft Environmental Impact Statement.	Environmental Impact Statement	- No Known Affiliation	NRC/NRO	05200018 05200019
1/19/2012	ML12039A142	Written Comment (13) from Dan Gamble Regarding Lee Nuclear Station Draft Environmental Impact Statement.	Environmental Impact Statement	- No Known Affiliation	NRC/NRO	05200018 05200019
1/19/2012	ML12039A143	Written Comment (14) from Ole P. Sorensen Regarding Lee Nuclear Station Draft Environmental Impact Statement.	Environmental Impact Statement	- No Known Affiliation	NRC/NRO	05200018 05200019
1/19/2012	ML12039A144	Written Comment (15) from Philip J. Bisesi Regarding Lee Nuclear Station Draft Environmental Impact Statement.	Environmental Impact Statement	Affiliated Consultants, Engineers	NRC/NRO	05200018 05200019
1/19/2012	ML12039A145	Written Comment (16) from Rachel Bliss Regarding Lee Nuclear Station Draft Environmental Impact Statement.	Environmental Impact Statement	- No Known Affiliation	NRC/NRO	05200018 05200019

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1/20/2012	ML12048A671	Comment (17) of Rebekah Dobrasko of the South Carolina Archives and History Center on the William States Lee III Nuclear Station Draft Environmental Impact Statement.	General FR Notice Comment Letter	State of SC, Dept of Archives and History	NRC/ADM/DAS/RDEB	05200018 05200019
1/25/2012	ML120240470	Lee Memo: Safeguards Evaluation Report Transmittal Memorandum for Duke Energy Carolinas Transportation and Physical Plans.	Memoranda	NRC/NSIR/DSP	NRC/NRO/DNRL	05200018 05200019
1/27/2012	ML120270235	W. S. Lee Report To File Geo Data Audits 02-2009 and 07-2009.	Audit Report	NRC/NRO/DNRL/LB4		05200018 05200019
1/27/2012	ML120270246	W. S. Lee Report To File Geo Data Audits 06-2010 and 10-2011.	Audit Report	NRC/NRO/DSEA		05200018 05200019
1/30/2012	ML12151A382	William States Lee Nuclear Station Comment on Draft EIS - A.E. Keil.	Letter	- No Known Affiliation	NRC/NRO	05200018 05200019
2/1/2012	ML120320233	W. S. Lee Report To File - Lee Site Field Audit - April-May 2008.	Audit Report Photograph Trip Report	NRC/NRO/DSEA/RGS2		05200018 05200019
2/1/2012	ML120320258	W. S. Lee Report To File - Lee Site Field Audit - January 2009 with USGS.	Audit Report Photograph Trip Report	NRC/NRO/DSEA/RGS2		05200018 05200019

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2/1/2012	ML12048A665	Comment (12) of Anne Craig Opposing the Contruction of William States Lee Plant.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
2/1/2012	ML12048A666	Comment (13) of Ray Heane Opposing the Construction of the William States Lee Nuclear Power Plant.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
2/1/2012	ML12048A668	Comment (14) of Deborah Acs Opposing Construction of William States Lee Plant.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
2/1/2012	ML12048A669	Comment (15) of Phyllis Genetti Opposing the Construction of a New Nuclear Facility in South Carolina.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
2/1/2012	ML12058A398	Comment (22) of Lewis E. Patrie, Opposing Duke Energy's License Application for William States Lee Nuclear Power Plant in Gaffney, SC.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
2/1/2012	ML12058A400	Comment (23) of Linda Burnet, Opposing the Building of William States Lee III Nuclear Power Plant in Gaffney, South Carolina.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
2/2/2012	ML12032A115	Restoration Church International Thank You Letter - Lee Nuclear Station DEIS Meetings.	Letter	NRC/NRO/DNRL/EPB1	- No Known Affiliation	05200018 05200019

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2/2/2012	ML12032A180	Gaffney City Hall Thank You Letter - Lee Nuclear Station DEIS Gov't-to-Gov't Meeting.	Letter	NRC/NRO/DNRL/EPB1	Gaffney, SC, Board of Public Works	05200018 05200019
2/2/2012	ML12048A664	Comment (11) of Cathleen Brogan Prindle Opposing the Construction of the William States Lee Nuclear Power Plant.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
2/4/2012	ML12060A278	Comment (25) of Josephine Andrews, Opposing Duke Energy's Combined License Application to Build William States Lee Nuclear Power Plant in Gaffney, SC.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
2/6/2012	ML102560445	Lee ASE Chapter 11 CLEAN 091010.	NRO Safety Evaluation Report (SER)-Delayed	NRC/NRO/DNRL/LB4		05200018 05200019
2/6/2012	ML111640498	Lee ASE Chapter 11 Cover Letter.	Letter	NRC/NRO/DNRL/NWE1	Duke Energy Carolinas, LLC	05200018 05200019
2/6/2012	ML111640502	Lee ASE Chapter 11 ACRS Memo.	Memoranda	NRC/NRO/DNRL	NRC/ACRS	05200018 05200019
2/6/2012	ML12058A396	Comment (20) of John Davis, Opposing Duke Energy's Combined License Application (COL) to Build William States Lee Nuclear Power Plant in Gaffney, SC.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019

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2/6/2012	ML12060A280	Comment (26) of Katherine Beattie Opposing Duke Energy's Combined License Application (COL) to build William States Lee Nuclear Power Plant in Gaffney, SC.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
2/7/2012	ML11171A002	William States Lee III Nuclear Station, Units 1 And 2 Combined License Application - Advanced Safety Evaluation Without Open Items For Chapter 12, "Radiation Protection."	Letter	NRC/NRO/DNRL/NWE1	Duke Energy Carolinas, LLC	05200018 05200019
2/7/2012	ML11171A005	William States Lee III Nuclear Station, Units 1 And 2 Combined License Application - Advanced Safety Evaluation Without Open Items For Chapter 12, "Radiation Protection."	Memoranda	NRC/NRO/DNRL	NRC/ACRS	05200018 05200019
2/7/2012	ML12072A081	Comment (39) of Harry Peterson Opposing the Construction of the William States Lee Nuclear Power Plant.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
2/7/2012	ML12072A082	Comment (40) of Manna J. Peterson on behalf of Self Opposing Duke Energy's Combined License Application to build William States Lee Nuclear Plant in Gaffney, South Carolina.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019

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2/8/2012	ML12032A104	Cherokee County Sheriff's Office Thank You Letter - Lee Nuclear Station DEIS Meetings.	Letter	NRC/NRO/DNRL/EPB1	Cherokee County, SC	05200018 05200019
2/10/2012	ML12044A128	Comment (8) of Ruth Lovinsohn on behalf of Self, Opposing Proposed William States Lee III, Units 1 & 2 License Application to Build Two New Nuclear Plant.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
2/10/2012	ML12058A397	Comment (21) of Ruth Lovinsohn, Opposing Proposed Building of William States Lee III Nuclear Stations - Units 1 & 2.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
2/13/2012	ML12032A228	Summary of the Public Meetings for the Draft Environmental Impact Statement to Support Review of the William States Lee III Nuclear Station Units 1 and 2 Combined Licenses Application.	Memoranda Meeting Briefing Package/Handouts Meeting Summary	NRC/NRO/DNRL/EPB1	NRC/NRO/DNRL/EPB1	05200018 05200019
2/14/2012	ML12047A292	William States Lee III, Units 1 and 2 - AP1000 Combined License Application, Supplemental Response to Request for Additional Information (RAI No. 6183).	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/14/2012	ML12048A670	Comment (16) of William A. Fisk Opposing the Construction of the William States Lee Nuclear Plant.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019

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2/15/2012	ML12052A209	Comment (19) of Cori Knudten, Opposing Draft Environmental Impact Statement for Combined Licenses for William States Lee III Nuclear Station Units 1 and 2: Cover through Chapter 8 (NUREG-2111, Volume 1).	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
2/17/2012	ML12072A084	Comment (42) of an Unknown Individual Opposing the Duke Energy's Combined License Application to build William States Lee Nuclear Plant in Gaffney, SC.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
2/19/2012	ML12062A233	Comment (28) of Lucy D. Christopher Opposing the Proposed William States Lee Nuclear Plant.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
2/21/2012	ML12062A070	Comment (27) of S. Flores on behalf of Self Opposing Duke Energy's Combined License Application to Build William States Lee Nuclear Power Plant in Gaffney, SC.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
2/22/2012	ML12072A083	Comment (41) of Will Leverette Opposing the Combine License Application to Build William States Lee Nuclear Plant in Gaffney South Carolina.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019

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2/23/2012	ML12058A440	William States Lee III Nuclear Station - AP1000 Combined License Application, Supplemental Response to Request for Additional Information (RAI No. 13.06-44).	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/24/2012	ML12072A078	Comment (36) of Debra K. Daily Opposing Proposed Building of William States Lee III Nuclear Stations - Units 1 & 2.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
2/24/2012	ML12072A080	Comment 38 of K. Rustin Opposing the Combined License Application to build William States Lee Nuclear Power Plant in Gaffney, South Carolina.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
2/26/2012	ML12072A079	Comment (37) of Unknown Submitter Opposing Proposed Building of William States Lee III Nuclear Stations - Units 1 & 2.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
2/28/2012	ML12059A492	2012/02/28 Lee RAI for SER - RAI LTR NO 103 RELATED TO SRP 2.2.2 Potential Accidents for the W.S. LEE Units 2 & 3 COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/NWE1	05200018 05200019
2/28/2012	ML12060A377	William States Lee III, Units 1 and 2 - Supplemental Response to Request for Additional Information (RAI No. 2350).	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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2/28/2012	ML12151A383	Comment on Lee Nuclear Station Draft EIS - K. Macko Meeting Feedback Form.	- No Document Type Applies	- No Known Affiliation	NRC/NRO	05200018 05200019
2/29/2012	ML12083A060	Comment (44) Joyce Stanley for Gregory Hogue, on Behalf of US Dept. of the Interior, No Comments on the Combined Licenses for William States Lee III Nuclear Station Units 1 & 2.	Letter	US Dept of Interior, Office of the Secretary	NRC/ADM/DAS	05200018 05200019
3/1/2012	ML12065A195	William States Lee III, Units 1 and 2, AP1000 Combined License Application, Schedule for Future Submittals Related to Combined Operating License Application.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
3/1/2012	ML12067A037	Comment (29) of Christopher M. Fallon on Draft Environmental Impact Statement for Combined Licenses (COLs) for William States Lee III Nuclear Station Units 1 and 2.	General FR Notice Comment Letter	Duke Energy Carolinas, LLC	NRC/ADM/DAS/RDEB	05200018 05200019
3/1/2012	ML12146A266	Comment (50) of Clare Hamahan, on Behalf of Self, Opposing Duke Energy's Combined License Application to Build William States Lee Nuclear Power Plant in Gaffney, SC.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019

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3/2/2012	ML12083A061	Comment (45) of Patricia B. McAfee Opposing the Environmental Combined License Application for the William States Lee III Nuclear Station Site, Units 1 and 2.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
3/3/2012	ML12083A063	Comment (47) of William Brockington and Mary Brockington on behalf of themselves Opposing William States Lee III, Units 1 and 2, Combined License, Proposed Nuclear License Application.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
3/5/2012	ML120760114	M. Caldwell, USFWS, Comments on Lee Nuclear Station Draft Environmental Impact Statement for Combined Licenses.	Letter	US Dept of Interior, Fish & Wildlife Service	NRC/ADM/DAS/RDEB NRC/NRO	05200018 05200019
3/5/2012	ML12083A062	Comment (46) of Ruth Thomas and Ellen Thomas Opposing the Environmental Combined License Application for the William States Lee III Nuclear Station Site, Units 1 and 2.	General FR Notice Comment Letter	Environmentalists, Inc Women's International League for Peace & Freedom	NRC/ADM/DAS/RDEB	05200018 05200019

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3/5/2012	ML12083A064	Comment (48) of Jay B. Herrington, on Behalf of U.S. Dept. of the Interior, Fish and Wild Life Service, on the Environmental Combined License Application for the William States Lee III Nuclear Station Site, Units 1 and 2.	General FR Notice Comment Letter	US Dept of Interior, Fish & Wildlife Service	NRC/ADM/DAS/RDEB	05200018 05200019
3/6/2012	ML12068A363	Comment (31) of Joyce Stanley of the US Department of the Interior, Office of the Secretary on the Combined Licenses for William States Lee III Nuclear Station Units 1 and 2.	General FR Notice Comment Letter	US Dept of Interior, Office of the Secretary US Dept of Interior, Office of Environmental Policy and Compliance	NRC/ADM/DAS/RDEB	05200018 05200019
3/6/2012	ML12068A364	Comment (32) of Ben Gregg on behalf of South Carolina Wildlife Federation, on Environmental Combined License at William States Lee III Nuclear Station Site, Units 1 and 2, Duke Energy Carolinas, LLC Availability of Draft Environmental Impact Statement.	General FR Notice Comment Letter	South Carolina Wildlife Federation	NRC/ADM/DAS/RDEB	05200018 05200019

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3/6/2012	ML12068A407	Comment (33) of Daniel Gamble Opposing the Environmental Combined License Application for the William States Lee III Nuclear Station Site, Units 1 and 2.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
3/6/2012	ML12068A408	Comment (34) of a Concerned Citizen on the Environmental Combined License Application for the William States Lee III Nuclear Station Site, Units 1 and 2.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
3/6/2012	ML12083A059	Comment (43) of Bob Perry on Behalf of South Carolina Department of Natural Resources on Draft Environmental Impact Statement for William States Lee III Combined Licenses, Unit 1 & 2.	General FR Notice Comment Letter	State of SC, Dept of Natural Resources	NRC/ADM/DAS/RDEB	05200018 05200019
3/7/2012	ML11157A018	Lee ASE Ch 18 Cover Letter.	Letter	NRC/NRO/DNRL/NWE1	Duke Energy Carolinas, LLC	05200018 05200019
3/7/2012	ML12072A077	Comment (35) of Brynn Schmitt on behalf of Self Opposing the Proposed Project of William States Lee III Nuclear Station, Units 1 and 2.	General FR Notice Comment Letter	- No Known Affiliation	NRC/ADM/DAS/RDEB	05200018 05200019
3/8/2012	ML11157A020	Lee ASE Ch 18 ACRS Memo.	Memoranda	NRC/NRO/DNRL	NRC/ACRS	05200018 05200019

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3/9/2012	ML12073A345	Duke Energy Corp - Southern Ohio Clean Energy Park, Addition of Name for Service/Distribution Listings for Routine Information.	Letter	Duke Energy Corp	NRC/Document Control Desk NRC/NMSS NRC/NRO	05000269 05000270 05000287 05000369 05000370 05000413 05000414 05200018 05200019 07200004
3/16/2012	ML12076A190	Commission Memorandum and Order CLI-12-07.	Legal-Order	NRC/SECY	NRC/Chairman NRC/OCM	05200018 05200019 05200025 05200026
3/16/2012	ML120790121	EPA Region 4 Comment Letter From H. Mueller on William States Lee III Nuclear Station Units 1 and 2 Draft Environmental Impact Statement.	Letter	US Environmental Protection Agency (EPA)	NRC/ADM/DAS/RDEB	05200018 05200019

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3/16/2012	ML120790165	M120316 - Affirmation: I. SECY-11-0170 Final Rule: Physical Protection of Byproduct Material; II. SECY-12-0014 Comanche Units 3 & 4; Columbia Gen. Station; Vogtle Units 3 & 4); William States Lee III Units 1 & 2, Petitions for Review of LBP-11-27	Commission Meeting Transcript/Exhibit	NRC/OCM		05000397 05200018 05200019 05200025 05200026 05200034 05200035
3/16/2012	ML12089A018	Comment (49) of Heinz J. Mueller of the US Environmental Protection Agency on the Combined Licenses (COLs) Application, Constructing and Operating Two New Nuclear Units at the Lee Nuclear Station Site.	General FR Notice Comment Letter	US Environmental Protection Agency (EPA)	NRC/ADM/DAS/RDEB	05200018 05200019
3/19/2012	ML12080A112	William States Lee III, Units 1 and 2 - AP1000 Combined License Application, Supplemental Partial Response to Request for Additional Information (RAI No. 6182).	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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3/19/2012	ML12080A128	William States Lee III, Units 1 and 2 - AP1000 Combined License Application - Westinghouse Electric Company Report on Site-Specific Analyses.	Legal-Affidavit Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
3/28/2012	ML12090A052	William States Lee III, Units 1 and 2, AP1000 Combined License Application, Response to Request for Additional Information No. 6339.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
3/29/2012	ML12093A005	AP1000 Combined License Application for the William States Lee III Nuclear Station, Units 1 & 2, Supplemental Information to the Environmental Report (Revision 1) Ltr. # WLG2012.03-10.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
3/29/2012	ML12093A006	William States Lee III Nuclear Station Units 1 and 2, AP1000 Combined License Application, Supplemental Response to Request for Additional Information Ltr# WLG2012.03-09.	Legal-Affidavit Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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3/29/2012	ML12093A197	AP1000 Combined License Application for the William States Lee III Nuclear Station Units 1 and 2 Supplemental Response to Request for Additional Information Ltr# WLG2012.03-12.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
3/29/2012	ML12096A077	William States Lee III, Units 1 and 2, AP1000 Combined License Application, Supplemental Response to Request for Additional Information.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
4/5/2012	ML12096A033	2012/04/05 Lee RAI for SER - LEE-RAI-LTR-104 Related to SRP Section 2.03.03 Onsite Met Measurement for the W.S. Lee COLA Units 1 & 2	E-Mail	NRC/NRO	NRC/NRO/DNRL/LB4	05200018 05200019
4/10/2012	ML12143A293	William States Lee III, Units 1 and 2 - AP1000 Combined License Application, Supplemental Response to Request for Additional Information, RAI 28 Supplement, Socioeconomics.	Legal-Affidavit Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
4/11/2012	ML121020431	Lee DEIS - Rocky Shoals Spider Lily Email.	E-Mail	NRC/NRO/DSEA/RENV	NRC/NRO/DNRL/EPB1	05200018 05200019

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4/12/2012	ML12107A118	William States Lee III, Units 1 and 2, AP1000 Combined License Application for 10 CFR 50.46 Annual Report.	Letter Annual Operating Report	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
4/16/2012	ML12109A156	William States Lee III Nuclear Station, Units 1 and 2, AP1000 Combined License Application for the Supplemental Response to Request for Additional Information, Letter No. 64, RAI 09.02.01-6.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
4/16/2012	ML121230315	Update for William States Lee III Nuclear Station, Units 1 and 2 Combined License Application.	Legal-Affidavit Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
4/16/2012	ML12137A370	Duke Energy WSL III Units 1 & 2 COLA (Generic DCD Departures Report) - Part 7, Departures and Exemptions Requests	Generic DCD Departures Report License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
4/16/2012	ML12137A382	Duke Energy WSL III Units 1 & 2 COLA (Emergency Plan) - Emergency Plan	License-Application for Combined License (COLA) Emergency Preparedness-Emergency Plan	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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4/16/2012	ML12137A384	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report) - FSAR Master Table of Contents	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
4/16/2012	ML12137A385	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report) - FSAR Chapter 01 Introduction and General Description of the Plant - Sections 01.01 - 01.10, Appendices 1A, 1AA, 1B	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
4/16/2012	ML12137A386	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report) - FSAR Chapter 01 Figure 1.1-201	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
4/16/2012	ML12137A387	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report) - FSAR Chapter 01 Figure 1.1-202	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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4/16/2012	ML12137A388	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report) - FSAR Chapter 01 Figure 1.2-201	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
4/16/2012	ML12137A389	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report) - FSAR Appendix 2AA, Attachments	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
4/16/2012	ML12137A390	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report) - FSAR Appendix 2BB, Attachments	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
4/16/2012	ML12137A391	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report) - FSAR Appendix 2CC, Attachments	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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4/16/2012	ML12137A841	Duke Energy WSL III Units 1 & 2 COLA (ITAAC) - Part 10, Conditions and ITAAC	Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
4/16/2012	ML12137A843	Duke Energy WSL III Units 1 & 2 COLA (Technical Specifications) - Part 4, Williams Lee III Nuclear Station Technical Specifications	Technical Specifications License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
4/16/2012	ML121230315	Update for William States Lee III Nuclear Station, Units 1 and 2 Combined License Application.	Letter, COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
4/16/2012	ML12137A370	Duke Energy WSL III Units 1 & 2 COLA (Generic DCD Departures Report) - Part 7, Departures and Exemptions Requests	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
4/16/2012	ML12137A382	Duke Energy WSL III Units 1 & 2 COLA (Emergency Plan) - Emergency Plan	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019

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4/25/2012	ML12116A336	2012/04/25 Lee RAI for SER - REQUEST FOR ADDITIONAL INFORMATION LTR 105 CONCERNING IMPLEMENTATION OF FUKUSHIMA TASK FORCE RECOMMENDATIONS FOR LEE UNITS 1 & 2 COL APPLICATION	E-Mail Request for Additional Information (RAI)	NRC/NRO	NRC/NRO/DNRL/LB4	05200018 05200019
4/30/2012	ML121220294	William States Lee III Nuclear Station, Units 1 and 2, Submittal of Information Supporting Update Roadmap for Combined License Application, Ltr WLG2012.04-07.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
4/30/2012	ML12123A712	William States Lee III Nuclear Station, AP1000 Combined License Application for the William States Lee III Nuclear Station Units 1 and 2 Supplemental Response to Request for Additional Information Ltr# WLG2012.04-06.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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4/30/2012	ML12123A714	William States Lee III, Units 1 and 2, AP1000 Combined License Application, Supplemental Response to Request for Additional Information, RAI 190, Site Layout and Plant Description and RAI 210, Ecology, Aquatic.	Legal-Affidavit Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
4/30/2012	ML12123A715	William States Lee III Nuclear Station Units 1 and 2 Supplemental Response to Request for Additional Information, Ltr# WLG2012.04-05.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
5/1/2012	ML12117A381	05/22/2012-Notice of Forthcoming Public Meeting w/Duke Energy Carolinas, LLC Re William States Lee III Units 1&2 COL Application & Request For Information Related To Seismic Reevaluation In Response To Lessons Learned From Fukushima Earthquake & Tsunami.	Meeting Notice Meeting Agenda Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019

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5/1/2012	ML12137A319	05/22/2012 - Cancellation Notice of Forthcoming Public Meeting w/Duke Energy Carolinas, LLC Re: William States Lee III Units 1&2 COL Application and RAI Related to Seismic Reevaluation in Response to Lessons Learned From Fukushima Earthquake and Tsunami.	Meeting Notice Meeting Agenda Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019
5/2/2012	ML12124A282	William States Lee III, Units 1 and 2, AP1000 Combined License Application for Response to Request for Additional Information Items 02.03.03-004 and 02.03.03-005.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
5/2/2012	ML12124A332	William States Lee III, Units 1 and 2 - AP1000 Combined License Application, Supplemental Response to Request for Additional Information (RAI No. 6183).	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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5/9/2012	ML12124A140	04/17/2012 Summary of Category 2 Meeting with the AP1000 Design-Centered Working Group (DWCG) and the General Public to Discuss the Completion and Inspection Strategy for the Design Acceptance Criteria (DAC) Associated with AP1000 Human Factors Design.	Meeting Summary Memoranda	NRC/NRO/DCIP/CIPB	NRC/NRO/DCIP/CIPB	05200014 05200015 05200018 05200019 05200022 05200023 05200025 05200026 05200027 05200028 05200029 05200030 05200040 05200041
5/16/2012	ML12137A836	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report) Revision 5	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019

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5/17/2012	ML12137A203	04/09/2012-NRC Audit Report For The William States Lee III COL Application Review Related To Structure Seismic Interaction (SSI).	Memoranda Project Plans and Schedules Audit Report	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019
5/17/2012	ML12138A118	2012/05/17 Lee RAI for SER - LEE-RAI-LTR-106.docx	E-Mail	NRC/NRO	NRC/NRO/DNRL/LB4	05200018 05200019
5/18/2012	ML12132A218	Summary Of William States Lee III Nuclear Station Units 1 and 2 Section 404 Joint Permit Application Mitigation Sites Visit.	Meeting Summary Memoranda	NRC/NRO/DNRL/EPB1	NRC/NRO/DNRL/EPB1	05200018 05200019
5/21/2012	ML12171A581	William States Lee Nuclear Station - Consultation Requirements E-mail to National Marine Fisheries Service.	E-Mail	US Dept of Commerce, National Marine Fisheries Service	NRC/NRO/DNRL/EPB1	05200018 05200019
5/23/2012	ML12144A056	2012/05/23 Lee RAI for SER - LEE-RAI-LTR-107.docx	E-Mail	NRC/NRO	NRC/NRO/DNRL/LB4	05200018 05200019
5/23/2012	ML12144A179	06/06/2012 Notice of Forthcoming Public Meeting To Discuss Licensing Actions Related To Duke Energy Carolinas, LLC, COLs For William States Lee, Units 1 and 2.	Meeting Notice Meeting Agenda Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019

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Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
5/24/2012	ML12151A110	William States Lee III, Units 1 and 2, AP1000 Combined License Application, Partial Response to Request for Additional Information (RAI No. 6419).	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
5/30/2012	ML12151A337	Lee Nuclear Station Reference - E-mail from J. Holling, SCDNR, Regarding State Species Status.	E-Mail	State of SC, Dept of Natural Resources	- No Known Affiliation NRC/NRO	05200018 05200019
5/31/2012	ML12156A212	William States Lee III, Units 1 and 2 - AP1000 Combined License Application, Response to Request for Additional Information.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
6/5/2012	ML12158A476	AP1000 Combined License Application for the William States Lee III Nuclear Station Units 1 and 2 Supplemental Response to Request for Additional Information (RAI No. 6182) Ltr# WLG2012.06-05.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
6/6/2012	ML12158A270	06/06/2012-William States Lee Meeting Slides.	Meeting Briefing Package/Handouts Slides and Viewgraphs	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019

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6/11/2012	ML12165A422	AP1000 Combined License Application for the William States Lee III Nuclear Station Units 1 and 2, Schedule for Future Submittals Related to the William State Lee III Station Combined Operating License Application.	Letter	Duke Energy Corp	NRC/Document Control Desk NRC/NRO	05200018 05200019
6/11/2012	ML12165A423	William States Lee III, Units 1 and 2 - AP1000 Combined License Application, Supplemental Response to Request for Additional Information (eRAI No. 6419).	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
6/12/2012	ML12166A288	William States Lee III, Units 1 and 2, AP1000 Combined License Application for Supplemental Response to Request for Additional Information (RAI) Letter No 83, RAI Nos. 13.03-77 and 13.03-87.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
6/13/2012	ML12173A053	Comment (51) of Jay B. Herrington, on Behalf of US Fish and Wildlife Service, Supporting Duke Energy's Combined License Application to Build William States Lee Nuclear Power Plant in Gaffney, SC.	General FR Notice Comment Letter	US Dept of Interior, Fish & Wildlife Service	NRC/ADM/DAS/RDEB	05200018 05200019

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6/18/2012	ML12170B124	Petition to Suspend Final Decisions in all Pending Reactor Licensing Proceedings Pending Completion of Remanded Waste Confidence Proceedings.	Legal-Pleading	Blue Ridge Environmental Defense League Harmon, Curran, Spielberg & Eisenberg, LLP	NRC/OCM	05200018 05200019
6/18/2012	ML12171A367	William States Lee III, Units 1 and 2 - Response to Request for Additional Information (eRAI 6497) Item 02.05.04-017.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
6/18/2012	ML12172A396	Petition to Suspend Final Decisions in All Pending Reactor Licensing Proceedings from Friends of the Earth.	Legal-Intervention Petition, Responses and Contentions	Friends of the Earth	NRC/SECY	05000275 05000323 05000346 05000391 05000416 05000443 05000483 05000498 05000499 05200012 05200013 05200014 05200015 05200016 05200017 05200018 05200019

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						05200022 05200023 05200024 05200029 05200030 05200033 05200034 05200035 05200039 05200040 05200041

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6/19/2012	ML12171A624	Waste Confidence Secy Order - Order of Acting Secretary to the Parties Inviting Answers to Suspend Final Licensing Decisions by 12 PM EDT, Monday, June 25, 2012.	Legal-Order	NRC/SECY	Duke Energy Carolinas, LLC NRC/OGC	05200018 05200019
6/20/2012	ML12174A272	AP1000 Combined License Application for the William States Lee III Nuclear Station Units 1 and 2, Response to Request for Additional Information (eRAI 6528).	Letter Drawing Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
6/21/2012	ML12172A364	06/06/2012 - Summary Of A Public Meeting And Conference Call With To Duke Energy Carolinas, LLC Regarding The Combined Licenses Application For William States Lee III Units 1 And 2.	Meeting Summary Meeting Agenda Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019
6/21/2012	ML12173A293	Lee Nuclear Station Environmental Review - Emails with FERC Regarding Ninety-Nine Islands Hydroelectric License Article 402 Amendment.	E-Mail	NRC/NRO/DNRL/EPB1	US Federal Energy Regulatory Commission	05200018 05200019

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6/21/2012	ML12173A301	March 7, 2012 Call Record with Federal Energy Regulatory Commission Regarding Amended Article 402 of Ninety-Nine Islands Hydroelectric License.	Note to File incl Telcon Record, Verbal Comm	NRC/NRO/DNRL/EPB1	US Federal Energy Regulatory Commission	05200018 05200019
6/21/2012	ML12178A450	William States Lee III, Units 1 and 2 - AP1000 Combined License Application, Supplemental Information to the Environmental Report (Revision 1).	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
6/25/2012	ML12177A116	Duke Energy's Answer Opposing Petition To Suspend Final Licensing Decisions.	Legal-Pleading	Duke Energy Carolinas, LLC Pillsbury, Winthrop, Shaw, Pittman, LLP	NRC/OCM	05200018 05200019
6/25/2012	ML12177A121	NRC Staff's Answer to Petition To Suspend Final Decisions In All Pending Reactor Licensing Proceedings Pending Completion Of Remanded Waste Confidence Proceedings.	Legal-Pleading	NRC/OGC	NRC/OCM	05200018 05200019
6/27/2012	ML12188A033	William States Lee III, Units 1 and 2, Supplemental Response to Request for Additional Information (eRAI 5507).	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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6/27/2012	ML12188A035	William States Lee III Nuclear Station Units 1 and 2, Supplemental Response to Request for Additional Information. Ltr# WLG2012.06-10.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
6/27/2012	ML12193A689	Declarations of Standing for Reopen 1-22.	Legal-Correspondence/Miscellaneous	Blue Ridge Environmental Defense League	NRC/OCM	05200018 05200019
6/27/2012	ML12194A317	Declarations of Standing for Reopen 23-43.	Legal-Correspondence/Miscellaneous	Blue Ridge Environmental Defense League	NRC/OCM	05200018 05200019
7/5/2012	ML12191A040	William States Lee III, Units 1 and 2 - AP1 000 Combined License Application, Response to Request for Additional Information (eRAI 2563).	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
7/6/2012	ML12250A611	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report) Revision 6	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
7/9/2012	ML12192A000	Motion to Reopen the Record for William State Lee III Units 1 and 2.	Legal-Motion	Blue Ridge Environmental Defense League	NRC/SECY	05200018 05200019
7/10/2012	ML12192A134	Letter to Commission Secretary re: Intervenors Filings in Dockets 52-018, 52-019 and 52-017.	Legal-Correspondence/Miscellaneous	Blue Ridge Environmental Defense League	NRC/SECY	05200018 05200019

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
7/12/2012	ML12198A011	William States Lee III, Units 1 and 2, AP1000 Combined License Application, 10 CFR 50.46 Thirty Day Report Regarding "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors."	Letter Licensee 30-Day Written Event Report	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
7/12/2012	ML12198A014	William States Lee III, Units 1 and 2, AP1000 Combined License Application, Supplemental Response to Request for Additional Information (RAI No. 2680).	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
7/19/2012	ML13221A166	Duke 2012 July 19 Update to SCDHEC on NPDES Permit Application.	Letter	Duke Energy Carolinas, LLC	NRC/NRO State of SC, Dept of Health & Environmental Control	05200018 05200019
7/26/2012	ML111640667	William States Lee III Nuclear Station, Units 1 and 2 COL Application - ASE Without Open Items For Chapter 10, "Steam and Power Conversion System."	Letter	NRC/NRO/DNRL/NWE1	Duke Energy Carolinas, LLC	05200018 05200019
7/26/2012	ML12142A131	Rev. 1 - Letter - William States Lee III Nuclear Station, Units 1 and 2 ASE Without Open Items For	Letter	NRC/NRO/DNRL/LB4	Duke Energy Carolinas, LLC	05200018 05200019

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
		Chapter 18, "Human Factors Engineering".				
7/26/2012	ML12142A136	Rev. 1 - William States Lee III Nuclear Station, Units 1 and 2 ASE Without Open Items for Chapter 18, "Human Factors Engineering."	Safety Evaluation	NRC/NRO/DNRL/LB4		05200018 05200019
7/27/2012	ML11147A122	William States Lee III Nuclear Station, Units 1 and 2 Combined License Application - Advanced Safety Evaluation Without Open Items For Chapter 8, "Electric Power."	Letter	NRC/NRO/DNRL/NWE1	Duke Energy Carolinas, LLC	05200018 05200019
7/27/2012	ML11147A124	Memorandum - William States Lee III Nuclear Station, Units 1 and 2 Combined License Application - Advanced Safety Evaluation Without Open Items For Chapter 8, "Electric Power."	Memoranda	NRC/NRO/DNRL	NRC/ACRS	05200018 05200019
7/27/2012	ML111640670	William States Lee III Nuclear Station, Units 1 and 2 Advanced Safety Evaluation without Open Items for Chapter 10, "Steam and Power Conversion".	Memoranda	NRC/NRO/DNRL/NWE1	NRC/ACRS	05200018 05200019

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7/27/2012	ML12142A121	Rev. 1 - Memorandum To ACRS: William States Lee Nuclear Station, Units 1 and 2 ASE without Open Items For Chapter 18, "Human Factors Engineering."	Memoranda	NRC/NRO/DNRL	NRC/ACRS	05200018 05200019
7/31/2012	ML12215A148	William States Lee III, Units 1 and 2, Supplemental Response to Request for Additional Information (RAI No. 2350) Ltr# WLG2012.07-04.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
7/31/2012	ML12215A149	Lee Nuclear Station Supplemental Response to Request for Additional Information Letter No. 067.	Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
8/2/2012	ML12215A371	NRC Staff'S Response to Bredl's Motion to Reopen the Record and Motion for Leave to File a New Contention Concerning Temporary Storage and Ultimate Disposal of Nuclear Waste at William States Lee III, Units 1 & 2.	Legal-Pleading	NRC/OGC	NRC/OCM	05200018 05200019
8/3/2012	ML12216A213	Duke Energy's Answer Opposing BREDL's Motion to Reopen the Record and Admission of Waste Confidence Contention.	Legal-Pleading	Duke Energy Carolinas, LLC Duke Energy Corp Pillsbury,	NRC/OCM	05200018 05200019

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				Winthrop, Shaw, Pittman, LLP		
8/3/2012	ML12256A896	Duke Energy Responses to 401 Permit Application Comments August 3 2012.	Letter	Duke Energy Carolinas, LLC	State of SC, Dept of Health & Environmental Control NRC/NRO	05200018 05200019
8/7/2012	ML12220A094	Commission Memorandum and Order (CLI-12-16), Granting Requests in Part and Deny the Requests in Part.	Legal-Order	NRC/SECY		05200018 05200019
8/7/2012	ML12221A396	William States Lee III Nuclear Station Units 1 and 2 - AP1000 Combined License Application for the Supplemental Response to Request for Additional Information.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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8/14/2012	ML12173A383	Endangered Species Act, Magnuson-Stevens Fishery Conservation and Management Act, and Fish and Wildlife Coordination Act Consultation Close Out For The William States Lee III Nuclear Station, Units 1 and 2 Combined Licenses Application Environmental.	Letter	NRC/NRO/DNRL/EPB1	State of FL, National Marine Fisheries Services	05200018 05200019
8/16/2012	ML12228A465	07/25-26/2012 Summary of Category 2 Meeting with Southern Nuclear, South Carolina Electric & Gas, Westinghouse & General Public to Discuss and Walk Through the Inspection Process and Procedures for the AP1000 Piping Systems Design Acceptance Criteria.	Meeting Summary Memoranda	NRC/NRO/DCIP/CIPB	NRC/NRO/DCIP/CIPB	05200014 05200015 05200018 05200019 05200022 05200023 05200025 05200026 05200028 05200029 05200030 05200040

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						05200041 05200027
8/21/2012	ML12237A320	Catawba, Units 1 & 2, McGuire, Units 1 & 2, Oconee, Units 1, 2 & 3, and ISFSI, William States Lee III, Units 1 & 2, and Southern Ohio Clean Energy Park, Submittal of Service/Distribution Listings for Routine Information, OUO and Safeguards Information.	Letter	Duke Energy Carolinas, LLC Duke Energy Corp	NRC/Document Control Desk NRC/NRR NRC/NMSS NRC/NRO	05000369 05000370 05000413 05000414 05200018 05200019 07200004 05000269 05000270

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						05000287
8/30/2012	ML12250A074	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report) - FSAR Master Table of Contents	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
8/30/2012	ML12250A075	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report) - FSAR Chapter 01 Introduction and General Description of the Plant - Sections 01.01 - 01.10, Appendices 1A, 1AA, 1B	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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8/30/2012	ML12250A076	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report) - FSAR Chapter 01 Figure 1.1-201	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
8/30/2012	ML12250A605	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report) - FSAR Chapter 15 Accident Analysis - Sections 15.00 - 15.08, Appendices 15A - 15B	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
8/30/2012	ML12255A448	Duke Submission to SCDHEC Regarding Comments on the Lee Nuclear Station Section 401 Permit Request.	Letter	Duke Energy Carolinas, LLC	NRC/NRO State of SC, Dept of Health & Environmental Control	05200018 05200019
9/3/2012	ML13247A005	USFA 2009.	- No Document Type Applies	NRC/NRO		05200018 05200019
9/19/2012	ML12265A066	William States Lee III Nuclear Station Units 1 and 2, AP 1000 Combined License Application for the AP 1000 Combined License Application for the Ltr# WLG2012.09-01.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
9/26/2012	ML13221A013	William States Lee III, Submittal of Response to Request for Additional Information to SCDHEC 401 Water Quality Certification.	Letter	Duke Energy Carolinas, LLC Duke Energy Corp	NRC/FSME State of SC, Dept of Health & Environmental Control	05200018 05200019
9/27/2012	ML12271A505	2012/09/27 Lee RAI for SER - REQUEST FOR ADDITIONAL INFORMATION LTR. No: 108 RELATED TO SRP: 08.03 STABILITY OF OFFSITE POWER SYSTEM FOR THE WILLIAMS STATES LEE III UNITS 1 AND 2 COMBINED LICENSE APPLICATION	E-Mail Request for Additional Information (RAI)	NRC/NRO	NRC/NRO/DNRL/LB4	05200018 05200019
9/27/2012	ML12271A506	2012/09/27 Lee RAI for SER - REQUEST FOR ADDITIONAL INFORMATION LTR No. 108 RELATED TO SRP 08.03 STABILITY OF OFFSITE POWER SYSTEM FOR THE WILLIAM STATES LEE III UNITS 1 AND 2 COMBINED LICENSE APPLICATION	E-Mail	NRC/NRO	NRC/NRO/DNRL/LB4	05200018 05200019
10/3/2012	ML12279A105	William States Lee III, Units 1 and 2 - AP1000 Combined License Application, 2012 Integrated Resource Plan.	Annual Report Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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10/6/2012	ML12280A014	William States Lee Nuclear Station - E-mail to R. Wylie Regarding Ponds A and B Drawdown.	E-Mail	NRC/NRO/DNRL/EPB1	Duke Energy Carolinas, LLC	05200018 05200019
10/17/2012	ML12293A238	AP1000 Combined License Application for the William States Lee III Nuclear Station Units 1 & 2, to RAI, Updated Schedule for Future Submittals Related to William States Lee III Station Combined Operating License...	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
10/22/2012	ML13247A007	Letter from R.L. Darden to W.G. Haire, dated October 22, 2012, regarding the Cultural Resources Management Plan and Memorandum of Agreement for William States Lee Nuclear Station, Units 1 and 2.	Letter	US Dept of the Army, Corps of Engineers, Charleston District	Catawba County, NC NRC/NRO	05200018 05200019
10/23/2012	ML12299A185	William States Lee III, Units 1 and 2, AP1000 Combined License Application, Response to Request for Additional Information (eRAI 6751).	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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10/23/2012	ML13252A254	10/23/2012 - Ltr from: R. Kitchen, Duke Energy Carolinas, LLC to R.L. Darden, USACE re: William S. Lee Pond A and B Drawdowns (NPDES) Letter# WLG2012 10-02.	Letter	Duke Energy Carolinas, LLC	NRC/NRO US Dept of the Army, Corps of Engineers, Charleston District	05200018 05200019
10/31/2012	ML13214A349	Attachment - Species & Communities Known to Occur Within 15 Miles of Lee Nuclear Station October 31, 2012 Received July 31, 2013.	Graphics incl Charts and Tables	Pacific Northwest National Lab	State of SC, Dept of Natural Resources NRC/NRO	05200018 05200019
10/31/2012	ML13221A562	Email dtd Oct 31, 2012, from R.L. Darden, USACE, to Corey Gray (to transmit an attachment).	E-Mail	US Dept of the Army, Corps of Engineers, Charleston District	NRC/NRO Atkins Global	05200018 05200019
10/31/2012	ML13221A569	Additional USACE Questions for Duke dated Oct 31, 2012. (Pertaining to William S. Lee).	Request for Additional Information (RAI)	US Dept of the Army, Corps of Engineers, Charleston District	Duke Energy Carolinas, LLC NRC/NRO	05200018 05200019
11/20/2012	ML13214A350	E-Mail dtd 11/20/2012 From: J. Holling, SCDNR, To: J.M. Becker, PNNL re: Lee Nuclear Request of May 25, 2012. Regarding W.S. Lee Nuclear Plant - Species Within a 15-mile Radius of Plant.	E-Mail	State of SC, Dept of Natural Resources	NRC/NRO Pacific Northwest National Lab	05200018 05200019

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12/3/2012	ML13221A404	Email transmitting Duke letter and attachments to USACE dated December 3, 2012 regarding 404 Application and Jurisdictional Determination.	E-Mail	NRC/NRO/DNRL/EPB1	NRC/NRO/DSEA/RENV	05200018 05200019
12/3/2012	ML13221A412	12/02/2012 Letter from R. Wylie, Duke Energy, to R.L. Darden re: Duke letter and attachments to USACE dated December 3, 2012, regarding 404 Application and Jurisdictional Determination.	Letter	Duke Energy Carolinas, LLC	US Dept of the Army, Corps of Engineers, Charleston District NRC/NRO	05200018 05200019
12/6/2012	ML13221A573	USACE Email Message November 1, 2012, from T.P. Eucker to J.M. Becker.	E-Mail	US Dept of the Army, Corps of Engineers, Charleston District	NRC/NRO US Dept of the Army, Corps of Engineers, Charleston District	05200018 05200019
12/11/2012	ML12345A291	01/03/2013 - Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center Combined License Review Issues.	Meeting Notice Meeting Agenda Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200027 05200028 05200029

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						05200030 05200040 05200041
12/11/2012	ML12345A294	01/10/2013 - Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center Combined License Review Issues.	Meeting Notice	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200027 05200028 05200029 05200030 05200040 05200041

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12/11/2012	ML12345A310	01/17/2013 - Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center Combined License Review Issues.	Meeting Notice Meeting Agenda Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200027 05200028 05200029 05200030 05200040 05200041
12/11/2012	ML12345A317	01/24/2013 - Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center Combined License Review Issues.	Meeting Notice Meeting Agenda Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200027 05200028 05200029 05200030

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						05200040 05200041
12/11/2012	ML12345A335	01/31/2013 - Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center Combined License Review Issues.	Meeting Notice Meeting Agenda Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200027 05200028 05200029 05200030 05200040 05200041

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12/19/2012	ML13221A500	Mixing Zone Request - William States Lee III Nuclear Station NPDES Permit - GA100195 NPDES Supplement Final (December 2012) Issue 3.	Environmental Report Amendment	GeoSyntec Consultants MMI Engineering, LTD	State of SC, Dept of Environmental Control, Bureau of Water NRC/NRO Duke Energy Carolinas, LLC	05200018 05200019
12/19/2012	ML13221A516	Mixing Zone Figures - Final (December 2012) Issue 3 - Computer Model Figures.	Environmental Report Amendment Map	GeoSyntec Consultants MMI Engineering, LTD	State of SC, Dept of Environmental Control, Bureau of Water NRC/NRO Duke Energy Carolinas, LLC	05200018 05200019
12/20/2012	ML12361A057	William States Lee III, Units 1 and 2, AP1000 Combined License Application for Supplemental Information Related to Design Changes to the Lee Units 1 and 2 Physical Locations.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
12/20/2012	ML12361A058	William States Lee III, Units 1 and 2, Relocation Description, Scope of Changes, and FSAR Impacts Due to Plant Relocation, Page 1 of 203 through Page 101 of 203.	- No Document Type Applies	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
12/20/2012	ML12361A060	William States Lee III, Units 1 and 2, Relocation Description, Scope of Changes, and FSAR Impacts Due to Plant Relocation, Page 102 of 203 through Page 203 of 203.	- No Document Type Applies	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
12/20/2012	ML12361A061	William States Lee III, Units 1 and 2, Enclosure 2, Preliminary Assessment - Evaluation of 2012 Field Investigation Results for Plant Relocation.	Report, Miscellaneous	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
1/9/2013	ML13008A513	01/10/2013-Cancelled Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center Combined License Review Issues.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200027 05200028 05200029

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						05200030 05200040 05200041
1/10/2013	ML13008A751	01-03-13-Cancelled Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center COL Review Issues.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200020 05200023 05200027 05200028 05200029 05200030 05200040 05200041

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1/10/2013	ML13213A399	Transmittal and Cultural Resource Management Plan and Agreement regarding William States Lee III Nuclear Station, Units 1 And 2 and Transmission Lines Combined Licenses Application	E-Mail Legal-Memorandum of Agreement/Understanding	US Dept of the Army, Corps of Engineers	- No Known Affiliation NRC/NRO/DNRL	05200018 05200019
1/10/2013	ML13219A947	E-mail With South Carolina DNR Concurrence With Proposed Work Plan For Drawdown Of Ponds A and B as Described in Correspondence From Duke Energy to the USACE dated October 23, 2012.	E-Mail	US Dept of the Army, Corps of Engineers, Charleston District	NRC/NRO	05200018 05200019
1/10/2013	ML13221A438	01/10/2013 Cover Letter from R. Wylie, Duke Energy, to R. Thompson, SCDHEC, re: SCDHEC NPDES Permit Application on Mixing Zone Report for Lee.	Letter	Duke Energy Carolinas, LLC	NRC/NRO State of SC, Dept of Health & Environmental Control	05200018 05200019
1/11/2013	ML13221A019	01/11/2013 Letter to Robert Wylie, Duke Energy From Richard L. Darden, USACE re: USACE Jurisdictional Determination (JD) Letter and Attachments.	Letter	US Dept of the Army, Corps of Engineers, Charleston District	Duke Energy Carolinas, LLC NRC/NRO	05200018 05200019

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1/11/2013	ML13221A024	01/11/2013 Letter to Robert Wylie, Duke Energy From Richard L. Darden, USACE re: USACE Jurisdictional Determination (JD) - Attachment Only with JD Numbers.	Graphics incl Charts and Tables	US Dept of the Army, Corps of Engineers, Charleston District	Duke Energy Carolinas, LLC NRC/NRO	05200018 05200019
1/15/2013	ML12332A248	Acknowledgment of Receipt of the Combined License Application Revision For Relocating The William States Lee III Units 1 and 2 Nuclear Islands.	Letter	NRC/NRO/DNRL/LB4	Duke Energy Carolinas, LLC	05200018 05200019
1/16/2013	ML13016A203	01/17/2013-CANCELLED-Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center COL Review Issues.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200027 05200028 05200029 05200030 05200040 05200041

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1/17/2013	ML13016A468	02/07/2013-Notice of Forthcoming Public Conference To Discuss William States Lee III Units 1 and 2 COL Review Issues.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019
1/22/2013	ML13017A194	02/07/2013-Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center COL Review Issues.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200027 05200028 05200029 05200030 05200040 05200041

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1/22/2013	ML13017A251	02/21/2013-Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center COL Review Issues	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200027 05200028 05200029 05200030 05200040 05200041
1/23/2013	ML13224A033	Attachment with January 23, 2013 Duke Email and Appendix A Pages 11-12 re: NPDES Update.	E-Mail	Duke Energy Carolinas, LLC	US Dept of the Army, Corps of Engineers, Charleston District NRC/NRO	05200018 05200019

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1/28/2013	ML13023A343	01/24/13-CANCELLED-Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center COL Review Issues.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200027 05200028 05200029 05200030 05200040 05200041
1/30/2013	ML13030A311	01/31/2013-CANCELLED-Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center Combined License Review Issues.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200027 05200028 05200029 05200030

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						05200040 05200041
1/31/2013	ML13224A007	01/31/2013 Letter from T.B. Hadden, USACE, to R. Wylie, Duke Energy Carolinas Subject: 404(b)(1) letter and Public Notice #SAC 2009-122-SIR	Letter	US Dept of the Army, Corps of Engineers, Charleston District	Duke Energy Carolinas, LLC	05200018 05200019
2/6/2013	ML13036A167	02/07/2013 - Cancelled 02/05/2013 Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center COL Review Issues.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200027 05200028 05200029 05200030 05200040

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
						05200041
2/13/2013	ML13044A310	03/07/2013-Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center Combined License Review Issues.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200027 05200028 05200029 05200030 05200040 05200041

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2/13/2013	ML13044A319	03/21/2013 Meeting Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center Combined License Review Issues.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200027 05200028 05200029 05200030 05200040 05200041
2/18/2013	ML13050A650	William States Lee III Nuclear Station Units 1 and 2, AP1000 Combined License Application for the Proposed Changes in Response to the Final Rule on Enhancements to Emergency Preparedness Regulation (76 FR 72560) Ltr# WLG2013.02-01.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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2/18/2013	ML13050A651	Enclosures 2 through 6: Markup of Lee COLA Part 2, Chapters 1 and 13, Part 10, Roadmap/Change Matrix and Summary of Regulatory Compliance.	Emergency Preparedness-Emergency Plan Final Safety Analysis Report (FSAR)	Duke Energy Corp	NRC/NRR	05200018 05200019
2/18/2013	ML13050A652	Enclosure 1, Markup of Lee Nuclear Station Emergency Plan, Rev. 4.	Emergency Preparedness-Emergency Plan Final Safety Analysis Report (FSAR)	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
2/22/2013	ML13057A017	William States Lee III Nuclear Station, Units 1 and 2 - AP1000 Combined License Application, Updated Schedule for Future Submittals to Combined Operating License Application.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/25/2013	ML13058A051	William States Lee III, Units 1 and 2, AP1000 Combined License Application, Departure Report Update.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
3/5/2013	ML13058A628	02/07/2013-Meeting Summary of Meeting with Duke Energy Regarding Lee Nuclear Islands Relocation.	Meeting Summary	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019

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3/5/2013	ML13058A646	02/07/2013 Lee Nuclear Islands Relocation Presentation.	Meeting Briefing Package/Handouts Slides and Viewgraphs	Duke Energy Carolinas, LLC	NRC/NRO/DNRL/LB4	05200018 05200019
3/13/2013	ML13087A299	William States Lee III, Units 1 and 2, AP1000 Combined License Application, Supplemental Information Regarding Environmental Review.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
3/14/2013	ML13072B428	03/27/2013-Public Meeting Notice For Levy Units 1 and 2 Containment Design Change.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200025 05200026 05200027 05200028 05200029 05200030 05200040

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
						05200041
3/22/2013	ML13087A201	William States Lee III, Units 1 and 2, AP1000 Combined License Application, Supplemental Information Regarding Environmental Review.	Legal-Affidavit Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
3/22/2013	ML13224A274	03/22/2013 Letter from Timikia Shafeek-Horton, Duke Energy, to Jocelyn Boyd, Public Service Commission of South Carolina regarding Retirement Dates for Certain Units; 2012 Integrated Resource Plan.	Letter	Duke Energy Carolinas, LLC Progress Energy Carolinas, Inc	State of SC, Public Service Commission NRC/NRO	05200018 05200019
3/31/2013	ML13087A203	Enclosure 1 - Brockington 2013 Archaeological Survey of Proposed Grading and Spoil Areas, W.S. Lee	Environmental Report	Brockington & Associates, Inc	NRC/NRO Duke Energy Carolinas, LLC	05200018 05200019

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		Nuclear Station (WLS) March 2013.				
4/3/2013	ML13220A505	04/03/2013 - Letter from South Carolina State History Preservation Office to U.S. Army Corps of Engineers Regarding South Carolina SHPO letter to USACE on Archaeological sites 38CK0185, 38CK0186, 38CK0187, and 38CK0188.	Letter	State of SC, Dept of Archives and History	US Dept of the Army, Corps of Engineers, Charleston District NRC/NRO	05200018 05200019
4/9/2013	ML13093A353	04/18/2013 - Revised - Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center COL Review Issues.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200029 05200030 05200040 05200041

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4/9/2013	ML13098A056	05/02/2013-Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center Combined License Review Issues.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200029 05200030 05200040 05200041
4/9/2013	ML13098A060	05/16/2013-Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center Combined License Review Issues.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200029 05200030 05200040 05200041

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
4/9/2013	ML13098A083	05/30/2013-Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center Combined License Review Issues.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200029 05200030 05200040 05200041
4/10/2013	ML13109A044	William States Lee III Nuclear Station, Units 1 and 2 - AP1000 Combined License Application.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
4/11/2013	ML13100A313	05/01/2013 Notice of Forthcoming Public Meeting to Discuss Levy Units 1 and 2 Combined License Application Request for Information Related to Bulletin 2012-01.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200025 05200026 05200027 05200028

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						05200029 05200030 05200040 05200041
4/12/2013	ML13098B018	03/27/2013-Summary of Public Meeting - Levy Nuclear Plant, Units 1 and 2 COL Application-Containment Building Design Change.	Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200025 05200026 05200027 05200028 05200029 05200030 05200040

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						05200041
4/12/2013	ML13100A302	04/04/2013-Summary of Public Teleconference Meeting Regarding Levy Nuclear Plant, Units 1 and 2 Combined License Application.	Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200029 05200030 05200040 05200041

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4/17/2013	ML13107B407	Revised-05/01/2013-Notice of Forthcoming Public Meeting to Discuss Approach to Address Electrical System Vulnerability Related to Bulletin 2012-01 for AP1000 Combined License Applicants and Licensees.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200025 05200026 05200027 05200028 05200029 05200030 05200040 05200041
4/22/2013	ML13108A337	CANCELLED-04/18/2013-Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center Combined License Review Issues.	Meeting Notice	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200029 05200030

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						05200040 05200041
4/24/2013	ML13120A014	William States Lee III Nuclear Station, Units 1 and 2 - AP1000 Combined License Application for the 10 CFR 50.46 Annual Report.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
5/2/2013	ML13122A458	05/16/2013-Public Meeting Notice for Duke Energy's Seismic Review for William States Lee III Units 1 and 2 COL Application.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019
5/2/2013	ML13127A224	AP1000 Combined License Application for the William States Lee III Nuclear Station Units 1 and 2 Supplemental Information Related to Design Changes to the Lee Units 1 and 2 Physical Locations and Additional Design Enhancements Ltr#: WLG2013.05-02.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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5/2/2013	ML13127A225	Lee Nuclear Station, Units 1 and 2 - Description, Scope of Changes, and FSAR Impacts due to Plant Relocation and Additional Design Enhancements.	Updated Final Safety Analysis Report (UFSAR)	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
5/2/2013	ML13127A226	Lee Nuclear Station, Units 1 and 2 - Evaluation of 2012 Field Investigation Results for Plant Relocation, Part 1 of 3.	Drawing Updated Final Safety Analysis Report (UFSAR)	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
5/2/2013	ML13127A227	Lee Nuclear Station, Units 1 and 2 - Evaluation of 2012 Field Investigation Results for Plant Relocation, Part 2 of 3.	Drawing Updated Final Safety Analysis Report (UFSAR)	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
5/2/2013	ML13127A228	Lee Nuclear Station, Units 1 and 2 - Evaluation of 2012 Field Investigation Results for Plant Relocation, Part 3 of 3.	Drawing Updated Final Safety Analysis Report (UFSAR)	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
5/9/2013	ML13144A150	William States Lee III Nuclear Station Update for Combined License Application, Revision 7.	Letter License-Application for Combined License (COLA)	Duke Energy Corp	NRC/Document Control Desk NRC/NRO	05200018 05200019

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5/9/2013	ML13144A244	Duke Energy WSL III Units 1 & 2 COLA (Generic DCD Departures Report) - Part 7, Departures and Exemptions Requests	Generic DCD Departures Report License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
5/9/2013	ML13144A721	Duke Energy WSL III Units 1 & 2 COLA (ITAAC) - Part 10, Conditions and ITAAC	Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
5/9/2013	ML13144A723	Duke Energy WSL III Units 1 & 2 COLA (Technical Specifications) - Part 4, Williams Lee III Nuclear Station Technical Specifications	Technical Specifications License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
5/15/2013	ML13135A471	05/16/2013 Presentation Slides For CEUS Public Meeting For Duke Energy's Seismic Review For William States Lee III Units 1 and 2 Combined License Application.	Meeting Briefing Package/Handouts Slides and Viewgraphs	Duke Energy Carolinas, LLC	NRC/NRO/DNRL	05200018 05200019

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
5/20/2013	ML13136A365	05/01/2013-Summary of Public Meeting For Levy Units 1 and 2 Electrical Bulletin.	Meeting Summary	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200025 05200027 05200028 05200029 05200030
5/21/2013	ML11154A153	Lee ASE Ch 06 Cover Letter.	Letter	NRC/NRO/DNRL/LB4	Duke Energy Carolinas, LLC	05200018 05200019
5/24/2013	ML11154A162	William States Lee III Nuclear Station, Units 1 and 2 Combined License Application - Advanced Safety Evaluation Without Open Items For Chapter 6, "Engineered Safety Features".	Memoranda	NRC/NRO/DNRL	NRC/ACRS	05200018 05200019
5/24/2013	ML13144A716	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report) Revision 7	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
5/29/2013	ML13140A311	Request for Additional Information Regarding The Environmental Review of The William States Lee III Nuclear Station, Units 1 and 2 Combined License Application.	Letter Request for Additional Information (RAI)	NRC/NRO/DNRL/EPB1	Duke Energy Carolinas, LLC	05200018 05200019

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5/29/2013	ML13149A150	06/13/2013-Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center Combined License Review Issues.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200029 05200030 05200040 05200041
5/29/2013	ML13149A287	06/27/2013-Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center Combined License Review Issues.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200029 05200030 05200040 05200041
5/30/2013	ML13149A509	05/30/2013 Notice of Cancelled Forthcoming Public Teleconference To Discuss AP1000 Design Center Combined License Review Issues.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200029 05200030 05200040 05200041

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5/31/2013	ML13247A094	DOE/EIA (U.S. Department of Energy/Energy Information Agency). 2012 Uranium Market Annual Report - May 2013. Washington, D.C.	Report, Administrative Annual Operating Report	US Dept of Energy, Energy Information Administration (EIA)	NRC/NRO	05200018 05200019
6/5/2013	ML13161A183	AP1000 Combined License Application for the William States Lee III Nuclear Station Units 1 and 2, Update Roadmap, Ltr#: WLG2013.06-01.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk	05200018 05200019
6/6/2013	ML13155A561	05/16/2013-Levy Combined License Units 1 and 2 Summary of Public Teleconference.	Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200029 05200030 05200040 05200041
6/10/2013	ML13247A051	U.S. Department of Energy (DOE). 2013. Manifest Information Management System.	Database File	US Dept of Energy (DOE)	NRC/NRO	05200018 05200019

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
6/10/2013	ML13259A074	U.S. Dept of Energy's Manifest Information Management System - Manifest Detail to Clive (Utah) Low-Level Radioactive Waste Facility in CY 2011.	- No Document Type Applies	EnergySolutions, LLC US Dept of Energy, Environmental Measurements Lab	NRC/NRO	05200018 05200019
6/19/2013	ML13156A404	05/16/2013 Summary Of Public Meeting With Duke Energy To Discuss The Seismic Review For The Licensing Action Related To The William States Lee III Units 1 And 2.	Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019
6/19/2013	ML13169A088	07/11/2013-Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center Combined License Review Issues.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200029 05200030 05200040 05200041

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6/19/2013	ML13169A091	07/25/2013-Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center Combined License Review Issues.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200029 05200030 05200040 05200041
6/19/2013	ML13175A265	William States Lee III, Units 1 & 2, Levy, Units 1 & 2, Submittal of Revision 9 to Quality Assurance Program Description (QAPD).	Letter Quality Assurance Program	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019 05200029 05200030
6/19/2013	ML13175A266	William States Lee III, Units 1 & 2, AP1000 Combined License Application Updated Schedule for Future Submittals.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
6/21/2013	ML13247A083	South Carolina Department of Health and Environmental Control (SCDHEC). 2013. Land and Waste Management. Chem-Nuclear Site, Barnwell County, South Carolina.	Database File	State of SC, Dept of Health & Environmental Control	NRC/NRO	05200018 05200019

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6/27/2013	ML13177A394	06/27/2013-CANCELLED-Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center Combined License Review Issues.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200029 05200030 05200040 05200041
6/27/2013	ML13182A472	William States Lee III, Units 1 & 2, Shearon Harris, Units 2 & 3 and Levy, Units 1 & 2 - Response to NRC Regulatory Issue Summary (RIS) 2013-08.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019 05200022 05200023 05200029 05200030
7/1/2013	ML13176A365	06/13/2013-Meeting Summary For Public Teleconference To Discuss AP1000 Design Center Combined License Review Issues.	Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200029 05200030 05200040 05200041

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Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
7/1/2013	ML13192A410	William States Lee III, Units 1 and 2, Response to Requests for Additional Information 7106, 7118, 7120, 7122 and 7123.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
7/3/2013	ML13190A479	William States Lee III, Units 1 and 2, AP1000 Combined License Application, Updates to Previous RAI Responses Impact by Plant Relocation and Additional Design Enhancements Submittal.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
7/12/2013	ML13247A064	National Oceanic and Atmospheric Administration (NOAA). 2013. Marine Pollution.	News Article	US Dept of Commerce, National Oceanic & Atmospheric Admin (NOAA)	NRC/NRO/DNRL	05200018 05200019
7/17/2013	ML13200A051	William States Lee Nuclear Station, Units 1 and 2, AP1000 Combined License Application, Partial Response to Request for Additional Information (RAI No. 6419).	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
7/17/2013	ML13200A052	William States Lee III Nuclear Station, Westinghouse Electric Company Report WLG-1000-S2R-003, Revision I, Summary of CEUS HRHF Task 1 - Evaluation of Floor Response Spectra and TR115 Component Margins.	Report, Technical	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
7/17/2013	ML13200A053	William States Lee III Nuclear Station, Units 1 and 2, AP1000 Combined License Application for Partial Response to Request for Additional Information (RAI No. 6419).	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
7/17/2013	ML13249A020	07/17/2013 - Permit from J.P. deBessonet, SCDHEC to Duke Energy Carolinas, LLC re: SC Dept of Health and Environmental Control "National Pollutant Discharge Elimination System Permit" (NPDES) (William S. Lee Nuclear Station) SC0049140 Final Permit.	- No Document Type Applies	State of SC, Dept of Radiological Health & Environmental Control	Duke Energy Carolinas, LLC NRC/NRO	05200018 05200019
7/22/2013	ML13151A063	William States Lee III Nuclear Station, Units 1 and 2 Combined License Application Review Schedule Revision.	Letter	NRC/NRO/DNRL	Duke Energy Carolinas, LLC	05200018 05200019

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7/23/2013	ML11151A255	Lee ASE Ch 19 LOLA Cover Letter.	Letter	NRC/NRO/DNRL/NWE1	Duke Energy Carolinas, LLC	05200018 05200019
7/23/2013	ML13193A206	07/11/2013-Cancelled Notice of Forthcoming Public Teleconference to Discuss AP1000 Design Center Combined License Review Issues.	Meeting Notice Meeting Agenda Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200029 05200030 05200040 05200041
7/24/2013	ML13205A093	2013/07/24 Lee RAI for SER - RAI LETTER NO. 109 RELATED TO SRP SECTION: 11.03 FOR THE WILLIAM STATES LEE III UNITS 1 AND 2 COL	E-Mail	NRC/NRO	NRC/NRO/DNRL/LB4	05200018 05200019
7/24/2013	ML13205A225	08/08/2013 - Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center Combined License Review Issues.	Meeting Notice Meeting Agenda Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200029 05200030 05200040 05200041

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7/24/2013	ML13205A252	08/22/2013 - Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center Combined License Review Issues.	Meeting Notice Meeting Agenda Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200029 05200030 05200040 05200041 05200022 05200023
7/25/2013	ML11151A258	Lee ASE Ch 19 LOLA ACRS Memo.	Memoranda	NRC/NRO/DNRL	NRC/ACRS	05200018 05200019
7/30/2013	ML13211A231	2013/07/30 Lee RAI for SER - LEE-RAI-LTR-110 RELATEDTO SRP SECTION 2.3.05 Dispersion for routine releasees for W.S. Lee Units 1& 2 COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/LB4	05200018 05200019

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Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
7/31/2013	ML13212A024	08/13/2013 Notice of Forthcoming Public Meeting with AP1000 Design Center Applicants and Licensees to Discuss Levy Unit 1 and 2 Combined License Application Passive Core Cooling System Condensate Return Exemption Request and Departure from the AP1000.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200025 05200026 05200027 05200028 05200029 05200030 05200040 05200041
7/31/2013	ML13214A093	E-Mail - Lee Nuclear Station - Cultural Resource Management Plan.	E-Mail	NRC/NRO/DNRL/EPB1	NRC/NRO/DNRL	05200018 05200019
8/1/2013	ML13213A439	North Carolina Department of Environment and Natural Resources' Updated Summary of North Carolina Species of Concern Records Within 15 Miles Of The Perkins Site.	- No Document Type Applies	NRC/NRO/DNRL/EPB1		05200018 05200019

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Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
8/1/2013	ML13213A450	E-Mail Transmitting North Carolina Department of Environment and Natural Resources' Updated Summary of North Carolina Species of Concern Records Within 15 Miles Of The Perkins Site.	E-Mail	NRC/NRO/DNRL/EPB1	NRC/NRO/DNRL/EPB1	05200018 05200019
8/8/2013	ML13218A259	07/25/2013-Summary of Public Teleconference with AP1000 Design Center Combined License Applicants To Discuss Application Review Issues.	Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200029 05200030 05200040 05200041
8/8/2013	ML13219A353	08/08/2013-CANCELLED-Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center Combined License Review Issues.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200029 05200030 05200040 05200041

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8/8/2013	ML13220A989	08/08/2013 E-MAIL from P. Vokoun, NRO/DNRL/EPB1 to J. Doub, NRO/DSEA/RENV FW: Lee Nuclear Station JD Letter (UNCLASSIFIED).	E-Mail	NRC/NRO/DNRL/EPB1	NRC/NRO/DSEA/RENV	05200018 05200019
8/9/2013	ML13221A199	Final numbers for waters of the U.S. that accompany Jurisdictional Determination Letter.	E-Mail	NRC/NRO/DNRL/EPB1	NRC/NRO/DSEA/RENV	05200018 05200019
8/9/2013	ML13221A201	Corps JD Numbers.	Spreadsheet File	NRC/NRO		05200018 05200019
8/9/2013	ML13221A445	Email from P. Vokoun to J. Doub Re: Transmitting Mixing Zone Request William States Lee III Nuclear Station NPDES Permit Date: December 19, 2012.	E-Mail	NRC/NRO/DNRL/EPB1	NRC/NRO/DSEA/RENV	05200018 05200019
8/11/2013	ML13224A005	Email from P. Vokoun to J. Doub re: transmitting USACE letter to Duke dated January 31, 2013 regarding 404(b)(1) Application	E-Mail	NRC/NRO/DNRL/EPB1	NRC/NRO/DSEA/RENV	05200018 05200019
8/11/2013	ML13224A052	E-mail from P. Vokoun to J. Doub re: Duke letter to Public Service Commission of South Carolina (PSCSC References) dated March 22, 2013 regarding Retirement Dates for Certain Units; 2012 Integrated Resource Plan.	E-Mail	NRC/NRO/DNRL/EPB1	NRC/NRO/DSEA/RENV	05200018 05200019

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8/19/2013	ML13226A135	09/12/2013-Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center Combined License Review Issues.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200029 05200030 05200040 05200041
8/19/2013	ML13226A139	09/26/2013-Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center Combined License Review Issues.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200029 05200030 05200040 05200041

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8/23/2013	ML13239A054	William States Lee III Nuclear Station - Response to Request for Additional Information (eRAI 7159).	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
8/23/2013	ML13239A055	Notification of Part 21 Report of Evaluation of Deviation Involving Technical Specifications Identified in Corrective Action Report as Having a Time Response Surveillance Requirement that Cannot be Directly Measured Due to Logic Pathway where Overlap...	Deficiency Report (per 10CFR50.55e and Part 21)	Westinghouse Electric Co	NRC/Document Control Desk NRC/NRO	05200014 05200015 05200018 05200019 05200022 05200023 05200025 05200026 05200027 05200028 05200029 05200030 05200040 05200041

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8/26/2013	ML13235A170	08/22/2013-CANCELLED-Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center Combined License Review Issues.	Meeting Agenda Meeting Notice	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200029 05200030 05200040 05200041
8/28/2013	ML13247A398	National Forest in North Carolina - About the Forest	FACT Sheet	US Dept of Agriculture, Forest Service		05200018 05200019
8/29/2013	ML13248A105	William States Lee III Nuclear Station Units 1 and 2 - Partial Response to Request for Additional Information (RAI) Letter 110, Related to SRP Section 02.03.05 - Long Term Atmospheric Dispersion Estimates for Routine Releases (RAI 7186).	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
9/3/2013	ML13247A003	National Education Center for Education Statistics - CCO Public School Data for School Years 2010-2011 and 2011-2012.	Report, Administrative	US Dept of Education	NRC/NRO	05200018 05200019

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
9/4/2013	ML13247A422	Major Uranium Recovery Licensing Applications	Spreadsheet File	NRC		05200018 05200019
9/12/2013	ML13253A222	10/24/2013-Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center Combined License Review Issues.	Meeting Agenda Meeting Notice	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200029 05200030 05200040 05200041
9/12/2013	ML13255A341	CANCELLED-09/12/2013-Notice of Forthcoming Public Teleconference to Discuss AP1000 Design Center Combined License Review Issues.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200029 05200030 05200040 05200041

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9/26/2013	ML13239A517	U.S. Department of Housing and Urban Development FY 2010 States Extremely Low Income Limit.	Report, Miscellaneous	US Dept of Housing & Urban Development	NRC/NRO	05200018 05200019
9/26/2013	ML13240A028	U.S. Department of Housing and Urban Development - Fiscal Year 2010 Median Family Income Documentation System Calculation.	Calculation	US Dept of Housing & Urban Development	NRC/NRO	05200018 05200019
9/26/2013	ML13269A212	09/26/2013-CANCELLED-Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center Combined License Review Issues.	Meeting Agenda Meeting Notice Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200029 05200030 05200040 05200041
9/26/2013	ML13277A214	09/26/2013 - Ltr from A.S. Meiburg, EPA, to Lt.Col. J.T. Litz and R.L. Darden, USACE Subject: William States Lee III Nuclear Station SAC-2009-122-SIR (Conformance to Section 404(b)(1) Guidelines.	Letter	US Environmental Protection Agency (EPA)	NRC/NRO US Dept of the Army, Corps of Engineers, Charleston District	05200018 05200019

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9/30/2013	ML13262A346	8/13/2013 - Summary of a Public Meeting to Discuss Levy Units 1 and 2 COL Application Passive Core Cooling System Condensate Return Exemption Request and Departure From the AP1000 Certified Design and Supporting Calculations.	Meeting Agenda Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200025 05200026 05200027 05200028 05200029 05200030 05200040 05200041
9/30/2013	ML13283A227	William States Lee III, Units 1 and 2 - AP1000 Combined License Application,Supplemental Response to Request for Additional Information (RAI) Letter 110, Related to SRP Section 02.03.05 - Long Term Atmospheric Dispersion Releases (RAI 7186).	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
10/9/2013	ML13281A899	10/10/2013-CANCELLED-Notice of Forthcoming Public Teleconference To Discuss AP1000 Design Center COL Review Issues.	Meeting Notice	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200029 05200030 05200040 05200041
10/10/2013	ML13283A080	Notice of the Secretary Regarding Agency Shutdown.	Legal-Order	NRC/SECY	Duke Energy Carolinas, LLC	05200018 05200019
10/17/2013	ML13290A406	Notice of Secretary Lifting Suspension of the Adjudicatory Activity in the Matter of Duke Energy Carolinas, LLC.	Legal-Order	NRC/SECY	Duke Energy Carolinas, LLC	05200018 05200019
1/29/2014	ML14029A073	12/12/2013-Summary of Closed Meeting with Duke Energy to Discuss William States Lee Central and Eastern United States (CEUS) Seismic Analyses information.	Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019

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1/30/2014	ML14030A187	2014/01/30 Lee RAI for SER - RAI LTR 111Related to SRP 13.03 the WS Lee Units 1 & 2 SRP 13.03 EPlan	E-Mail	NRC/NRO	NRC/NRO/DNRL/LB4	05200018 05200019
1/30/2014	ML14064A433	Enclosure 2, Duke Energy Letter Dated: January 30, 2014, Page 181 of 240 and Enclosure 3, Pages 1 through 7.	Report, Technical	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
1/30/2014	ML14064A434	Enclosure 4 to WLG2014.01-02, WLG-GW-GLR-815, Revision 0, "Effect of William S. Lee Site Specific Seismic Requirements on AP1000 SSCs".	Report, Technical	Duke Energy Carolinas, LLC Westinghouse Electric Co	NRC/NRO	05200018 05200019
1/30/2014	ML14064A435	William States Lee III, Units 1 and 2, Supplemental Response to Request for Additional Information Letter No. 105, Concerning Implementation of Fukushima Near-Term Task Force Recommendations (RAI 6419) Ltr# WLG2014.01-02.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
1/30/2014	ML14064A436	Enclosure 2, Duke Energy Letter Dated: January 30, 2014, Pages 58 of 100.	Report, Technical	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
1/30/2014	ML14064A438	Enclosure 2, Duke Energy Letter Dated: Jaunary 30, 2014, Pages 100 of 180.	Report, Technical	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
2/14/2014	ML14045A287	02/24/2014 Notice of Forthcoming Category 2 Public Teleconference to Discuss AP1000 Design Center Combined License Review Issues. Superseded by ML14050A328.	Meeting Agenda Meeting Notice	NRC/NRO		05200018 05200019 05200029 05200030 05200040 05200041
2/18/2014	ML14043A472	January 22, 2014 Summary Of Category 2 Meeting With Southern Nuclear, South Carolina Electric & Gas, Westinghouse And The General Public To Discuss Topics Related To The AP1000 Shield Building (Lower Sections) Detailed Design.	Meeting Summary Memoranda	NRC/NRO/DCIP/CIPB	NRC/NRO/DCIP/CIPB	05200014 05200015 05200018 05200019 05200022 05200023 05200025 05200026 05200027 05200028 05200029 05200030

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						05200040 05200041
2/19/2014	ML14050A328	02/27/2014 Meeting Notice with Public to Discuss AP100 Design Center Combined License Review Issues. Supersedes ML14045A287.	Meeting Agenda Meeting Notice	NRC/NRO		05200018 05200019 05200025 05200026 05200027 05200028 05200029 05200030 05200040

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						05200041
2/20/2014	ML14058A611	Letter to M. Sutton, NRC from EPA Re: Review and Comments on Final Environmental Impact Statement (FEIS) for Combined Licenses William States Lee III Nuclear Station Units 1 and 2, NUREG-2111, CEQ No. 20130379.	Letter	US Environmental Protection Agency (EPA)	NRC/NRO/DNRL/EPB1	05200018 05200019
2/24/2014	ML14055A353	03/13/2014 Notice of Forthcoming Category 2 Public Teleconference with Duke Energy to Discuss AP1000 Design Center Combined License Review Issues.	Meeting Agenda Meeting Notice	NRC/NRO		05200018 05200019 05200029 05200030 05200040 05200041

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2/26/2014	ML14057A470	02/05/2014 Summary of Category 2 Meeting with Southern Nuclear, South Carolina Electric & Gas, Westinghouse and the General Public to Discuss Topics Related to AP1000 Instrumentation and Control Systems Testing.	Meeting Summary Memoranda	NRC/NRO/DCIP/CIPB	NRC/NRO/DCIP/CIPB	05200014 05200015 05200018 05200019 05200022 05200023 05200025 05200026 05200027 05200028 05200029 05200030 05200040 05200041

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2/28/2014	ML14064A286	William States Lee III Nuclear Station, Units 1 and 2, AP1000 Combined License Application, Response to Request for Additional Information (RAI) Letter III, Related to SRP Section 13.03 Emergency Planning (RAI-7398).	Emergency Preparedness-Emergency Plan Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/28/2014	ML14065A024	William States Lee III, Units 1 & 2, AP1000 Combined License Application, Supplemental Response to Request for Additional Information Letter 105 Concerning Implementation of Fukushima Near-Term Task Force Recommendations (RAI 6419).	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
3/5/2014	ML14064A489	03/20/2014 Notice of Forthcoming Category 2 Public Teleconference with Duke Energy to Discuss AP1000 Design Center Combined License Review Issues.	Meeting Agenda Meeting Notice	NRC/NRO		05200018 05200019 05200029 05200030 05200040 05200041

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3/12/2014	ML14071A521	2014/03/12 Lee RAI for SER - LEE-RAI-LTR-112 related to the stability of the subsurface M and F for William S Lee Units 1 and 2 COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/LB4	05200018 05200019
3/21/2014	ML14080A014	04/03/2014 Notice of Public Teleconference to Discuss AP1000 Design Center COL Regulatory Issues with the Nuclear Regulatory Commission (NRC).	Meeting Agenda Meeting Notice	NRC/NRO		05200018 05200019 05200029 05200030 05200040 05200041
3/26/2014	ML14072A346	12/18/2013-Summary of Levy COL Units 1 and 2 Public Meeting - Staff Questions for Design Change.	Meeting Agenda Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200025 05200026 05200027 05200028 05200029 05200030 05200040

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
						05200041
3/27/2014	ML14090A054	William States Lee III, Units 1 and 2 - AP1000 Combined License Application Revised Response to Request for Additional Information (RAI) Related to SRP Section 09.02.01.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
3/27/2014	ML14090A056	William States Lee III, Units 1 and 2 - Revised Supplemental Response to Request for Additional Information Letter 064, RAI 09.02.01-6.	Drawing Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
3/27/2014	ML14091B035	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report) - FSAR Chapter 14 Initial Test Program - Sections 14.01 - 14.04	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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3/31/2014	ML14085A431	03/20/2014 Summary of Public Teleconference with AP1000 Design Center COL Applicants to Discuss Application Review Issues.	Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200029 05200030 05200040 05200041
3/31/2014	ML14115A334	Attachment 1 - Letter from Paul A. Russ, Westinghouse Electric Company (WEC), to the Nuclear Regulatory Commission (NRC), 10 CFR 50.46 Annual Report for the AP1 000 Standard Plant Design, Letter No. DCP_NRC_003262, dated March 31, 2014.	Annual Report Letter	Westinghouse Electric Co, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019 05200022 05200023 05200029 05200030
4/1/2014	ML14091B042	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report) Revision 8	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019

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4/2/2014	ML14083A396	02/27/2014 - Summary of Public Teleconference for Levy Combined License, Units 1 and 2.	Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200029 05200030 05200040 05200041
4/10/2014	ML14104A022	William States Lee III Nuclear Station - Docket Nos. 52-018 and 52-019 AP1000 Combined License Application for the William States Lee III Nuclear Station Units 1 and 2 Update Roadmap.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
4/23/2014	ML14115A335	William States Lee III Nuclear Station (WLS), Units 1 & 2 and Levy Nuclear Plant (LNP), Units 1 and 2 - 10 CFR 50.46 Annual Report.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019 05200022 05200023 05200029 05200030

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4/24/2014	ML14115A329	William States Lee III AP1000 Combined License Application, Units 1 and 2, Response to Request for Additional Information Letter No. 112 (eRAI 7436),	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
5/7/2014	ML14127A596	2014/05/07 Lee RAI for SER - RAI Letter No. 114 SRP SECTION 3.10 SEISMIC and Mechanical for thr W. S. Lee Units 1 & 2	E-Mail	NRC/NRO	NRC/NRO/DNRL/LB4	05200018 05200019
5/9/2014	ML14119A414	04/03/2014-Levy COL Units 1 and 2 Public Teleconference Meeting Summary.	Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200022 05200023 05200029 05200030 05200040 05200041
5/14/2014	ML14134A573	2014/05/14 Lee RAI for SER - RAI Letter 116 related to SRP Section 3.12 Piping System & Componenets for the WS LEE COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/LB4	05200018 05200019

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5/15/2014	ML14135A538	2014/05/15 Lee RAI for SER - RAI Letter 115 SER SECTION 03.07.5.4 FOR THE WILLIAM STATES LEE III UNITS 1 AND 2 COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/LB4	05200018 05200019
5/16/2014	ML14135A335	05/01/2014-Summary of Public Meeting For Levy COL, Units 1 and 2.	Meeting Summary	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200029 05200030 05200040 05200041
5/20/2014	ML14126A584	02/10/2014 - Trip Report Geologic Site Visit in Support of the William States Lee III Combined License Application.	Memoranda Trip Report	NRC/NRO/DNRL/LB1	NRC/NRO/DNRL/LB4	05200018 05200019
5/27/2014	ML14148A001	2014/05/27 Lee RAI for SER - RAI Letter related to 0.2.5.2 - Vibratory Ground Motion For the W. S. Lee COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/LB4	05200018 05200019
6/3/2014	ML14150A120	05/22/2014 Summary of Public Teleconference With AP1000 Design Center Combined License Applicants To Discuss Application Review Issues.	Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200029 05200030

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						05200040 05200041
6/5/2014	ML14160A720	William States Lee III, AP1000 Combined License Application Response to Request for Additional Information Letter No. 114 (eRAI 7482).	Graphics incl Charts and Tables Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
6/9/2014	ML14160B254	2014/06/09 Lee RAI for SER - RAI LETTER 118 RELATED TO SRP 3.7.1 Seismic Design for the W.S. LEE COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/LB4	05200018 05200019
6/11/2014	ML14163A571	William States Lee III, Units 1 & 2, APl000 Combined License Application - Response to Request for Additional Information Letter No. 116 (eRAI 7539) Ltr#: WLG2014.06-04.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC Duke Energy Corp	NRC/Document Control Desk NRC/NRO	05200018 05200019

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6/11/2014	ML14163A572	William States Lee III Units 1 and 2, AP1000 Combined License Application - Response to Request for Additional Information Letter No. 115 (eRAI 7510) Ltr#: WLG2014.06-03.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC Duke Energy Corp	NRC/Document Control Desk NRC/NRO	05200018 05200019
6/16/2014	ML14157A015	06/05/2014 Summary of a Public Meeting with Members of AP1000 Design Center to Discuss Request for Exemption and Departure Related to AP1000 Containment Condensate Return Design.	Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200025 05200026 05200027 05200028 05200029 05200030 05200040 05200041

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6/19/2014	ML14168A260	June 10, 2014, Summary of Public Teleconference with Members of AP1000 Design Center to Discuss AP1000 Licensing and Technical Issues, Levy Combined License Units, 1 and 2.	Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200025 05200026 05200027 05200028 05200029 05200030 05200040 05200041
6/26/2014	ML14182A440	AP1000 Combined License Application for the William States Lee III Nuclear Station Units 1 and Supplemental Response to Request for Additional Information (RAI) Letter 111, Related to SRP Section 13.03 Emergency Planning (RAI-7398).	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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6/30/2014	ML14154A523	Request for Additional Information Letter No. 113, Related to SRP Section 13.06.01 Physical Security for the William States Lee III Units 1 and 2 Combined License Application (RAI 7479).	Letter	NRC/NRO/DNRL/LB4	Duke Energy Florida, Inc	05200018 05200019
6/30/2014	ML14154A524	Enclosure 1 - Request for Additional Information 113, Issue Date: 06/26/2014, William States Lee III, Units 1 and 2, Review Section 13.06.01, Physical Security.	Request for Additional Information (RAI)	NRC/NRO		05200018 05200019
7/10/2014	ML14195A018	William States Lee III, Units 1 and 2, AP1000 Combined License Application, Response to Request for Additional Information (eRAI 7500).	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
7/11/2014	ML14192B073	2014/07/11 Lee RAI for SER - RAI LTR No 120 SRP SECTION 3-7-2 SEISMIC SYSTEM ANALYSIS FOR THE LEE SCOL	E-Mail	NRC/NRO	NRC/NRO/DNRL/LB4	05200018 05200019
7/14/2014	ML14195A483	2014/07/14 Lee RAI for SER - RAI LTR 122 RELATED TO SRP 3.10 SEISMIC and DYNAMIC QUALS for the LEE COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/LB4	05200018 05200019
7/15/2014	ML14196A303	2014/07/15 Lee RAI for SER - RAI Letter 121 Related to	E-Mail	NRC/NRO	NRC/NRO/DNRL/LB4	05200018 05200019

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		SRP 3.8.5 Foundations for the Lee COLA				
7/17/2014	ML14198A300	2014/07/17 Lee RAI for SER - RAI Letter 119 Related to SRP Section 3.7 Seismic Design for the WLS COLA	E-Mail	NRC/NRO	NRC/NRO/DNRL/LB4	05200018 05200019
7/24/2014	ML14206A950	William States Lee III, Units 1 and 2, AP1000 Combined License Application, Response to Request for Additional Information (eRAI 7544).	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
7/28/2014	ML14189A036	Levy Combined License Units 1 and 2, Summary of June 17, 2014, Public Meeting.	Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200025 05200026 05200027 05200028 05200029 05200030 05200040 05200041

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7/29/2014	ML14205A144	July 16, 2014 Summary of Public Teleconference With Licensees Concerning the Levy Combined License, Units 1 and 2.	Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200025 05200026 05200027 05200028 05200029 05200030 05200040 05200041
8/1/2014	ML14189A042	July 1, 2014, Summary of Public Teleconference with Members of the AP1000 Design Center to Discuss AP1000 Licensing and Technical Issues.	Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200025 05200026 05200027 05200028 05200029 05200030

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						05200040 05200041
8/4/2014	ML14205A111	June 24, 2014, Summary of a Public Teleconference with Members of AP1000 Design Center to Discuss AP1000 Licensing and Technical Issues.	Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200025 05200026 05200027 05200028 05200029 05200030 05200040 05200041

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8/4/2014	ML14205A131	July 10, 2014, Summary of Public a Public Teleconference With Members of AP1000 Design Center To Discuss AP1000 Licensing and Technical Issues	Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200025 05200026 05200027 05200028 05200029 05200030 05200040 05200041
8/7/2014	ML14220A432	William States Lee III Nuclear Station Units 1 & 2, AP1000 Combined License Application, Response to Request for Additional Information Letter No. 120 (eRAI 7570) Ltr#: WLG2014.08-01.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC Duke Energy Corp	NRC/Document Control Desk NRC/NRO	05200018 05200019

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8/13/2014	ML14227A708	AP1000 Combined License Application for the William States Lee III Nuclear Station Units 1 and 2, Response to Request for Additional Information Letter No. 122 (eRAI 7573) Ltr: WLG2014.08-03.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
8/14/2014	ML14227A706	William States Lee III Nuclear Station Units 1 and 2 AP1000 Combined License Application, Response to Request for Additional Information Letter No. 121 (eRAI 7571).	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
8/19/2014	ML14210A383	July 23, 2014, Summary of Public Teleconference Regarding Levy Combined License, Units 1 and 2.	Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200025 05200026 05200027 05200028 05200029 05200030
8/26/2014	ML14238A242	Commission Memorandum and Order CLI-14-08.	Legal-Order	NRC/SECY		05200018 05200019

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
8/28/2014	ML14245A470	William States Lee III, Units 1 and 2, AP1000 Combined License Application, Supplemental Response to Request for Additional Information (eRAI 6751) Ltr#: WLG2014.08-02.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

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8/29/2014	ML14245A386	NRDC v. NRC et. al., No. 13-1311 (Scheduled oral Argument November. 21, 2014).	Legal-Memorandum and Order Letter	NRC/OGC	US Federal Judiciary, US Court of Appeals for the District of Columbia Circuit	05000247 05000275 05000286 05000323 05000327 05000328 05000346 05000352 05000353 05000391 05000416 05000443 05000483 05000498 05000499 05200012 05200013

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						05200014 05200015 05200016 05200017 05200018 05200019 05200022 05200023 05200024 05200029 05200030 05200033 05200039 05200040 05200041 07200010

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9/29/2014	ML14252A225	August 28, 2014, Summary of Public Teleconference with Members of the AP1000 Design Center to Discuss AP1000 Licensing and Technical Issues Levy Combined License, Units 1 and 2, .	Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200025 05200026 05200027 05200028 05200029 05200030 05200040 05200041
11/11/2014	ML14329B215	Duke Energy WSL III Units 1 & 2 COLA (ITAAC) - Part 10, Conditions and ITAAC	Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019

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11/11/2014	ML14329B217	Duke Energy WSL III Units 1 & 2 COLA (Technical Specifications) - Part 4, Williams Lee III Nuclear Station Technical Specifications	License-Application for Combined License (COLA) Technical Specifications	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
11/11/2014	ML14329B205	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report) Revision 9	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
1/8/2015	ML15014A034	William States Lee III, Units 1 & 2 - AP1000 Combined License Application, Supplemental Response to Request for Additional Information (RAI No. 50) RAI 13.03-55 Ltr#: WLG2015.01-01.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC Duke Energy Corp	NRC/Document Control Desk NRC/NRO	05200018 05200019
1/12/2015	ML15007A268	12/11/2014 Meeting Summary with Members Of AP1000 Design Center To Discuss AP1000 Licensing And Technical Issues.	Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB1	05200018 05200019 05200029 05200030 05200040 05200041

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1/13/2015	ML15014A443	William States Lee III, Units 1 and 2, AP1000 Combined License Application Revised Response to Request for Additional Information Letter No. 116 (eRAI 7539) Ltr#: WLG2015.01-02.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC Duke Energy Corp	NRC/Document Control Desk NRC/NRO	05200018 05200019
1/21/2015	ML15023A037	William States Lee III, AP1000 Combined License Application, Process for Documenting Emergent Generic Changes to AP1000 Design Basis Information.	Letter	Duke Energy Carolinas, LLC Duke Energy Corp	NRC/Document Control Desk NRC/NRO	05200018 05200019
1/28/2015	ML15029A086	Petition to Supplement Reactor-Specific Environmental Impact Statements to Incorporate by Reference the Generic Environmental Impact Statement for Continued Spent Fuel Storage (William States Lee III Nuclear Station, Units 1 and 2).	Legal-Petition To Intervene/Request for Hearing	Blue Ridge Environmental Defense League	NRC/OCM	05200018 05200019
1/29/2015	ML15029A713	Order of the Secretary Establishing Petition Answer and Response Due Dates.	Legal-Order	NRC/SECY	Duke Energy Carolinas, LLC	05200018 05200019
2/2/2015	ML15015A309	01/14/2015 - Summary of Meeting with Members of the AP1000 Design Center to Discuss AP1000 Licensing and Technical Issues.	Meeting Agenda Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200029 05200030

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						05200040 05200041
2/10/2015	ML15037A440	01/29/2015 Summary of Public Teleconference with Between NRC and Members of AP1000 Design Center to Discuss AP1000 Licensing and Technical Issues.	Meeting Summary Memoranda	NRC/NRO/DNRL/LB1	NRC/NRO/DNRL/LB1	05200018 05200019
2/12/2015	ML15043A619	Notice Of Appearance Of Timothy J. V. Walsh.	Legal-Correspondence/Miscellaneous	Duke Energy Carolinas, LLC Pillsbury, Winthrop, Shaw, Pittman, LLP	NRC/OCM	05200018 05200019
2/12/2015	ML15043A622	Answer Of Duke Energy Carolinas, LLC Opposing Petition To Supplement W. S. Lee Final Environmental Impact Statement.	Legal-Pleading	Duke Energy Carolinas, LLC Pillsbury, Winthrop, Shaw, Pittman, LLP	NRC/OCM	05200018 05200019
2/12/2015	ML15043A623	Notice of Change of Address - Lewis.	Legal-Correspondence/Miscellaneous	Duke Energy Carolinas, LLC Pillsbury,	NRC/OCM	05200018 05200019

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				Winthrop, Shaw, Pittman, LLP		
2/12/2015	ML15043A769	NEI Motion and Amicus Brief.	Legal-Motion	Nuclear Energy Institute (NEI)	NRC/OCM	05200018 05200019
2/12/2015	ML15043A777	NRC Staff Opposition to the "Petition to Supplement Reactor-Specific Environmental Impact Statements to Incorporate by Reference the Generic Environmental Impact Statement for Continued Spent Fuel Storage."	Legal-Pleading	NRC/OGC	NRC/OCM	05200018 05200019
2/18/2015	ML15049A657	Petitioners' Reply to Oppositions to Petition to Supplement Reactor-Specific Environmental Impact Statements to Incorporate by Reference the Generic Environmental Impact Statement for Continued Spent Fuel Storage.	Legal-Motion	Blue Ridge Environmental Defense League	NRC/OCM	05200018 05200019

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2/25/2015	ML15061A045	AP1000 Combined License Application for the William States Lee III Nuclear Station, Units 1 and 2 - Supplemental Response to Request for Additional Information (RAI) Letter 123, Related to SRP Section 13.03 Emergency Planning (RAI-7686).	Emergency Preparedness-Emergency Plan Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/26/2015	ML15057A278	Commission Memorandum and Order CLI-15-04.	Legal-Order	NRC/SECY		05200018 05200019
3/16/2015	ML15077A176	William States Lee III Nuclear Station, Units 1 and 2 - AP1000 Combined License Application for the Supplemental Response 2 to Request for Additional Information Letter No. 25 (eRAI 50), RAI 13.03-061, SITE-8 Item J.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
3/23/2015	ML15040A027	12/16/2014-Summary of Closed Meeting with Members of the AP1000 Design Center to Discuss Changes to Proprietary Calculations Supporting AP1000 Condensate Return Design Change, Departure, and Exemption Request.	Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200025 05200026 05200027 05200028

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						05200029 05200030 05200040 05200041
4/7/2015	ML15092A287	03/26/2015-Meeting Summary With Members of The AP1000 Design Center To Discuss AP1000 Licensing and Technical Issues.	Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200029 05200030 05200040 05200041
4/15/2015	ML15083A218	Letter to Mr. Fallon from Mr. Tracy reg. Response to Duke Energy Carolinas' Letter Requesting Guidance Clarifying Appropriate Methods For Resolving Generic Errors in Certified Design Information (Levy and Lee COL Applications).	Letter	NRC/NRO	Duke Energy Florida, Inc	05200018 05200019 05200029 05200030

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4/16/2015	ML15119A012	William States Lee III Nuclear Station (WLS), Units 1 and 2, Levy Nuclear Plant (LNP), Units 1 and 2 & Shearon Harris Nuclear Power Plant (HAR), Units 2 and 3 - 10 CFR 50.46 Annual Report.	Annual Report Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019 05200022 05200023 05200029 05200030
4/22/2015	ML15112B197	Blue Ridge Environmental Defense League's Motion to Reopen the Record of Combined License Proceeding for W.S. Lee Nuclear Power Plant.	Legal-Motion	Blue Ridge Environmental Defense League	NRC/SECY	05200018 05200019
4/22/2015	ML15112B198	Blue Ridge Environmental Defense League's Hearing Request and Petition to Intervene In Combined License Proceeding for W.S. Lee Nuclear Power Plant.	Legal-Petition To Intervene/Request for Hearing	Blue Ridge Environmental Defense League	NRC/SECY	05200018 05200019
4/22/2015	ML15114A363	William States Lee III Nuclear Station - AP1000 Combined License Application for the William States Lee III Nuclear Station Units 1 and 2 Departure Report Update.	Letter Report, Miscellaneous	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
4/23/2015	ML15113A304	Commission Memorandum and Order CLI-15-10.	Legal-Order	NRC/SECY	Duke Energy Carolinas, LLC	05200018 05200019

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4/29/2015	ML15119A584	NRC Staff Answer To Blue Ridge Environmental Defense League's Motion To Reopen The Record And Petition To Intervene.	Legal-Pleading	NRC/OGC	NRC/OCM	05200018 05200019
5/6/2015	ML15126A478	Reply By Beyond Nuclear, Blue Ridge Environmental Defense League, Nuclear Information And Resource Service, Seed Coalition And Southern Alliance For Clean Energy To Oppositions By Applicants And NRC Staff To Motions To Admit New Contentions.	Legal-Pleading	Blue Ridge Environmental Defense League	NRC/SECY	05000327 05000328 05000346 05000391 05000498 05000499 05200012 05200013 05200017 05200018 05200019 05200029 05200030 05200034 05200035

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5/7/2015	ML15127A213	Duke's Answer Opposing Blue Ridge Environmental Defense League's Motion To Reopen And Petition To Intervene In The Combined License Proceeding For William States Lee III Nuclear Station, Units 1 And 2.	Legal-Pleading	Duke Energy Carolinas, LLC Pillsbury, Winthrop, Shaw, Pittman, LLP	NRC/OCM	05200018 05200019
5/7/2015	ML15127A251	Reply By Beyond Nuclear, Blue Ridge Environmental Defense League, Nuclear Information And Resource Service, Seed Coalition And Southern Alliance For Clean Energy To Oppositions By Applicants And NRC Staff To Motions To Admit New Contentions.	Legal-Pleading	Robert V. Eye Law Office, LLC SEED Coalition	NRC/SECY	05000017 05000327 05000328 05000346 05000391 05000498 05000499 05200012 05200013 05200018 05200019 05200029 05200030

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
						05200034 05200035
5/7/2015	ML15132A098	William States Lee III, Units 1 and 2, AP1000 Combined License Application for Supplemental Response for Request for Additional Information Letter No. 105 Concerning Implementation of Fukushima Near-Term Task Force Recommendations.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
5/7/2015	ML15132A099	William States Lee III, Units 1 and 2, AP1000 Combined License Application for Supplemental Response for Request for Additional Information Letter No. 105 and 01.05-2 (eRAI 6419).	Report, Miscellaneous	Westinghouse Electric Co, LLC	NRC/NRO	05200018 05200019

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5/8/2015	ML15128A747	Petitioners' Reply to Oppositions by Applicant and NRC Staff to Motions to Admit New Contentions.	Legal-Pleading	Blue Ridge Environmental Defense League	NRC/SECY	05200018 05200019
5/18/2015	ML15294A105	William States Lee III, Units 1 and 2 - Bat Acoustic Monitoring Report - Summer 2015.	E-Mail Environmental Report	US Dept of Interior, Fish & Wildlife Service	NRC/NRO	05200018 05200019
5/20/2015	ML15005A343	Lee Phase B Chapter 8 SER - Electric Power.	Final Safety Analysis Report (FSAR)	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
5/20/2015	ML15113A652	Letter - William States Lee III Nuclear Station, Combined License Application - Safety Evaluation for Chapter 08, Electric Power.	Letter	NRC/NRO/DNRL/LB1	Duke Energy Florida, Inc	05200018 05200019
5/21/2015	ML15113A956	ACRS Memo William States Lee III Nuclear Station, Units 1 & 2 Combined License Application - Advanced Safety Evaluation without Open Items for Chapter 8, "Electric Power."	Memoranda	NRC/NRO/DNRL	NRC/ACRS	05200018 05200019
5/28/2015	ML11131A088	William States Lee, Units 1 and Combined License Application - Advanced Safety Evaluation without Open Items for Section 2.4, "Hydrologic Engineering."	Memoranda	NRC/NRO/DNRL	NRC/ACRS	05200018 05200019
5/28/2015	ML15055A352	Lee Phase B Section 2 4 AP1000.	Safety Evaluation	NRC/NRO		05200018 05200019

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6/9/2015	ML15160A184	Commission Memorandum And Order CLI-15-15.	Legal-Order	NRC/SECY	Duke Energy Carolinas, LLC	05200018 05200019
6/18/2015	ML15160A258	Letter - William States Lee III Nuclear Station, Units 1 And 2 COL Application ASE With No Open Items For Chapter 12, "Radiation Protection".	Letter	NRC/NRO/DNRL	Duke Energy Carolinas, LLC	05200018 05200019
6/23/2015	ML15149A138	05/14/2015-Summary of Meeting With Members of the AP1000 Design Center to Discuss AP1000 Licensing and Technical Issues.	Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200025 05200026 05200027 05200028 05200029 05200030 05200040 05200041
6/24/2015	ML15160A231	ACRS Memo - William States Lee III Nuclear Station, Units 1 and 2 COL Application ASE With No Open Items For Chapter 12, Radiation Protection.	Memoranda	NRC/NRO/DNRL	NRC/ACRS	05200018 05200019

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6/26/2015	ML15154B450	05/27/2015-Summary of Meeting with Members of the AP1000 Design Center to Discuss AP1000 Licensing and Technical Issues.	Meeting Summary	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200025 05200026 05200027 05200028 05200029 05200030 05200040 05200041
6/26/2015	ML15169A199	06/11/2015 - Summary of Meeting with Members of the AP1000 Design Center to Discuss AP1000 Licensing and Technical Issues.	Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200029 05200030 05200040 05200041

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6/26/2015	ML15176A260	06/18/2015-Summary of Meeting with Members of the AP1000 Design Center to Discuss AP1000 Licensing and Technical Issues.	Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200029 05200030 05200040 05200041
7/21/2015	ML15195A183	Issuance Letter to Applicant - William States Lee III Nuclear Station Combined License Application Advanced Safety Evaluation with No Open Items for Chapter 2.0 - 2.3, Site Characteristics and Site Parameters.	Letter	NRC/NRO/DNRL/LB1	Duke Energy Florida, Inc	05200018 05200019
7/23/2015	ML15195A182	ACRS Memorandum - William States Lee III Nuclear Station Combined License Application Advanced Safety Evaluation with no Open Items for Chapter 2.0 - 2.3, Site Characteristics and Site Parameters.	Memoranda	NRC/NRO/DNRL	NRC/ACRS	05200018 05200019

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Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
8/25/2015	ML15341A325	FWS Letter to USACE: Concurrence Regarding Potential Impacts to Federally Protected Resources resulting from William States Lee III Nuclear Station Units 1 and 2.	Letter	US Dept of Interior, Fish & Wildlife Service	US Dept of the Army, Corps of Engineers US Dept of the Army, Corps of Engineers, Charleston District	05200018 05200019
8/31/2015	ML15225A435	07/16/2015-Summary of Meeting with Members of the AP1000 Design Center to Discuss AP1000 Licensing and Technical Issues.	Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200029 05200030 05200040 05200041
9/22/2015	ML15240A215	Letter to the Applicant - William States Lee III Nuclear Station, Units 1 and 2 ASE Without Open Items for Chapter 20, "Requirements Resulting From Fukushima Near-Term Task Force Recommendations."	Letter	NRC/NRO/DNRL/LB4	Duke Energy Florida, Inc	05200018 05200019

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
9/23/2015	ML15240A187	Memorandum to E. Hackett - Re: William S. Lee Nuclear Station, Units 1 and 2 ASE Without Open Items for Chapter 20, "Requirements Resulting From Fukushima Near-Term Force Recommendations."	Memoranda	NRC/NRO/DNRL	NRC/ACRS	05200018 05200019
9/29/2015	ML15259A111	09/10/2015-Meeting Summary With Members of the AP1000 Design Center to Discuss AP1000 Licensing and Technical Issues.	Meeting Agenda Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200029 05200030 05200040 05200041
9/29/2015	ML15274A134	William States Lee, Units 1 and 2 - Voluntary Submittal of Exemption Request and Design Change Description for Departure from API1000 DCD Revision 19 to Address Compliance With IEEE 603 - 1991.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
9/30/2015	ML15254A261	08/27/2015 - Summary of Meeting With Members of the AP1000 Design Center To Discuss AP1000 Licensing and Technical Issues.	Meeting Agenda Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200029 05200030

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
						05200040 05200041
10/6/2015	ML15252A478	Letter To Applicant - William States Lee Chapter 6, "Engineered Safety Features."	Letter	NRC/NRO/DNRL/LB4	Duke Energy Carolinas, LLC	05200018 05200019
10/7/2015	ML15272A470	Memorandum to E. M. Hackett - William States Lee III Nuclear Station, Units 1 and 2 Combined License Application - Advanced Safety Evaluation Without Open Items for Chapters 1 Through 20.	Memoranda	NRC/NRO/DNRL	NRC/ACRS	05200018 05200019
10/19/2015	ML15293A518	William States Lee III, Units 1 and 2 - Departure Report Update.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
10/19/2015	ML15294A104	William States Lee III, Units 1 and 2 - Submittal of Bat Acoustic Monitoring Report - Summer 2015.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
10/29/2015	ML15272A200	09/17/2015-Meeting Summary With Members of the AP1000 Design Center to Discuss AP1000 Licensing and Technical Issues.	Meeting Agenda Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200029 05200030

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Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
						05200040 05200041
10/29/2015	ML15288A142	09/23/2015-Summary of Meeting With Members of the AP1000 Design Center to Discuss AP1000 Licensing and Technical Issues.	Meeting Agenda Meeting Summary	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200004 05200018 05200019 05200025 05200027 05200028 05200029 05200030 05200040 05200041
11/3/2015	ML15254A320	Letter To Applicant: W. S. Lee III COL Application - ASE With No Open Items For Chapter 13, "Conduct of Operations."	Letter	NRC/NRO/DNRL/LB4	Duke Energy Florida, Inc	05200018 05200019
11/3/2015	ML15308A586	William States Lee, Units 1 and 2 - Information to Address ACRS	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk	05200018 05200019

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
		Subcommittee Follow-up Items.			NRC/NRO	
11/12/2015	ML15254A333	Memorandum to ACRS: William States Lee III, Units 1 and 2 COL Application - ASE With No Open Items For Chapter 13, "Conduct of Operations."	Memoranda	NRC/NRO/DNRL	NRC/ACRS	05200018 05200019
11/24/2015	ML15336A079	William States Lee III Nuclear Station, Submittal of Update for Combined License Application.	Letter License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
11/24/2015	ML15336A333	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report) - FSAR Chapter 02 Figure 2.4.4-201	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
11/24/2015	ML15336A688	Duke Energy WSL III Units 1 & 2 COLA (ITAAC) - Part 10, Conditions and ITAAC	Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019

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Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
11/24/2015	ML15336A692	Duke Energy WSL III Units 1 & 2 COLA (Technical Specifications) - Part 4, Williams Lee III Nuclear Station Technical Specifications	License-Application for Combined License (COLA) Technical Specifications	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
12/2/2015	ML15336A678	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report) Revision 10	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
12/15/2015	ML15351A470	William States Lee, Units 1 and 2 - COLA Submittal 13 Update Roadmap.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
1/22/2016	ML16012A131	LTR-15-0622 Ltr. to John W. Stetkar from Victor McCree re: Report on the Safety Aspects of the Duke Energy Carolinas, LLC, Combined License Application for William States Lee III Nuclear Station, Units 1 and 2.	Letter	NRC/EDO	NRC/ACRS	05200018 05200019
1/29/2016	ML16034A062	William States Lee III, Units 1 and 2 - Voluntary Submittal of Exemption Request and Design Change Description for Departure from AP1000 DCD Revision 19 to Address Combustible Gas Control in Containment.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
						05000261 05000269 05000270 05000287 05000302 05000324 05000325 05000369 05000370 05000400 05000413 05000414 05200018
2/1/2016	ML16053A093	Certificate of Liability Insurance.	Financial Assurance Document	Marsh USA, Inc	NRC/Document Control Desk NRC/NRO NRC/NRR	05200019 05200029 05200030

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Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
2/9/2016	ML16041A586	William States Lee III, Units 1 and 2 - Endorsement of Voluntary Submittal of Response to the LNP (DEE) Request for Additional Information Letter No. 135 Related To IEEE 603 And Source Range Nuclear Instrumentation Flux Doubling.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/9/2016	ML16043A123	William States Lee III, Units 1 and 2 - Voluntary Submittal of Exemption Request and Design Change Description for Departure from AP1000 DCD Revision 19 to Address Main Control Room Habitability Analysis.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/12/2016	ML16013A339	12/10/2015 - Summary of Meeting with Members of the AP1000 Design Center to Discuss AP1000 Licensing and Technical Issues.	Meeting Agenda Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200029 05200030 05200040 05200041

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2/12/2016	ML16049A411	William States Lee III, Units 1 and 2, Voluntary Submittal of Exemption Request and Design Change Description for Departure from AP1000 DCD Revision 19 to Address Main Control Room Dose Analysis.	Letter Response to Request for Additional Information (RAI)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
2/17/2016	ML16050A173	William States Lee III, Units 1 and 2, Vogtle, Units 3 and 4, Response to Request for Additional Information Letters No. 04 and 05, "Mitigating Strategies for Beyond Design Basis External Events."	Letter Response to Request for Additional Information (RAI)	Duke Energy Florida, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
2/19/2016	ML16056A014	Marsh USA, Inc. - 2016 Nuclear Liability Certificates of Insurance.	Letter	Marsh USA, Inc	NRC/Document Control Desk NRC/NMSS NRC/NRO NRC/NRR	05000010 05000029 05000133 05000146 05000171 05000213 05000219 05000220 05000237 05000244 05000245 05000249 05000250 05000251 05000254 05000259 05000260

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						05000261 05000263 05000265 05000269 05000270 05000272 05000275 05000277 05000278 05000280 05000281 05000282 05000285 05000287 05000289 05000295 05000296

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						05000302
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						05000336 05000338 05000339 05000346 05000348 05000352 05000353 05000361 05000362 05000364 05000366 05000369 05000370 05000373 05000374 05000387 05000388

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						05000389 05000390 05000391 05000395 05000397 05000400 05000410 05000413 05000414 05000423 05000424 05000425 05000440 05000441 05000443 05000444 05000445

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Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
						05000446 05000454 05000455 05000456 05000457 05000461 05000482 05000528 05000529 05000530 05200018 05200019 05200029 05200030 07000027

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Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
3/1/2016	ML16043A492	01/25/2016-Summary of Meeting With Members of the AP1000 Design Center To Discuss AP1000 Licensing and Technical Issues	Meeting Agenda Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200029 05200030 05200040 05200041
3/1/2016	ML16043A519	02/26/2015 - Summary of Meeting with Members of the AP1000 Design Center to Discuss AP1000 Design Matters for the Levy Nuclear Plant, Units 1 and 2 COL-Estimated Dose to Main Control Room Operators, MCR Heatup, and Hydrogen Vent Location, ITAAC.	Meeting Summary	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200025 05200026 05200027 05200028 05200029 05200030 05200040 05200041

Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
3/2/2016	ML16062A020	2016/03/02 Levy County COL - FW: RE: AP1000 Generic License Conditions	E-Mail	NRC/NRO	NRC/NRO/DNRL/LB4	05200018 05200019 05200029 05200030 05200040 05200041
3/7/2016	ML16057A668	William States Lee Nuclear Station, Units 1 and 2 Combined License Application Review Schedule Revision.	Letter Project Plans and Schedules	NRC/NRO/DNRL	Duke Energy Carolinas, LLC	05200018 05200019
3/15/2016	ML15337A529	Request for Concurrence or Comments Regarding Potential Impacts to Federally Protected Resources Resulting from William States Lee III Nuclear Station, Units 1 and 2.	Letter	NRC/NRO/DNRL/EPB	US Dept of Interior, Fish & Wildlife Service	05200018 05200019
3/15/2016	ML16074A219	03/10/2016 Summary of Meeting with Members of the AP1000 Design Center to Discuss AP1000 Design Matters for the Levy Nuclear Plant Units 1 and 2 Combined License Application: AP1000 Generic License Condition Language	Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DSRA/SRSB	05200018 05200019 05200029 05200030 05200040

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Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
						05200041
3/22/2016	ML16084A099	Levy, Units 1 and 2, William States Lee, Units 1 and 2 - Acceptance of NRC License Conditions.	Letter	Duke Energy Florida, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019 05200029 05200030
3/22/2016	ML16084A100	Williams States Lee, Units 1 and 2, Levy, Units 1 and 2, Shearon Harris, Units 2 and 3 - Transmittal of 10 CFR 50.46 Annual Report for the AP1000 Standard Plant Design.	Annual Operating Report Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019 05200022 05200023 05200029 05200030

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Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
3/24/2016	ML16088A022	Williams States Lee, Units 1 and 2 - Resubmittal of Supplement 2 to Voluntary Submittal of Exemption Request and Design Change Description for Departure from AP1000 DCD Revision 19 to Address Containment Condensate Return Cooling Design.	Design Control Document (DCD) Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
4/1/2016	ML16076A184	02/05/2016-Summary of Meeting with Members of the AP1000 Design Center to Discuss AP1000 Licensing and Technical Issues.	Meeting Agenda Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200029 05200030 05200040 05200041
4/11/2016	ML16104A132	William States Lee III, Units 1 and 2 - AP1000 Combined License Application Update Roadmap.	Letter	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
4/11/2016	ML16124A672	Duke Energy WSL III Units 1 & 2 COLA (ITAAC) - Part 10, Conditions and ITAAC	Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019

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Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
4/11/2016	ML16124A674	Duke Energy WSL III Units 1 & 2 COLA (Technical Specifications) - Part 4, Williams Lee III Nuclear Station Technical Specifications	License-Application for Combined License (COLA) Technical Specifications	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
4/11/2016	ML16124A854	William States Lee III Nuclear Station, Update for Combined License Application.	License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/Document Control Desk NRC/NRO	05200018 05200019
4/11/2016	ML16124A642	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report) - FSAR Chapter 13 Figure 13.1-202	Final Safety Analysis Report (FSAR) License-Application for Combined License (COLA)	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
4/11/2016	ML16124A665	Duke Energy WSL III Units 1 & 2 COLA (Final Safety Analysis Report) Revision 11	COLA	Duke Energy Carolinas, LLC	NRC/NRO	05200018 05200019
4/19/2016	ML16013A374	December 17, 2015, Summary of a Public Teleconference with Members of the Ap1000 Design Center to Discuss the Ap1000 Design Center's Combined License Issues.	Meeting Agenda Meeting Summary Memoranda	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200029 05200030 05200040

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Document Date	Accession Number	Title	Document Type	Author Affiliation	Addressee Affiliation	Docket Number
						05200041
5/13/2016	ML16091A415	Summary Report Of The Audit Of The Results Of The Duke Process For Identifying New And Potentially Significant Information Related To The Levy Nuclear Plants Project Units 1 and 2 And The William States Lee III Nuclear Station Units 1 and 2 Environmental	Audit Report Memoranda Report, Miscellaneous	NRC/NRO/DNRL/EPB	NRC/NRO/DNRL/EPB	05200018 05200019 05200029 05200030
5/27/2016	ML16145A265	05/12/2016 Summary of Meeting With Members of the AP1000 Design Center on AP1000 Design Center's Combined License Issues.	Meeting Summary	NRC/NRO/DNRL/LB4	NRC/NRO/DNRL/LB4	05200018 05200019 05200029 05200030 05200040 05200041
6/20/2016	ML16111B263	Inimicality Review Letter: William States Lee Nuclear Plant, Units 1 and 2 Combined License Application.	Memoranda	NRC/NSIR	NRC/NRO	05200018 05200019

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Appendix C Electronic Request for Additional Information Database

Throughout the course of the review of the William States Lee III Nuclear Station (WLS) combined license (COL) application, the staff requested additional information (RAIs) of Duke Energy Carolinas, LCC (DEC). The following is a list of these RAIs and the responses.

As noted in Section 1.2.3 of this report, a design-centered review approach (DCRA) was used in the review of the WLS COL application. The first COL application submitted for NRC staff review in a design center is designated as the reference COL (RCOL), and the subsequent applications in the design center are designated as subsequent COL (SCOL) applications. The Bellefonte Nuclear Plant (BLN) Units 3 and 4 COL application was originally designated as the RCOL application for the AP1000 design center, and the staff issued a safety evaluation report (SER) with open items that documented its review of both standard and site-specific information (for all chapters except Sections 3.7, 3.8, 13.6, 13.7, and 13.8 and Appendix 19A). The RCOL for the AP1000 COL design center switched from the Bellefonte COL application to the Vogtle COL application after the issuance of the Bellefonte SER with open items. The LNP COL application has been designated as an SCOL. Therefore, in addition to the list of RAIs that follows that are based on site-specific information, DEF had to endorse RAI responses from the RCOLs (both Bellefonte and Vogtle) that were determined to be standard to the AP1000 COL design center. The endorsement of these standard RAIs can be found in the following letters:

- Summary Identification of Concurrence with Standard Content in Response to Requests for Additional Information, dated February 5, 2009, ADAMS accession number ML090400619. This letter provides endorsement of standard responses that were provided in a Tennessee Valley Authority (TVA) letter dated October 24, 2008.

- Summary Identification of Concurrence with Standard Content in Response to Requests for Additional Information, dated May 22, 2009, ADAMS accession number ML091470055. This letter provides endorsement of standard responses that were provided in a TVA letter dated April 15, 2009, which supplements the TVA letter dated October 24, 2008.

- Summary Identification of Concurrence with Standard Content RAIs and Safety Evaluation Report Open Items dated December 18, 2009, ML093570280. This letter provides endorsement of standard responses that were provided in a TVA letter dated November 16, 2009 and a SNC letter dated November 20, 2009.

- Endorsement of Vogtle R-COLA Response to BLN SER Confirmatory Item CI 04.04-01, dated June 11, 2010, ADAMS accession number ML101660701. This letter provides endorsement to the response to Confirmatory Item 04.04-01 that was provided in an SNC letter dated January 8, 2010.

- Endorsement of Vogtle Electric Generating Plant – Units 3 and 4 Response to Request for Additional Information Letters No. 04 04 “Mitigating Strategies for Beyond Design Basis External Events” dated February 17, 2016, ML16050A173. This letter provides endorsement of the SNC RAI responses that were provided in an SNC letters dated December 4, 2014 and February 26, 2015 and reference to a supplement to an exemption request and endorsement in a DEC letter dated February 17, 2016.

- Summary Identification of Concurrence with Standard Content RAIs and Safety

Evaluation Report Open Items dated November 4, 2010, ML103130134. This letter provides endorsement of standard responses that were provided in TVA letter dated August 26, 2010 and a SNC letter dated August 23, 2010.

- Concurrence with Standard Content Related to Unidentified RCS Leakage, dated March 14, 2011, ML110760143. This letter provides endorsement of revisions to the standard content that were provided in a SNC letter dated August 5, 2010.

- Concurrence with Standard Content Regarding Safety-Related Coatings, dated March 24, 2011, ML110840709. This letter provides endorsement of the standard content that was provided in the SNC letter dated August 13, 2010.

- Concurrence with Standard Content Related to Initial Test Program License Conditions dated February 4, 2011, ML110390048. This letter provides endorsement of the standard content that was provided in a SNC letter dated October 15, 2010.

- Concurrence with Standard Content Related to Qualifications of Test Engineers dated February 4, 2011, ML110390049. This letter provides endorsement of the standard content that was provided in a SNC letter dated November 11, 2010.

- Concurrence with Standard Content Related to Augmented Inservice Inspection dated March 14, 2011, ML110760144. This letter provides endorsement of the standard content that was provided in SNC letter dated August 27, 2010.

- Concurrence with Standard Content Related to Caldon LEFM CheckPlus System dated March 24, 2011, ML110840710. This letter provides endorsement of the standard content that was provided in SNC letters dated August 6, 2010, October 29, 2010, and February 8, 2011.

- Summary Identification of Concurrence with Standard Content RAIs and Safety Evaluation Report Open Items, dated April 25, 2011, ADAMS accession number ML11116A162. This letter provides endorsement of standard responses that were provided in an SNC letter dated March 31, 2011.

- Concurrence with Standard Content Regarding In-Transit Requirements for New Fuel Shipments dated May 18, 2011, ADAMS accession number ML11139A409. This letter provides endorsement of the revisions that were provided in aSNC letter dated May 6, 2011.

Concurrence with Standard Content Regarding Supplemental Information in Support of a Special Nuclear Material License Application and a Voluntary Revision to Part 10, dated July 28, 2011, ML11214A028. This letter provides endorsement of standard responses that were provided in a SNC letters dated June 22, 2011 and May 13, 2011.
The following notes pertain to the table on the proceeding following pages:

- The request for additional information (RAI) question numbers were assigned based on the section of the Standard Review Plan (SRP) that was associated with the question (e.g., question 02.01.02-1 was generated based on the staff’s review of the application against Section 2.1.2 of the SRP).

- The applicant's responses to security-related and sensitive information questions (e.g., physical security) are not publically available.

FSER Appendix C Report

Application Title: William States Lee III, Units 1 and 2 - Dockets 52-018 and 52-019

Question No.	NRC Letter No.	System RAI No.	SRP Section Title	RAI Issued Date	RAI Accession Number	Response Date	Response Accession Number
01.05-1	105	6419	Other Regulatory Considerations	4/25/12	ML12116A336	2/28/14	ML14065A024
01.05-1	105	6419	Other Regulatory Considerations	4/25/12	ML12116A336	1/30/14	ML14064A435
01.05-1	105	6419	Other Regulatory Considerations	4/25/12	ML12116A336	6/11/12	ML12165A423
01.05-1	105	6419	Other Regulatory Considerations	4/25/12	ML12116A336	6/11/12	ML12165A423
01.05-2	105	6419	Other Regulatory Considerations	4/25/12	ML12116A336	2/28/14	ML14065A024
01.05-3	105	6419	Other Regulatory Considerations	4/25/12	ML12116A336	6/11/12	ML12165A423
01.05-3	105	6419	Other Regulatory Considerations	4/25/12	ML12116A336	6/20/12	ML12174A272
01.05-4	105	6419	Other Regulatory Considerations	4/25/12	ML12116A336	2/28/14	ML14065A024
01.05-4	105	6419	Other Regulatory Considerations	4/25/12	ML12116A336	9/24/09	ML092710230
01-1	7	663	Introduction and Interfaces	8/21/08	ML082340944	9/17/08	ML082630541
01-2	8	741	Ultimate Heat Sink	8/28/08	ML082410291	9/26/08	ML082750079
01-3	42	777	Introduction and Interfaces	10/22/08	ML082960154	12/3/08	ML083440291
01-4	58	1760	Introduction and Interfaces	1/21/09	ML090210325	2/20/09	ML090560406
01-5	58	1760	Introduction and Interfaces	1/21/09	ML090210325	2/20/09	ML090560406
01-6	58	1760	Introduction and Interfaces	1/21/09	ML090210325	2/20/09	ML090560406
01-7	58	1760	Introduction and Interfaces	1/21/09	ML090210325	2/20/09	ML090560406
01-8	75	3208	Introduction and Interfaces	8/5/09	ML092170363	9/24/09	ML092710230
02.01.03-1	23	835	Population Distribution	9/23/08	ML082671058	4/18/82	ML083090781
02.02.01-02.02.02-1	36	836	2.02.02 - Identification of Potential Hazards in Site Vicinity	10/6/08	ML082800341	11/7/08	ML083180157
02.02.01-02.02.02-2	36	836	2.02.02 - Identification of Potential Hazards in Site Vicinity	10/6/08	ML082800341	11/7/08	ML083180157
02.02.03-1	43	837	Evaluation of Potential Accidents	10/22/08	ML082960488	12/23/08	ML083660212
02.02.03-2	43	837	Evaluation of Potential Accidents	10/22/08	ML082960488	12/23/08	ML083660212
02.02.03-3	43	837	Evaluation of Potential Accidents	10/22/08	ML082960488	12/23/08	ML083660212
02.02.03-4	43	837	Evaluation of Potential Accidents	10/22/08	ML082960488	12/23/08	ML083640466
02.02.03-5	43	837	Evaluation of Potential Accidents	10/22/08	ML082960488	12/23/08	ML083640466
02.02.03-6	43	837	Evaluation of Potential Accidents	10/22/08	ML082960488	12/23/08	ML083660212
02.02.03-7	43	837	Evaluation of Potential Accidents	10/22/08	ML082960488	12/23/08	ML083640466

02.02.03-8	103	6339	Evaluation of Potential Accidents	2/27/12	ML12059A492	3/28/12	ML12090A052
02.03.01-1	10	446	Regional Climatology	9/11/08	ML082550296	10/10/08	ML082890416
02.03.01-10	10	446	Regional Climatology	9/11/08	ML082550296	10/10/08	ML082890416
02.03.01-11	81	3798	Regional Climatology	10/29/09	ML09050010	4/14/10	ML101090072
02.03.01-2	10	446	Regional Climatology	9/11/08	ML082550296	10/10/08	ML082890416
02.03.01-3	10	446	Regional Climatology	9/11/08	ML082550296	10/10/08	ML082890416
02.03.01-4	10	446	Regional Climatology	9/11/08	ML082550296	10/10/08	ML082890416
02.03.01-5	10	446	Regional Climatology	9/11/08	ML082550296	10/10/08	ML082890416
02.03.01-6	10	446	Regional Climatology	9/11/08	ML082550296	10/10/08	ML082890416
02.03.01-7	10	446	Regional Climatology	9/11/08	ML082550296	10/10/08	ML082890416
02.03.01-8	10	446	Regional Climatology	9/11/08	ML082550296	10/10/08	ML082890416
02.03.01-9	10	446	Regional Climatology	9/11/08	ML082550296	10/10/08	ML082890416
02.03.02-1	11	447	Local Meteorology	9/12/08	ML082890416	11/25/08	ML083360557
02.03.02-1	11	447	Local Meteorology	9/12/08	ML082890416	5/2/13	ML13127A225
02.03.02-10	81	3799	Local Meteorology	10/29/09	ML093050010	4/14/10	ML101090072
02.03.02-10	81	3799	Local Meteorology	10/29/09	ML093050010	5/2/13	ML13127A225
02.03.02-11	82	3800	Local Meteorology	10/29/09	ML093050020	3/23/10	ML100850543
02.03.02-12	90	4959	Local Meteorology	8/12/10	ML102240279	11/22/11	ML11327A146
02.03.02-2	11	447	Local Meteorology	9/12/08	ML082890416	11/25/08	ML083360557
02.03.02-3	11	447	Local Meteorology	9/12/08	ML082890416	11/25/08	ML083360557
02.03.02-4	11	447	Local Meteorology	9/12/08	ML082890416	11/25/08	ML083360557
02.03.02-5	11	447	Local Meteorology	9/12/08	ML082890416	11/25/08	ML083360557
02.03.02-6	11	447	Local Meteorology	9/12/08	ML082890416	11/25/08	ML083360557
02.03.02-7	11	447	Local Meteorology	9/12/08	ML082890416	11/25/08	ML083360557
02.03.02-8	11	447	Local Meteorology	9/12/08	ML082890416	11/25/08	ML083360557
02.03.02-9	11	447	Local Meteorology	9/12/08	ML082890416	11/25/08	ML083360557
02.03.03-1	47	448	Onsite Meteorological Measurements Programs	11/5/08	ML083100209	12/17/08	ML083590244
02.03.03-3	47	448	Onsite Meteorological Measurements Programs	11/5/08	ML083100209	12/17/08	ML083590244
02.03.03-3	47	448	Onsite Meteorological Measurements Programs	11/5/08	ML083100209	5/2/13	ML13127A225
02.03.03-4	104	6357	Onsite Meteorological Measurements Programs	4/3/12	ML12096A033	5/2/12	ML12124A282
02.03.03-4	104	6357	Onsite Meteorological Measurements Programs	4/3/12	ML083100209	5/2/13	ML13127A225
02.03.03-5	104	6357	Onsite Meteorological Measurements Programs	4/3/12	ML12096A033	5/2/12	ML12124A282
02.03.04-1	12	449	Short Term Atmospheric Dispersion Estimates for Accident Releases	9/12/08	ML082560187	10/10/08	ML082910110
02.03.04-2	12	449	Short Term Atmospheric Dispersion Estimates for Accident Releases	9/12/08	ML082560187	10/10/08	ML082910110
02.03.04-3	12	449	Short Term Atmospheric Dispersion Estimates for Accident Releases	9/12/08	ML082560187	10/10/08	ML082910110
02.03.04-4	80	3726	Short Term Atmospheric Dispersion Estimates for Accident Releases	10/29/09	ML093050007	4/6/10	ML101060138
02.03.04-4	80	3726	Short Term Atmospheric Dispersion Estimates for Accident Releases	10/29/09	ML093050007	5/2/13	ML13127A225
02.03.04-5	80	3726	Short Term Atmospheric Dispersion Estimates for Accident Releases	10/29/09	ML093050007	5/2/13	ML13127A225
02.03.04-5	80	3726	Short Term Atmospheric Dispersion Estimates for Accident Releases	10/29/09	ML093050007	4/6/10	ML101060138

02.03.04-6	92	4961	Short Term Atmospheric Dispersion Estimates for Accident Releases	8/23/10	ML102360015	5/2/13	ML13127A225
02.03.04-6	92	4961	Short Term Atmospheric Dispersion Estimates for Accident Releases	8/23/10	ML102360015	9/28/10	ML102740218
02.03.05-1	40	451	Long-Term Atmospheric Dispersion Estimates for Routine Releases	10/20/08	ML082940494	12/17/08	ML083650408
02.03.05-2	40	451	Long-Term Atmospheric Dispersion Estimates for Routine Releases	10/20/08	ML082940494	12/17/08	ML083650408
02.03.05-3	40	451	Long-Term Atmospheric Dispersion Estimates for Routine Releases	10/20/08	ML082940494	12/17/08	ML083650408
02.03.05-4	80	3727	Long-Term Atmospheric Dispersion Estimates for Routine Releases	10/29/09	ML093050007	4/6/10	ML101060138
02.03.05-4	80	3727	Long-Term Atmospheric Dispersion Estimates for Routine Releases	10/29/09	ML093050007	5/2/13	ML13127A225
02.03.05-5	91	4960	Long-Term Atmospheric Dispersion Estimates for Routine Releases	8/12/10	ML102240279	9/16/10	ML102640040
02.03.05-5	91	4960	Long-Term Atmospheric Dispersion Estimates for Routine Releases	8/12/10	ML102240279	5/2/13	ML13127A225
02.03.05-6	110	7186	Long-Term Atmospheric Dispersion Estimates for Routine Releases	7/30/13	ML13211A231	8/29/13	ML13248A105
02.03.05-6	110	7186	Long-Term Atmospheric Dispersion Estimates for Routine Releases	7/30/13	ML13211A231	9/30/13	ML13283A227
02.04.01-1	28	818	Hydrologic Description	10/3/08	ML082770560	11/18/08	ML083290333
02.04.01-2	28	818	Hydrologic Description	10/3/08	ML082770560	11/18/08	ML083290333
02.04.01-3	28	818	Hydrologic Description	10/3/08	ML082770560	11/18/08	ML083290333
02.04.01-3	28	818	Hydrologic Description	10/3/08	ML082770560	5/2/13	ML13127A225
02.04.01-4	28	818	Hydrologic Description	10/3/08	ML082770560	5/2/13	ML13127A225
02.04.01-4	28	818	Hydrologic Description	10/3/08	ML082770560	11/18/08	ML083290333
02.04.01-5	28	818	Hydrologic Description	10/3/08	ML082770560	11/18/08	ML083290333
02.04.01-6	28	818	Hydrologic Description	10/3/08	ML082770560	11/18/08	ML083290333
02.04.01-7	28	818	Hydrologic Description	10/3/08	ML082770560	11/18/08	ML083290333
02.04.01-8	28	818	Hydrologic Description	10/3/08	ML082770560	11/18/08	ML083290333
02.04.02-1	17	820	Floods	9/22/08	ML082660247	10/27/08	ML083040525
02.04.02-2	17	820	Floods	9/22/08	ML082660247	10/27/08	ML083040525
02.04.02-2	17	820	Floods	9/22/08	ML082660247	5/2/13	ML13127A225
02.04.02-3	72	2679	Floods	6/8/09	ML091730493	7/17/09	ML092030129
02.04.02-3	72	2679	Floods	6/8/09	ML091730493	5/2/13	ML13127A225
02.04.03-1	17	821	Probable Maximum Flood (PMF) on Streams and Rivers	9/22/08	ML082660247	10/27/08	ML083040525
02.04.03-10	69	2680	Probable Maximum Flood (PMF) on Streams and Rivers	5/27/09	ML091470226	6/9/09	ML091750090
02.04.03-10	69	2680	Probable Maximum Flood (PMF) on Streams and Rivers	5/27/09	ML091470226	5/2/13	ML13127A225
02.04.03-2	17	821	Probable Maximum Flood (PMF) on Streams and Rivers	9/22/08	ML082660247	10/27/08	ML083040525
02.04.03-3	17	821	Probable Maximum Flood (PMF) on Streams and Rivers	9/22/08	ML082660247	10/27/08	ML083040525

02.04.03-4	17	821	Probable Maximum Flood (PMF) on Streams and Rivers	9/22/08	ML082660247	10/27/08	ML083040525
02.04.03-4	17	821	Probable Maximum Flood (PMF) on Streams and Rivers	9/22/08	ML082660247	5/2/13	Ml13127A225
02.04.03-5	17	821	Probable Maximum Flood (PMF) on Streams and Rivers	9/22/08	ML082660247	5/2/13	ML13127A225
02.04.03-5	17	821	Probable Maximum Flood (PMF) on Streams and Rivers	9/22/08	ML082660247	10/27/08	ML083040525
02.04.03-6	69	2680	Probable Maximum Flood (PMF) on Streams and Rivers	5/27/09	ML091470226	6/9/09	ML091750090
02.04.03-7	69	2680	Probable Maximum Flood (PMF) on Streams and Rivers	5/27/09	ML091470226	6/9/09	ML091750090
02.04.03-8	69	2680	Probable Maximum Flood (PMF) on Streams and Rivers	5/27/09	ML091470226	6/9/09	ML091750090
02.04.03-9	69	2680	Probable Maximum Flood (PMF) on Streams and Rivers	5/27/09	ML091470226	6/9/09	ML091750090
02.04.04-1	17	822	Potential Dam Failures	9/22/08	ML082660247	10/27/08	ML083040525
02.04.04-2	17	822	Potential Dam Failures	9/22/08	ML082660247	10/27/08	ML083040525
02.04.04-2	17	822	Potential Dam Failures	9/22/08	ML082660247	5/2/13	ML13127A225
02.04.05-1	17	823	Probable Maximum Surge and Seiche Flooding	9/22/08	ML082660247	10/27/08	ML083040525
02.04.05-2	17	823	Probable Maximum Surge and Seiche Flooding	9/22/08	ML082660247	10/27/08	ML083040525
02.04.06-1	17	824	Probable Maximum Tsunami Flooding	9/22/08	ML082660247	10/27/08	ML083040525
02.04.06-2	17	824	Probable Maximum Tsunami Flooding	9/22/08	ML082660247	10/27/08	ML083040525
02.04.11-1	17	825	Low Water Considerations	9/19/08	ML082660247	10/27/08	ML083040525
02.04.11-2	17	825	Low Water Considerations	9/19/08	ML082660247	10/27/08	ML083040525
02.04.12-1	17	826	Groundwater	9/19/08	ML082660247	12/11/08	ML083520336
02.04.12-10	17	826	Groundwater	9/19/08	ML082660247	12/11/08	ML083520336
02.04.12-11	17	826	Groundwater	9/19/08	ML082660247	12/11/08	ML083520336
02.04.12-12	17	826	Groundwater	9/19/08	ML082660247	12/11/08	ML083520336
02.04.12-13	17	826	Groundwater	9/19/08	ML082660247	12/11/08	ML083520336
02.04.12-14	17	826	Groundwater	9/19/08	ML082660247	5/12/09	ML091340410
02.04.12-15	70	2685	Groundwater	6/16/09	ML091690038	12/18/09	ML093570129
02.04.12-16	70	2685	Groundwater	6/16/09	ML091690038	7/31/09	ML092170378
02.04.12-17	70	2685	Groundwater	6/16/09	ML091690038	7/31/09	ML092170378
02.04.12-18	70	2685	Groundwater	6/16/09	ML091690038	7/31/09	ML092170378
02.04.12-19	91	4870	Groundwater	8/25/10	ML102371060	9/30/10	ML102770372
02.04.12-2	17	826	Groundwater	9/19/08	ML082660247	12/11/08	ML083520336
02.04.12-20	94	5507	Groundwater	4/13/11	ML111030569	11/22/11	ML11332A156
02.04.12-20	94	5507	Groundwater	4/13/11	ML111030569	5/2/13	ML13127A225
02.04.12-21	94	5507	Groundwater	4/13/11	ML111030569	5/2/13	ML13127A225
02.04.12-21	94	5507	Groundwater	4/13/11	ML111030569	5/18/11	ML11139A408
02.04.12-3	17	826	Groundwater	9/19/08	ML082660247	12/11/08	ML083520336
02.04.12-4	17	826	Groundwater	9/19/08	ML082660247	5/12/09	ML091340410
02.04.12-5	17	826	Groundwater	9/19/08	ML082660247	12/11/08	ML083520336
02.04.12-6	17	826	Groundwater	9/19/08	ML082660247	12/11/08	ML083520336
02.04.12-7	17	826	Groundwater	9/19/08	ML082660247	12/11/08	ML083520336
02.04.12-8	17	826	Groundwater	9/19/08	ML082660247	12/11/08	ML083520336

02.04.12-9	17	826	Groundwater	9/19/08	ML082660247	12/11/08	ML083520336
02.04.13-1	34	722	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	9/19/08	ML082630116	12/3/08	ML083440292
02.04.13-10	17	828	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	9/19/08	ML082660247	11/25/08	ML083360506
02.04.13-11	17	828	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	9/19/08	ML082660247	11/25/08	ML083360506
02.04.13-12	17	828	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	9/19/08	ML082660247	11/25/08	ML083360506
02.04.13-13	17	828	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	9/19/08	ML082660247	11/25/08	ML083360506
02.04.13-14	17	828	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	9/19/08	ML082660247	11/25/08	ML083360506
02.04.13-15	17	828	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	9/19/08	ML082660247	11/25/08	ML083360506
02.04.13-16	17	828	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	9/19/08	ML082660247	11/25/08	ML083360506
02.04.13-17	17	828	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	9/19/08	ML082660247	11/25/08	ML083360506
02.04.13-18	17	828	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	9/19/08	ML082660247	11/25/08	ML083360506
02.04.13-19	73	2686	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	6/30/09	ML091970441	12/12/09	ML093210477
02.04.13-2	34	721	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	9/19/08	ML082630116	12/3/08	ML083440292
02.04.13-20	73	2686	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	6/30/09	ML091970441	12/12/09	ML093210477
02.04.13-21	73	2686	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	6/30/09	ML091970441	12/12/09	ML093210477
02.04.13-22	73	2686	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	6/30/09	ML091970441	12/12/09	ML093210477
02.04.13-23	73	2686	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	6/30/09	ML091970441	12/12/09	ML093210477
02.04.13-24	73	2686	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	6/30/09	ML091970441	12/12/09	ML093210477

02.04.13-25	73	2686	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	6/30/09	ML091970441	12/12/09	ML093210477
02.04.13-26	73	2686	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	6/30/09	ML091970441	12/12/09	ML093210477
02.04.13-27	73	2686	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	6/30/09	ML091970441	12/12/09	ML093210477
02.04.13-28	73	2686	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	6/30/09	ML091970441	12/12/09	ML093210477
02.04.13-29	73	2686	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	6/30/09	ML091970441	12/12/09	ML093210477
02.04.13-3	17	828	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	9/19/08	ML082660247	11/25/08	ML083360506
02.04.13-4	17	828	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	9/19/08	ML082660247	11/25/08	ML083360506
02.04.13-4	17	828	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	9/19/08	ML082660247		ML13127A225
02.04.13-5	17	828	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	9/19/08	ML082660247	11/25/08	ML083360506
02.04.13-6	17	828	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	9/19/08	ML082660247	11/25/08	ML083360506
02.04.13-6	17	828	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	9/19/08	ML082660247	5/2/13	ML13127A225
02.04.13-7	17	828	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	9/19/08	ML082660247	5/2/13	ML13127A225
02.04.13-7	17	828	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	9/19/08	ML082660247	11/25/08	ML083360506
02.04.13-8	17	828	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	9/19/08	ML082660247	11/25/08	ML083360506
02.04.13-9	17	828	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	9/19/08	ML082660247	11/25/08	ML083360506
02.04.13-9	17	828	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	9/19/08	ML082660247	5/2/13	ML13127A225
02.05.01-1	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-10	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-11	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603

02.05.01-12	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-13	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-14	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-15	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-16	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-17	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-18	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-19	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-2	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-20	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-21	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-22	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-23	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-24	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-25	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-26	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-27	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-28	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-29	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-3	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-30	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-30	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/2/13	ML13127A226
02.05.01-31	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-32	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-33	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-34	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-35	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-36	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-36	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/2/13	ML13127A226
02.05.01-37	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-38	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-39	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-4	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-40	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-41	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603

02.05.01-42	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-43	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-44	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-45	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-45	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/2/13	ML13127A226
02.05.01-46	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-47	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-48	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-48	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/2/13	ML13127A226
02.05.01-49	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-5	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-50	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-51	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-51	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/2/13	ML13127A226
02.05.01-52	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-53	86	3584	Basic Geologic and Seismic Information	11/4/09	ML093090013	12/11/09	ML093490249
02.05.01-54	86	3584	Basic Geologic and Seismic Information	11/4/09	ML093090013	12/11/09	ML093490249
02.05.01-55	86	3584	Basic Geologic and Seismic Information	11/4/09	ML093090013	12/11/09	ML093490249
02.05.01-6	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-7	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-8	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.01-9	59	1657	Basic Geologic and Seismic Information	1/8/09	ML090080767	5/21/08	ML091480603
02.05.02-1	27	1141	Vibratory Ground Motion	10/3/08	ML082770560	3/4/09	ML090680359
02.05.02-10	55	1244	Vibratory Ground Motion	12/3/08	ML083380306	3/9/09	ML090700576
02.05.02-11	55	1244	Vibratory Ground Motion	12/3/08	ML083380306	3/9/09	ML090700576
02.05.02-12	55	1244	Vibratory Ground Motion	12/3/08	ML083380306	3/9/09	ML090700576
02.05.02-13	55	1244	Vibratory Ground Motion	12/3/08	ML083380306	3/9/09	ML090700576
02.05.02-14	55	1244	Vibratory Ground Motion	12/3/08	ML083380306	3/9/09	ML090700576
02.05.02-15	55	1244	Vibratory Ground Motion	12/3/08	ML083380306	3/9/09	ML090700576
02.05.02-16	55	1244	Vibratory Ground Motion	12/3/08	ML083380306	3/9/09	ML090700576
02.05.02-17	55	1244	Vibratory Ground Motion	12/3/08	ML083380306	3/9/09	ML090700576
02.05.02-18	55	1244	Vibratory Ground Motion	12/3/08	ML083380306	3/9/09	ML090700576
02.05.02-19	55	1244	Vibratory Ground Motion	12/3/08	ML083380306	3/9/09	ML090700576
02.05.02-2	27	1141	Vibratory Ground Motion	10/3/08	ML082770560	11/20/08	ML083360493
02.05.02-20	55	1244	Vibratory Ground Motion	12/3/08	ML083380306	3/9/09	ML090700576
02.05.02-21	55	1244	Vibratory Ground Motion	12/3/08	ML083380306	3/9/09	ML090700576
02.05.02-22	55	1244	Vibratory Ground Motion	12/3/08	ML083380306	3/9/09	ML090700576
02.05.02-23	55	1244	Vibratory Ground Motion	12/3/08	ML083380306	3/9/09	ML090700576
02.05.02-24	55	1244	Vibratory Ground Motion	12/3/08	ML083380306	3/9/09	ML090700576
02.05.02-25	55	1244	Vibratory Ground Motion	12/3/08	ML083380306	3/9/09	ML090700576
02.05.02-26	55	1244	Vibratory Ground Motion	12/3/08	ML083380306	3/9/09	ML090700576

02.05.02-27	55	1244	Vibratory Ground Motion	12/3/08	ML083380306	3/9/09	ML090700576
02.05.02-28	55	1244	Vibratory Ground Motion	12/3/08	ML083380306	3/9/09	ML090700576
02.05.02-29	55	1244	Vibratory Ground Motion	12/3/08	ML083380306	3/9/09	ML090700576
02.05.02-3	45	1116	Vibratory Ground Motion	10/23/08	ML082970177	11/24/08	ML083330284
02.05.02-30	55	1244	Vibratory Ground Motion	12/3/08	ML083380306	3/9/09	ML090700576
02.05.02-31	55	1244	Vibratory Ground Motion	12/3/08	ML083380306	3/9/09	ML090700576
02.05.02-32	55	1244	Vibratory Ground Motion	12/3/08	ML083380306	3/9/09	ML090700576
02.05.02-33	55	1244	Vibratory Ground Motion	12/3/08	ML083380306	3/9/09	ML090700576
02.05.02-34	55	1244	Vibratory Ground Motion	12/3/08	ML083380306	3/9/09	ML090700576
02.05.02-35	63	1487	Vibratory Ground Motion	1/14/09	ML090140271	4/17/09	ML091110030
02.05.02-36	63	1487	Vibratory Ground Motion	1/14/09	ML090140271	4/17/09	ML091110030
02.05.02-37	63	1487	Vibratory Ground Motion	1/14/09	ML090140271	4/17/09	ML091110030
02.05.02-38	63	1487	Vibratory Ground Motion	1/14/09	ML090140271	4/17/09	ML091110030
02.05.02-4	45	1116	Vibratory Ground Motion	10/23/08	ML082970177	11/24/08	ML083330284
02.05.02-41	63	1487	Vibratory Ground Motion	1/14/09	ML090140271	4/17/09	ML091110030
02.05.02-43	63	1487	Vibratory Ground Motion	1/14/09	ML090140271	4/17/09	ML091110030
02.05.02-44	63	1487	Vibratory Ground Motion	1/14/09	ML090140271	4/17/09	ML091110030
02.05.02-45	63	1487	Vibratory Ground Motion	1/14/09	ML090140271	4/17/09	ML091110030
02.05.02-46	63	1487	Vibratory Ground Motion	1/14/09	ML090140271	4/17/09	ML091110030
02.05.02-47	63	1487	Vibratory Ground Motion	1/14/09	ML090140271	4/17/09	ML091110030
02.05.02-48	51	1589	Vibratory Ground Motion	1/14/09	ML090140208	4/28/2009	ML091200402
02.05.02-49	76	3621	Vibratory Ground Motion	11/4/09	ML093070753	3/12/2010	ML100760097
02.05.02-5	49	1292	Vibratory Ground Motion	11/7/08	ML083120081	12/3/2008	ML083440291
02.05.02-50	84	3549	Vibratory Ground Motion	11/4/09	ML093080146	12/18/2009	ML093570262
02.05.02-51	84	3549	Vibratory Ground Motion	11/4/09	ML093080146	12/18/2009	ML093570262
02.05.02-52	84	3549	Vibratory Ground Motion	11/4/09	ML093080146	12/18/2009	ML093570262
02.05.02-53	117	7500	Vibratory Ground Motion	5/28/14	ML14148A001	7/14/2014	ML14195A018
02.05.02-6	55	1244	Vibratory Ground Motion	12/3/08	ML083380306	3/9/2009	ML090700576
02.05.02-7	55	1244	Vibratory Ground Motion	12/3/08	ML083380306	3/9/2009	ML090700576
02.05.02-8	55	1244	Vibratory Ground Motion	12/3/08	ML083380306	3/9/2009	ML090700576
02.05.02-9	55	1244	Vibratory Ground Motion	12/3/08	ML083380306	3/9/2009	ML090700576
02.05.03-1	26	1053	Surface Faulting	10/3/08	ML082770571	12/3/2008	ML083460382
02.05.03-10	65	2002	Surface Faulting	3/12/09	ML090710152	4/14/2009	ML091060499
02.05.03-11	85	3590	Surface Faulting	11/4/09	ML093080312	12/18/2009	ML093570131
02.05.03-12	85	3590	Surface Faulting	11/4/09	ML093080312	12/18/2009	ML093570131
02.05.03-13	85	3590	Surface Faulting	11/4/09	ML093080312	12/18/2009	ML093570131
02.05.03-2	26	1053	Surface Faulting	10/3/08	ML082770571	12/3/2008	ML083460382
02.05.03-3	26	1053	Surface Faulting	10/3/08	ML082770571	12/3/2008	ML083460382
02.05.03-4	26	1053	Surface Faulting	10/3/08	ML082770571	12/3/2008	ML083460382
02.05.03-5	26	1053	Surface Faulting	10/3/08	ML082770571	12/3/2008	ML083460382
02.05.03-6	26	1053	Surface Faulting	10/3/08	ML082770571	12/3/2008	ML083460382
02.05.03-7	26	1053	Surface Faulting	10/3/08	ML082770571	12/3/2008	ML083460382
02.05.03-8	26	1053	Surface Faulting	10/3/08	ML082770571	12/3/2008	ML083460382
02.05.03-9	26	1053	Surface Faulting	10/3/08	ML082770571	12/3/2008	ML083460382
02.05.04-1	44	1473	Stability of Subsurface Materials and Foundations	10/29/08	ML083030282	10/30/2009	ML093080101
02.05.04-1	44	1473	Stability of Subsurface Materials and Foundations	10/29/08	ML083030282	5/2/2013	ML13127A227
02.05.04-10	60	1881	Stability of Subsurface Materials and Foundations	1/9/09	ML090090150	10/30/2009	ML093080101

02.05.04-10	60	1881	Stability of Subsurface Materials and Foundations	1/9/09	ML090090150	5/2/2013	ML13127A227
02.05.04-11	61	1874	Stability of Subsurface Materials and Foundations	1/12/09	ML090120621	10/30/2009	ML093080101
02.05.04-11	61	1874	Stability of Subsurface Materials and Foundations	1/12/09	ML090120621	5/2/2013	ML13127A227
02.05.04-12	61	1874	Stability of Subsurface Materials and Foundations	1/12/09	ML090120621	10/30/2009	ML13127A227
02.05.04-12	61	1874	Stability of Subsurface Materials and Foundations	1/12/09	ML090120621	5/2/2013	ML093080101
02.05.04-13	61	1874	Stability of Subsurface Materials and Foundations	1/12/09	ML090120621	10/30/2009	ML093080101
02.05.04-13	61	1874	Stability of Subsurface Materials and Foundations	1/12/09	ML090120621	5/2/2013	ML13127A227
02.05.04-14	61	1874	Stability of Subsurface Materials and Foundations	1/12/09	ML090120621	10/30/2009	ML093080101
02.05.04-14	61	1874	Stability of Subsurface Materials and Foundations	1/12/09	ML090120621	5/2/2013	ML13127A227
02.05.04-15	66	2098	Stability of Subsurface Materials and Foundations	3/20/09	ML090820014	10/30/2009	ML093080101
02.05.04-15	66	2098	Stability of Subsurface Materials and Foundations	3/20/09	ML090820014	5/2/2013	ML13127A227
02.05.04-16	95	5417	Stability of Subsurface Materials and Foundations	2/23/11	ML110560443	3/17/2011	ML110800597
02.05.04-17	106	6497	Stability of Subsurface Materials and Foundations	5/17/12	ML12138A118	6/18/2012	ML12171A367
02.05.04-17	106	6497	Stability of Subsurface Materials and Foundations	5/17/12	ML12138A118	5/2/2013	ML13127A227
02.05.04-18	112	7436	Stability of Subsurface Materials and Foundations	3/12/14	ML14071A521	4/11/2014	ML14104A021
02.05.04-19	112	7436	Stability of Subsurface Materials and Foundations	3/12/14	ML14071A521	4/24/14	ML14115A329
02.05.04-2	44	1473	Stability of Subsurface Materials and Foundations	10/29/08	ML083030282	10/30/2009	ML13127A227
02.05.04-2	44	1473	Stability of Subsurface Materials and Foundations	10/29/08	ML083030282	5/2/2013	ML093080101
02.05.04-20	112	7436	Stability of Subsurface Materials and Foundations	3/12/14	ML14071A521	4/24/14	ML14115A329
02.05.04-3	44	1473	Stability of Subsurface Materials and Foundations	10/29/08	ML083030282	12/23/2008	ML083659363
02.05.04-4	44	1473	Stability of Subsurface Materials and Foundations	10/29/08	ML083030282	10/30/2009	ML093080101
02.05.04-4	44	1473	Stability of Subsurface Materials and Foundations	10/29/08	ML083030282	5/2/2013	ML13127A227
02.05.04-5	44	1473	Stability of Subsurface Materials and Foundations	10/29/08	ML083030282	10/30/2009	ML13127A227
02.05.04-5	44	1473	Stability of Subsurface Materials and Foundations	10/29/08	ML083030282	5/2/2013	ML093080101
02.05.04-6	44	1473	Stability of Subsurface Materials and Foundations	10/29/08	ML083030282	12/23/2008	ML083659363
02.05.04-7	44	1473	Stability of Subsurface Materials and Foundations	10/29/08	ML083030282	12/23/2008	ML083659363
02.05.04-8	44	1473	Stability of Subsurface Materials and Foundations	10/29/08	ML083030282	10/30/2009	ML093080101
02.05.04-8	44	1473	Stability of Subsurface Materials and Foundations	10/29/08	ML083030282	5/2/2013	ML13127A227
02.05.04-8	44	1473	Stability of Subsurface Materials and Foundations	10/29/08	ML083030282	5/2/2013	ML13127A228

02.05.04-9	44	1473	Stability of Subsurface Materials and Foundations	10/29/08	ML083030282	12/23/2008	ML083659363
03.02.01-1	20	1020	Seismic Classification	9/22/08	ML082660248	10/17/2008	ML082950297
03.02.01-2	53	1395	Seismic Classification	11/12/08	ML083170329	11/20/2008	ML083300283
03.02.02-1	46	1021	System Quality Group Classification	10/31/08	ML083050245	11/20/2008	ML083300282
03.07.01-1	41	1003	Seismic Design Parameters	10/21/08	ML082950542	12/17/2008	ML083570396
03.07.01-2	41	1003	Seismic Design Parameters	10/21/08	ML082950542	10/30/2009	ML093080101
03.07.01-2	41	1003	Seismic Design Parameters	10/21/08	ML082950542	5/2/2013	ML13127A227
03.07.01-3	41	1003	Seismic Design Parameters	10/21/08	ML082950542	12/17/2008	ML083570396
03.07.01-4	67	2350	Seismic Design Parameters	3/20/09	ML090820071	8/24/2010	ML102380042
03.07.01-4	67	2350	Seismic Design Parameters	3/20/09	ML090820071	5/2/2013	ML13127A228
03.07.01-5	87	3644	Seismic Design Parameters	11/24/09	ML093280269	12/18/2009	ML093570286
03.07.01-5	87	3644	Seismic Design Parameters	11/24/09	ML093280269	5/2/2013	ML13127A228
03.07.01-6	118	7544	Seismic Design Parameters	6/9/14	ML141160B524	7/24/2014	ML14206A950
03.07.01-6	118	7544	Seismic Design Parameters	6/9/14	ML141160B524	7/24/2014	ML14206A950
03.07.01-7	119	7569	Seismic Design Parameters	7/1/14	ML14198A300	8/14/2014	ML14227A705
03.07.01-7	119	7569	Seismic Design Parameters	7/1/14	ML14198A300	8/14/2014	ML14227A705
03.07.02-1	100	6182	Seismic System Analysis	11/9/11	ML11313A213		
03.07.02-2	107	6528	Seismic System Analysis	5/23/12	ML12144A056	6/20/2012	ML12174A272
03.07.02-2	107	6528	Seismic System Analysis	5/23/12	ML12144A056	6/20/2012	ML12174A272
03.07.02-2	107	6528	Seismic System Analysis	5/23/12	ML12144A056	6/12/2012	ML12174A272
03.07.02-3	120	7570	Seismic System Analysis	7/11/14	ML14192B073	8/7/2014	ML14220A432
03.07.02-4	120	7570	Seismic System Analysis	7/11/14	ML14192B073	8/7/2014	ML14220A432
03.07.02-5	120	7570	Seismic System Analysis	7/11/14	ML14192B073		
03.08.05-1	41	1004	Foundations	10/21/08	ML082950542	12/17/2008	ML083570396
03.08.05-2	41	1004	Foundations	10/21/08	ML082950542	12/17/2008	ML083570396
03.08.05-2	41	1004	Foundations	10/21/08	ML082950542	5/2/2013	ML13127A228
03.08.05-3	41	1004	Foundations	10/21/08	ML082950542	12/17/2008	ML083570396
03.08.05-4	41	1004	Foundations	10/21/08	ML082950542	12/17/2008	ML083570396
03.08.05-5	68	2563	Foundations	4/15/09	ML091050662	6/11/2009	ML091660230
03.08.05-6	102	6183	Foundations	12/19/11	ML11355A063	1/17/2012	ML12018A388
03.08.05-7	121	7571	Foundations	7/15/14	ML14196A303	8/14/2014	ML14297A027
03.08.05-7	121	7571	Foundations	7/15/14	ML14196A303	8/14/2014	ML14297A027
03.08.05-7	121	7571	Foundations	7/15/14	ML14196A303	8/14/2014	ML14227A706
03.08.05-7	121	7571	Foundations	7/15/14	ML14196A303	10/22/2014	ML14227A706
03.08.05-7	121	7571	Foundations	7/15/14	ML14196A303	10/22/2014	ML14227A706
03.10-1	114	7482	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	5/7/14	ML14127A596	6/5/2014	ML14160A720
03.10-1	114	7482	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	5/7/14	ML14127A596	8/13/2014	ML14160A720
03.10-1	114	7482	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	5/7/14	ML14127A596	8/13/2014	ML14160A720
03.10-2	114	7482	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	5/7/14	ML14127A596	6/5/2014	ML14160A720
03.10-3	114	7482	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	5/7/14	ML14127A596	6/5/2014	ML14160A720
03.10-4	122	7573	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	7/14/14	ML141195A483	8/13/2014	ML14227A708

03.10-4	122	7573	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	7/14/14	ML141195A483	8/13/2014	ML14227A708
03.10-4	122	7573	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	7/14/14	ML141195A483	8/13/2014	ML14227A708
03.12-1	115	7510	ASME Code Class 1, 2, and 3 Piping Systems and Piping Components and TheirAssociated Supports	5/15/14	ML14135A538	11/10/2014	ML14163A572
03.12-1	115	7510	ASME Code Class 1, 2, and 3 Piping Systems and Piping Components and TheirAssociated Supports	5/15/14	ML14135A538	6/11/2014	ML14163A572
03.12-2	116	7539	ASME Code Class 1, 2, and 3 Piping Systems and Piping Components and TheirAssociated Supports	5/14/14	ML14134A573	6/11/2014	ML14163A571
03.12-2	116	7539	ASME Code Class 1, 2, and 3 Piping Systems and Piping Components and TheirAssociated Supports	5/14/14	ML14134A573	1/13/2015	ML15014A443
03.12-2	116	7539	ASME Code Class 1, 2, and 3 Piping Systems and Piping Components and TheirAssociated Supports	5/14/14	ML14134A573	1/13/2015	ML15014A443
06.04-1	19	907	Control Room Habitability System	9/22/08	ML082660246	12/23/2008	ML083660090
06.04-2	19	907	Control Room Habitability System	9/22/08	ML082660246	12/23/2008	ML083660090
06.04-3	19	907	Control Room Habitability System	9/22/08	ML082660246	12/23/2008	ML083660090
06.04-4	19	907	Control Room Habitability System	9/22/08	ML082660246	12/23/2008	ML083660090
06.04-5	19	907	Control Room Habitability System	9/22/08	ML082660246	12/23/2008	ML083660090
06.04-6	19	907	Control Room Habitability System	9/22/08	ML082660246	12/23/2008	ML083660090
06.04-7	19	908	Control Room Habitability System	9/22/08	ML082660246	5/31/2012	ML12156A212
06.04-8	19	925	Control Room Habitability System	9/22/08	ML082660246	10/27/2008	ML083040526
07.05-1	4	538	Information Systems Important to Safety	8/14/08	ML082270369	10/1/2009	ML092780249
08.02-1	1	666	Offsite Power System	8/6/08	ML082190373	9/5/2008	ML082530446
08.02-2	1	666	Offsite Power System	8/6/08	ML082190373	9/5/2008	ML082530446
08.02-3	5	431	Offsite Power System	8/27/08	ML082401031	9/26/2008	ML082750081
08.02-4	5	431	Offsite Power System	8/27/08	ML082401031	9/26/2008	ML082750081
08.02-5	5	431	Offsite Power System	8/27/08	ML082401031	9/26/2008	ML082750081
08.02-6	5	431	Offsite Power System	8/27/08	ML082401031	9/26/2008	ML082750081
08.02-7	5	431	Offsite Power System	8/27/08	ML082401031	9/26/2008	ML082750081
08.02-8	15	665	Offsite Power System	9/18/08	ML082620379	10/17/2008	ML082950294
08.02-9	74	3337	Offsite Power System	7/21/09	ML092020642	8/20/2009	ML092360176
03-Aug	108	6751	3 Branch Technical Position - Stability of Offsite Power Systems	8/14/13	ML13226A124	8/28/2014	ML14245A470
09.02.01-1	22	714	Station Service Water System	9/23/08	ML082671055	10/28/2008	ML083080067
09.02.01-2	22	714	Station Service Water System	9/23/08	ML082671055	10/28/2008	ML083080067
09.02.01-3	22	714	Station Service Water System	9/23/08	ML082671055	10/28/2008	ML083080067
09.02.01-4	22	714	Station Service Water System	9/23/08	ML082671055	10/28/2008	ML083080067
09.02.01-5	64	1922	Station Service Water System	1/28/09	ML090280416	5/15/2009	ML091400207

09.02.01-6	64	1922	Station Service Water System	1/28/09	ML090280416	11/19/2009	ML093280308
09.02.01-6	64	1922	Station Service Water System	1/28/09	ML090280416	4/1/2014	ML14090A560
09.02.01-7	64	1922	Station Service Water System	1/28/09	ML090280416	5/15/2009	ML091400207
09.02.01-8	93	5464	Station Service Water System	2/14/11	ML110450397	3/14/2011	ML110750044
09.02.02-1	38	1291	Reactor Auxiliary Cooling Water Systems	11/4/08	ML083090579	11/24/2008	ML083330285
09.03.03-1	89	4507	Equipment and Floor Drainage System	3/29/10	ML100880061	5/4/2010	ML101260121
09.05.01-1	9	222	Fire Protection Program	8/28/08	ML082410882	9/19/2008	ML082670701
09.05.01-2	9	222	Fire Protection Program	8/28/08	ML082410882	9/19/2008	ML082670701
09.05.01-3	9	222	Fire Protection Program	8/28/08	ML082410882	9/19/2008	ML082670701
09.05.01-4	57	1705	Fire Protection Program	12/8/08	ML083430017	12/17/2008	ML083540415
09.05.01-5	57	1705	Fire Protection Program	12/8/08	ML083430017	12/17/2008	ML083540415
09.05.02-1	21	662	Communications Systems	9/23/08	ML082670533	10/28/2008	ML083080077
09.05.02-2	21	662	Communications Systems	9/23/08	ML082670533	10/28/2008	ML083080077
09.05.02-3	21	662	Communications Systems	9/23/08	ML082670533	10/28/2008	ML083080077
09.05.02-3	21	662	Communications Systems	9/23/08	ML082670533	10/28/2008	ML083080077
10.04.05-1	2	483	Circulating Water System	8/11/08	ML082240647	9/10/2008	ML082560247
10.04.05-2	3	484	Circulating Water System	8/11/08	ML082240712	11/22/2011	ML11332A157
10.04.05-2	3	484	Circulating Water System	8/11/08	ML082240712	5/2/2013	ML13127A225
10.04.05-2	3	484	Circulating Water System	8/11/08	ML082240712	5/2/2013	ML13127A225
10.04.05-2	3	484	Circulating Water System	8/11/08	ML082240712	5/2/2013	ML13127A225
10.04.06-1	6	871	Condensate Cleanup System	8/18/08	ML082310699	9/17/2008	ML082630542
10.04.06-2	6	871	Condensate Cleanup System	8/18/08	ML082310699	9/17/2008	ML082630542
10.04.06-3	6	871	Condensate Cleanup System	8/18/08	ML082310699	9/17/2008	ML082630542
11.02-1	13	701	Liquid Waste Management System	9/16/08	ML082600101	12/11/2008	ML083520211
11.02-2	13	702	Liquid Waste Management System	9/16/08	ML082600101	12/11/2008	ML083520211
11.02-3	13	703	Liquid Waste Management System	9/16/08	ML082600101	12/11/2008	ML083520211
11.02-4	13	704	Liquid Waste Management System	9/16/08	ML082600101	12/11/2008	ML083520211
11.03-1	14	705	Gaseous Waste Management System	9/17/08	ML082610045	11/11/2008	ML083180221
11.03-1	14	705	Gaseous Waste Management System	9/17/08	ML082610045	5/2/2013	ML13127A225
11.03-2	14	717	Gaseous Waste Management System	9/17/08	ML082610045	10/17/2008	ml082950296
11.03-2	14	717	Gaseous Waste Management System	9/17/08	ML082610045	5/2/2013	ML13127A225
11.03-3	14	718	Gaseous Waste Management System	9/17/08	ML082610045	10/17/2008	ML082950296
11.03-4	109	7159	Gaseous Waste Management System	7/24/13	ML13205A093	8/8/2013	ML13239A054
12.03-12.04-1	24	1156	2.04 - Radiation Protection Design Features	9/26/08	ML082701011	12/4/2008	ML083440290
12.03-12.04-2	24	1162	2.04 - Radiation Protection Design Features	9/26/08	ML082701011	12/4/2008	ML083440290
12.03-12.04-2	24	1162	2.04 - Radiation Protection Design Features	9/26/08	ML082701011	5/2/2013	ML13127A225
13.01.01-1	29	918	Management and Technical Support Organization	10/3/08	ML082770744	11/7/2008	ML083180222
13.01.01-10	30	919	Management and Technical Support Organization	10/3/08	ML082770759	11/18/2008	ML083250484

13.01.01-2	29	918	Management and Technical Support Organization	10/3/08	ML082770744	11/7/2008	ML083180222
13.01.01-3	29	918	Management and Technical Support Organization	10/3/08	ML082770744	11/7/2008	ML08318022
13.01.01-4	29	918	Management and Technical Support Organization	10/3/08	ML082770744	11/7/2008	ML083180222
13.01.01-5	29	918	Management and Technical Support Organization	10/3/08	ML082770744	11/7/2008	ML083180222
13.01.01-6	29	918	Management and Technical Support Organization	10/3/08	ML082770744	11/7/2008	ML083180222
13.01.01-7	29	918	Management and Technical Support Organization	10/3/08	ML082770744	11/7/2008	ML083180222
13.01.01-8	30	919	Management and Technical Support Organization	10/3/08	ML082770759	11/25/2008	ML083330092
13.01.01-9	30	919	Management and Technical Support Organization	10/3/08	ML082770759	11/18/2008	ML083250484
13.01.02-13.01.03-1	31	920	3.01.03 - Operating Organization	10/3/08	ML082770760	11/4/2008	ML083110375
13.01.02-13.01.03-2	31	920	3.01.03 - Operating Organization	10/3/08	ML082770760	11/4/2008	ML083110375
13.01.02-13.01.03-3	31	920	3.01.03 - Operating Organization	10/3/08	ML082770760	11/4/2008	ML083110375
13.01.02-13.01.03-4	31	920	3.01.03 - Operating Organization	10/3/08	ML082770760	11/4/2008	ML083110375
13.03-1	25	50	Emergency Planning	9/25/08	ML082690889	11/7/2008	ML083180156
13.03-10	25	50	Emergency Planning	9/25/08	ML082690889	11/11/2008	ML083180158
13.03-100	123	7686	Emergency Planning	10/3/14	ML14279A219	11/6/2014	ML14314A035
13.03-101	123	7686	Emergency Planning	10/3/14	ML14279A219	11/6/2014	ML14314A035
13.03-102	123	7686	Emergency Planning	10/3/14	ML14279A219	11/6/2014	ML14314A035
13.03-103	123	7686	Emergency Planning	10/3/14	ML14279A219	11/6/2014	ML14314A035
13.03-104	123	7686	Emergency Planning	10/3/14	ML14279A219	11/6/2014	ML14314A035
13.03-11	25	50	Emergency Planning	9/25/08	ML082690889	11/11/2008	ML083180158
13.03-12	25	50	Emergency Planning	9/25/08	ML082690889	3/4/2010	ML100690444
13.03-13	25	50	Emergency Planning	9/25/08	ML082690889	11/11/2008	ML083180158
13.03-14	25	50	Emergency Planning	9/25/08	ML082690889	3/4/2010	ML100690444
13.03-15	25	50	Emergency Planning	9/25/08	ML082690889	3/4/2010	ML100690444
13.03-16	25	50	Emergency Planning	9/25/08	ML082690889	11/20/2008	ML083300288
13.03-17	25	50	Emergency Planning	9/25/08	ML082690889	11/20/2008	ML083300288
13.03-18	25	50	Emergency Planning	9/25/08	ML082690889	11/20/2008	ML083300288
13.03-19	25	50	Emergency Planning	9/25/08	ML082690889	11/20/2008	ML083300288
13.03-2	25	50	Emergency Planning	9/25/08	ML082690889	11/7/2008	ML083180156
13.03-20	25	50	Emergency Planning	9/25/08	ML082690889	11/20/2008	ML083300288
13.03-21	25	50	Emergency Planning	9/25/08	ML082690889	11/20/2008	ML083300288
13.03-22	25	50	Emergency Planning	9/25/08	ML082690889	11/20/2008	ML083300288
13.03-23	25	50	Emergency Planning	9/25/08	ML082690889	11/20/2008	ML083300288
13.03-24	25	50	Emergency Planning	9/25/08	ML082690889	11/20/2008	ML083300288
13.03-25	25	50	Emergency Planning	9/25/08	ML082690889	12/17/2008	ML083540416
13.03-26	25	50	Emergency Planning	9/25/08	ML082690889	3/4/2010	ML100690444
13.03-27	25	50	Emergency Planning	9/25/08	ML082690889	11/25/2008	ML090690313
13.03-28	25	50	Emergency Planning	9/25/08	ML082690889	3/4/2010	ML100690444
13.03-29	25	50	Emergency Planning	9/25/08	ML082690889	3/4/2010	ML100690444
13.03-3	25	50	Emergency Planning	9/25/08	ML082690889	11/7/2008	ML083180156

13.03-30	25	50	Emergency Planning	9/25/08	ML082690889	3/4/2010	ML100690444
13.03-31	25	50	Emergency Planning	9/25/08	ML082690889	3/4/2010	ML100690444
13.03-32	25	50	Emergency Planning	9/25/08	ML082690889	3/4/2010	ML100690444
13.03-33	25	50	Emergency Planning	9/25/08	ML082690889	3/4/2010	ML100690444
13.03-34	25	50	Emergency Planning	9/25/08	ML082690889	11/25/2008	ML090690313
13.03-35	25	50	Emergency Planning	9/25/08	ML082690889	3/4/2010	ML100690444
13.03-36	25	50	Emergency Planning	9/25/08	ML082690889	3/4/2010	ML100690444
13.03-37	25	50	Emergency Planning	9/25/08	ML082690889	3/4/2010	ML100690444
13.03-38	25	50	Emergency Planning	9/25/08	ML082690889	3/4/2010	ML100690444
13.03-39	25	50	Emergency Planning	9/25/08	ML082690889	12/9/2008	ML083460112
13.03-4	25	50	Emergency Planning	9/25/08	ML082690889	11/7/2008	ML083180156
13.03-40	25	50	Emergency Planning	9/25/08	ML082690889	11/24/2008	ML083450604
13.03-41	25	50	Emergency Planning	9/25/08	ML082690889	12/9/2008	ML083460112
13.03-42	25	50	Emergency Planning	9/25/08	ML082690889	12/9/2008	ML083460112
13.03-43	25	50	Emergency Planning	9/25/08	ML082690889	12/9/2008	ML083460112
13.03-44	25	50	Emergency Planning	9/25/08	ML082690889	11/24/2008	ML083450604
13.03-45	25	50	Emergency Planning	9/25/08	ML082690889	3/4/2010	ML100690444
13.03-46	25	50	Emergency Planning	9/25/08	ML082690889	12/9/2008	ML083460112
13.03-47	25	50	Emergency Planning	9/25/08	ML082690889	11/24/2008	ML083450604
13.03-48	25	50	Emergency Planning	9/25/08	ML082690889	3/4/2010	ML100690444
13.03-49	25	50	Emergency Planning	9/25/08	ML082690889	11/24/2008	ML083450604
13.03-5	25	50	Emergency Planning	9/25/08	ML082690889	11/7/2008	ML083180156
13.03-50	25	50	Emergency Planning	9/25/08	ML082690889	11/24/2008	ML083450604
13.03-51	25	50	Emergency Planning	9/25/08	ML082690889	11/24/2008	ML083450604
13.03-52	25	50	Emergency Planning	9/25/08	ML082690889	12/17/2008	ML083540416
13.03-53	25	50	Emergency Planning	9/25/08	ML082690889	12/9/2008	ML083460112
13.03-54	25	50	Emergency Planning	9/25/08	ML082690889	12/23/2008	ML090020175
13.03-55	25	50	Emergency Planning	9/25/08	ML082690889	12/23/2008	ML090020175
13.03-56	25	50	Emergency Planning	9/25/08	ML082690889	12/23/2008	ML083660272
13.03-57	25	50	Emergency Planning	9/25/08	ML082690889	12/17/2008	ML083540416
13.03-58	25	50	Emergency Planning	9/25/08	ML082690889	12/23/2008	ML083660272
13.03-59	25	50	Emergency Planning	9/25/08	ML082690889	12/23/2008	ML083660272
13.03-6	25	50	Emergency Planning	9/25/08	ML082690889	3/4/2010	ML100690444
13.03-60	25	50	Emergency Planning	9/25/08	ML082690889	12/23/2008	ML083660272
13.03-61	25	50	Emergency Planning	9/25/08	ML082690889	12/23/2008	ML090020175
13.03-61	25	50	Emergency Planning	9/25/08	ML082690889	3/16/2015	ML15077A176
13.03-62	25	50	Emergency Planning	9/25/08	ML082690889	12/23/2008	ML090020175
13.03-63	25	50	Emergency Planning	9/25/08	ML082690889	12/23/2008	ML090020175
13.03-64	25	50	Emergency Planning	9/25/08	ML082690889	12/23/2008	ML090020175
13.03-65	25	50	Emergency Planning	9/25/08	ML082690889	12/17/2008	ML083540416
13.03-66	25	50	Emergency Planning	9/25/08	ML082690889	12/23/2008	ML083660272
13.03-67	25	50	Emergency Planning	9/25/08	ML082690889	12/17/2008	ML083540416
13.03-68	25	50	Emergency Planning	9/25/08	ML082690889	12/17/2008	ML083540416
13.03-69	25	50	Emergency Planning	9/25/08	ML082690889	12/17/2008	ML083540416
13.03-7	25	50	Emergency Planning	9/25/08	ML082690889	11/11/2008	ML083180158
13.03-70	25	50	Emergency Planning	9/25/08	ML082690889	12/17/2008	ML083540416

13.03-71	25	50	Emergency Planning	9/25/08	ML082690889	12/17/2008	ML083540416
13.03-72	25	50	Emergency Planning	9/25/08	ML082690889	12/17/2008	ML083540416
13.03-73	25	50	Emergency Planning	9/25/08	ML082690889	12/23/2008	ML083660272
13.03-74	25	50	Emergency Planning	9/25/08	ML082690889	12/23/2008	ML083660272
13.03-75	62	1826	Emergency Planning	1/14/09	ML090140072	6/12/2009	ML091670459
13.03-76	83	3255	Emergency Planning	11/2/09	ML093060265	12/11/2009	ML093490764
13.03-77	83	3255	Emergency Planning	11/2/09	ML093060265	12/11/2009	ML093490764
13.03-78	83	3255	Emergency Planning	11/2/09	ML093060265	12/11/2009	ML093490764
13.03-79	83	3255	Emergency Planning	11/2/09	ML093060265	12/11/2009	ML093490764
13.03-8	25	50	Emergency Planning	9/25/08	ML082690889	11/11/2008	ML083180158
13.03-80	83	3255	Emergency Planning	11/2/09	ML093060265	12/11/2009	ML093490764
13.03-81	83	3255	Emergency Planning	11/2/09	ML093060265	12/11/2009	ML093490764
13.03-82	83	3255	Emergency Planning	11/2/09	ML093060265	12/11/2009	ML093490764
13.03-83	83	3255	Emergency Planning	11/2/09	ML093060265	12/11/2009	ML093490764
13.03-84	83	3255	Emergency Planning	11/2/09	ML093060265	12/11/2009	ML093490764
13.03-85	83	3255	Emergency Planning	11/2/09	ML093060265	12/11/2009	ML093490764
13.03-86	83	3255	Emergency Planning	11/2/09	ML093060265	12/11/2009	ML093490764
13.03-87	83	3255	Emergency Planning	11/2/09	ML093060265	12/11/2009	ML093490764
13.03-88	94	4751	Emergency Planning	3/8/11	ML110670530	4/25/2011	ML11116A159
13.03-89	94	4751	Emergency Planning	3/8/11	ML110670530	4/25/2011	ML11116A159
13.03-9	25	50	Emergency Planning	9/25/08	ML082690889	3/4/2010	ML100690444
13.03-90	111	7398	Emergency Planning	6/16/14	ML14030A187	6/26/14	ML14182A440
13.03-91	111	7398	Emergency Planning	6/16/14	ML14030A187	6/26/14	ML14182A440
13.03-92	111	7398	Emergency Planning	6/16/14	ML14030A187	6/26/14	ML14182A440
13.03-93	111	7398	Emergency Planning	6/16/14	ML14030A187	6/26/14	ML14182A440
13.03-94	111	7398	Emergency Planning	6/16/14	ML14030A187	6/26/14	ML14182A440
13.03-95	111	7398	Emergency Planning	6/16/14	ML14030A187	6/26/14	ML14182A440
13.03-96	123	7686	Emergency Planning	10/3/14	ML14279A219	11/6/2014	ML14314A035
13.03-96	123	7686	Emergency Planning	10/3/14	ML14279A219	11/6/2014	ML14314A035
13.03-96	123	7686	Emergency Planning	10/3/14	ML14279A219	11/6/2014	ML14314A035
13.03-96	123	7686	Emergency Planning	10/3/14	ML14279A219	11/6/2014	ML14314A035
13.03-97	123	7686	Emergency Planning	10/3/14	ML14279A219	11/6/2014	ML14314A035
13.03-97	123	7686	Emergency Planning	10/3/14	ML14279A219	11/6/2014	ML14314A035
13.03-98	123	7686	Emergency Planning	10/3/14	ML14279A219	11/6/2014	ML14314A035
13.03-99	123	7686	Emergency Planning	10/3/14	ML14279A219	11/6/2014	ML14314A035
13.06-1	78	3528	Physical Security	10/5/09	ML093220976	11/12/2009	ML093220239
13.06-10	78	3536	Physical Security	10/5/09	ML092940581	11/12/2009	ML093220239
13.06-11	78	3536	Physical Security	10/5/09	ML092940581	11/12/2009	ML093220239
13.06-12	78	3535	Physical Security	10/5/09	ML092790323	11/12/2009	ML093220239
13.06-13	78	3535	Physical Security	10/5/09	ML092790323	11/12/2009	ML093220239
13.06-14	78	3535	Physical Security	10/5/09	ML092790323	11/12/2009	ML093220239
13.06-15	78	3534	Physical Security	10/5/09	ML092790323	11/12/2009	ML093220239
13.06-16	78	3534	Physical Security	10/5/09	ML092790323	11/12/2009	ML093220239
13.06-17	78	3531	Physical Security	10/5/09	ML092790323	11/12/2009	ML093220239
13.06-18	78	3531	Physical Security	10/5/09	ML092790323	11/12/2009	ML093220239
13.06-19	78	3531	Physical Security	10/5/09	ML092790323	11/12/2009	ML093220239
13.06-2	78	3528	Physical Security	10/5/09	ML093220976	11/12/2009	ML093220239
13.06-20	78	3530	Physical Security	10/5/09	ML092790323	11/12/2009	ML093220239
13.06-21	78	3530	Physical Security	10/5/09	ML092790323	11/12/2009	ML093220239
13.06-22	78	3530	Physical Security	10/5/09	ML092790323	11/12/2009	ML093220239

13.06-23	78	3529	Physical Security	10/5/09	ML092790323	11/12/2009	ML093220239
13.06-24	79	3543	Physical Security	10/6/09	ML092940581	11/12/2009	ML093220239
13.06-25	79	3543	Physical Security	10/6/09	ML092940581	11/12/2009	ML093220239
13.06-26	79	3545	Physical Security	10/6/09	ML092940581	11/12/2009	ML093220239
13.06-27	79	3545	Physical Security	10/6/09	ML092940581	11/12/2009	ML093220239
13.06-28	79	3545	Physical Security	10/6/09	ML092940581	11/12/2009	ML093220239
13.06-29	79	3545	Physical Security	10/6/09	ML092940581	11/12/2009	ML093220239
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13.07-2	99	6140	Fitness for Duty (Future SRP Section)	10/26/11	ML11299A120	11/22/2011	ML11327A146
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14.03.12-2	97	4225	Physical Security Hardware - Inspections, Tests, Analyses, and Acceptance Criteria	6/9/11	ML11160A103	7/6/11	ML11192A054
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19-9	71	2744	Probabilistic Risk Assessment and Severe Accident Evaluation	7/15/09	ML091960539	8/17/2009	ML092310486

APPENDIX D

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APPENDIX E. PRINCIPAL CONTRIBUTORS

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Andrukat, Dennis	Fire Protection
Arora, Surinder	Project Management
Ashley, Clinton	Containment Systems
Barss, Daniel	Emergency Preparedness
Berrios, Ilka	Plant Systems
Bieganousky, Wayne	Geotechnical Engineering
Bowers, Anthony	Emergency Preparedness
Branch, Richard	Plant Systems
Brandt, Philip	Reactor Projects
Brown, David	Health Physics
Campe, Kazimieras	Site Hazards
Caruso, Mark	Probabilistic Risk Assessment (PRA)/Severe Accidents
Caverly, Jill	Project Management
Chalk, Wayne	Fitness for Duty
Chapman, Greg	Nuclear Fuel Radiation Protection and Criticality Control
Chazell, Russell	Technical Assistant
Chen, Pei-Ying	Engineering Mechanics
Chien, Nan	Ventilation Systems
Chopra, Om	Electrical Engineering
Chuang, Tze-Jer	Structural Engineering
Cicotte, George	Radioactive Waste Management/Process & Effluent Monitoring Systems
Clark, Phyllis	Project Management
Coflin, Monika	Cyber Security
Comar, Manny	Project Management
Curran, Gordon	Plant Systems
Devlin-Gill, Stephanie	Seismology
Dickson, Elijah	Health Physics
Dinh, Thinh	Fire Protection
Downs, James	Nuclear Fuel Fire Protection
Dudley, Noel	Project Management
Dusaniwskyj, Michael	Project Management

Name	Responsibility
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Echols, Frances	Special Nuclear Material Safety Analysis
Eudy, Michael	Project Management
Fitzpatrick, Robert	Electrical Engineering, Electrical Engineering and Equipment Qualification
Forrest, Edwin	Ventilation Systems
Forsaty, Fred	Reactor Systems
Frost, John	Physical Security, Physical Security ITAAC
Galletta, Thomas	Project Management
Garcia-Santos, Norma	Project Management
Giacinto, Joseph	Hydrology
Giebel, Stephen	Health Physics
Gilles, Nanette	Project Management
Gleaves, William	Project Management
Goel, Raj	Containment Systems
Goetz, Sujata	Project Management
Gordon, Dennis	Physical Security
Grady, Anne-Marie	Containment and Severe Accidents
Graizer, Vladimir	Seismology
Gran, Zachary	Health Physics
Habib, Donald	Project Management
Haggerty, Sharon	License Assistant
Haider, Syed	Control Room Habitability, Ventilation Systems, Containment Thermal Hydraulics
Harbuck, Charles	Technical Specifications
Harris, Larry	Material Control and Accounting Fuel Cycles and Transportation
Hart, Michelle	Accident Analysis
Harvey, Brad	Meteorology
Hearn, Peter	Project Management
Hernandez, Jorge	Plant Systems
Hernandez, Raul	Plant Systems
Hinson, Charles	Health Physics
Hoellman, Jordan	Project Management
Honcharik, John	Component Integrity Performance Testing; Materials
Hood, Tanya	Project Management
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Huang, Jason	Engineering Mechanics
Hughes, Brian	Project Management
Jain, Bhagwat	Structural Engineering
Jenkins, Joel	Materials Engineering
Jones, Henry	Hydrology
Joshi, Ravindra	Project Management
Kang, Peter	Electrical Engineering and Equipment Qualification
Keefe, Molly	Human Factors Engineering
Kellum, James	Operator Training
Kelly, Glenn	Probabilistic Risk Assessment (PRA)/Severe Accidents
Kleeh, Edmund	ITAAC Quality/Inspectability
Lavie, Stephen	Emergency Preparedness
Le, Hien	Technical Specifications
Le, Tuan	Engineering Mechanics
Lee, Eric	Electrical Instruments
Li, Chang-Yang	Plant Systems
Li, Yueh-Li (Renee)	Engineering Mechanics
Lintz, Mark	Operator Training
Lopas, Sarah	Project Management
Ma, John	Structural Engineering
Makar, Gregory	Materials Engineering
Marshall, Amanda	Emergency Preparedness
Martinez-Navedo, Tania	Electrical Engineering
Masters, Anthony	Project Management
Mazaika, Michael	Meteorology
Mcbride, Mark	Ground Water Hydrology
Mcgovern, Denise	Project Management
Mcnally, Richard	Engineering Mechanics
Miernicki, Michael	Project Management
Minarik, Anthony	Project Management, Reactor Systems
Misenhimer, David	Project Management
Moody, Robert	Emergency Preparedness
Morton, Wendell	Emergency Communications
Moser, Michelle	Project Management
Ng, Ching	Reliability and Risk

Name	Responsibility
Nolan, Ryan	Plant Systems
Olvera, Eric	Insurance/Indemnity
Orders, William	Project Management
Patel, Chandu	Project Management
Patel, Pravin	Structural Engineering
Patel, Raju	Initial Test Program
Patterson, Malcolm	Probabilistic Risk Assessment (PRA)/Severe Accidents
Pelton, Richard	Operator Training
Pieringer, Paul	Human Factors
Plaza-Toledo, Meralis	Geology
Poehler, Jeffrey	Materials Engineering
Pohida, Marie	Probabilistic Risk Assessment (PRA)/Severe Accidents
Powell, Eric	Probabilistic Risk Assessment (PRA)/Severe Accidents
Purciarello, Gerard	Plant Systems
Quinlan, Kevin	Meteorology
Radlinski, Robert	Plant Systems
Ray, Neil	Materials Engineering
Ray, Sheila	Electrical Engineering, Electrical Engineering and Equipment Qualification
Redden, Adrienne	Health Physics
Reddy, Devender	Plant Systems
Roach, Edward	Health Physics
Robinson, Edward	Emergency Planning
Roche-Rivera, Robert	Structural Engineering
Rodriguez, Michael	Plant Security
Rodriguez, Rafael	Instrumentation
Roggenbrodt, William	Project Management
Rycyna, John	Cyber Security
Sadollah, Mohammad	Electrical Engineering
Santos, Cayetano	Project Management
Sastre-Fuentes, Eduardo	Chemical Engineering
Scarbrough, Thomas	Component Integrity Performance Testing
Schaffer, Steven	Radioactive Waste Management/Process & Effluent Monitoring Systems
Schleicher, Lisa	Project Management
Schroer, Suzanne	Probabilistic Risk Assessment (PRA)/Severe Accidents

Name	Responsibility
Scully, Derek	Technical Specifications
Shepherd, James	Regulations
Shum, David	Plant Systems
Simmons, Anneliese	Financial Analyst
Sisk, David	Site Hazards
Smith, Tanya	Project Management
Smith, Will	Plant Security
Steckel, James	Project Management
Steingass, Tim	Materials Engineering
Stirewalt, Gerry	Geology
Strnisha, James	Component Integrity
Stubbs, Angelo	Plant Systems
Sweat, Tarico	Plant Systems
Takacs, Michael	Project Management
Talbot, Frank	Initial Test Program
Tammara, Rao	Site Hazards
Tardiff, Albert	Physical protection of special nuclear material of low strategic significance
Tegeler, Bret	Structural Engineering
Tello, Linda	Project Management
Tetter, Keith	Probabilistic Risk Assessment (PRA)/Severe Accidents
Thomas, Kenneth	Emergency Preparedness
Thomas, Vaughn	Structural Engineering
Tiruneh, Nebiyu	Hydrology
Tjader, Theodore	Technical Specifications
Truong, Tung	Instrumentation & Controls
Tseng, Ian	Mechanical Engineering
Tsirigotis, Alexander	Engineering Mechanics
Vaaler, Marylana	Project Management
Valentin-Olmeda, Milton	Structural Engineering
Vechioli, Lucieann	Project Management
Vera, Marieliz	Structural Engineering
Vettori, Robert	Fire Protection
Vokoun, Patricia	Project Management
Voveris, Jessica	Meteorology
Wang, Weijun	Geotechnical Engineering

Name	Responsibility
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Wentzel, Michael	Project Manager
Wheeler, Larry	Plant Systems
White, Jason	Meteorology
Wilson, Jerry	Project Management
Wilson, Josh	Plant Systems
Wong, Yuken	Engineering Mechanics
Wray, Barry	Material Control and Accounting Fuel Cycles and Transportation
Wright, Walter	Emergency Preparedness
Wu, Cheng-Ih	Engineering Mechanics
Zhang, Jing	Emergency Communications

APPENDIX F

REPORT BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

December 14, 2015

The Honorable Stephen G. Burns
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE DUKE ENERGY CAROLINAS, LLC, COMBINED LICENSE APPLICATION FOR WILLIAM STATES LEE III NUCLEAR STATION, UNITS 1 AND 2

Dear Chairman Burns,

During the 630th meeting of the Advisory Committee on Reactor Safeguards (ACRS), December 3-4, 2015, we reviewed the Duke Energy Carolinas, LLC (Duke Energy or applicant) Combined License Application (COLA) for the William States Lee III Nuclear Station (Lee), Units 1 and 2 and the NRC staff's Advanced Final Safety Evaluation Report (AFSER). The COLA incorporates the Westinghouse Electric Company AP1000 certified design, the standard content material from the AP1000 Reference COLA, and the Lee plant-specific information. Our AP1000 Subcommittee held a two-day meeting on October 21-22, 2015, to review the plant-specific information in the COLA and the staff's AFSER.

During the meeting, we had the benefit of discussions with representatives of the staff, Duke Energy and its vendors, and we had input from members of the public. We also had the benefit of the documents referenced. This report fulfills the requirement of 10 CFR 52.87 that the ACRS report on those portions of the application which concern safety.

CONCLUSION AND RECOMMENDATIONS

1. There is reasonable assurance that Lee, Units 1 and 2, can be built and operated without undue risk to the health and safety of the public.

2. Site seismic inputs requiring a departure from the AP1000 certified design have been adequately addressed by the applicant and the staff, and this departure should be approved.

3. The departure providing for a consolidated Technical Support Center for the two units should be approved.

4. The location exception for a consolidated Emergency Operations Facility should be approved.

5. The Duke Energy COLA for Lee should be approved following approval of generic changes which are pending submittal and which affect standard content material for the AP1000.

BACKGROUND

In accordance with 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Duke Energy provided to the NRC, by letter dated December 12, 2007, the initial application for Lee Units 1 and 2. In the application, Duke Energy stated that Lee will consist of two Westinghouse AP1000 certified design pressurized water reactor units, each with a core thermal power rating of 3,400 megawatts. The units are to be located in Cherokee County, South Carolina.

The application conforms to the design-centered review approach (DCRA), which is a Commission policy that allows the staff to perform one review and reach a decision for all plants which reference a certified design. The first COLA that receives a complete staff review for a certified design is designated as the Reference COLA (RCOLA). Any subsequent application referencing the same design is designated as a Subsequent COLA (SCOLA). We reviewed the COLA for Southern Nuclear Operating Company's Vogtle Electric Generating Plant, Units 3 and 4, and issued a letter report on January 24, 2011. This plant became the AP1000 RCOLA. We also reviewed COLAs for South Carolina Electric and Gas Company's V.C. Summer Nuclear Station, Units 2 and 3, and Progress Energy Florida's Levy Nuclear Plant, Units 1 and 2, and issued letter reports on February 17 and December 7, 2011, respectively. These latter plants are AP1000 SCOLAs.

DISCUSSION

Our review of the Lee SCOLA and AFSER was limited to plant-specific information, consistent with the reference to the current AP1000 certified design and RCOLA. Duke Energy identified five generic changes to standard content material:

- Condensate return and passive residual heat removal cooling
- Main Control Room operator dose
- Main Control Room heat load
- Hydrogen vent in containment
- Plant monitoring system flux doubling to comply with IEEE 603

These will be resolved as a part of the DCRA process and are applicable to Lee. They should be submitted and approved prior to approval of the Lee application. The results of our review of Lee plant-specific information are described below.

The Lee units will be located on the partially developed site of the Cherokee Nuclear Station for which the NRC issued a Construction Permit in the late 1970's. The site is adjacent to the Broad River in north-central South Carolina. Each Lee unit will be served by forced draft cooling towers. Makeup to the cooling towers is from the river or from onsite holding ponds during periods of low river flow. Processed discharges are to the nearby Ninety-Nine Islands Reservoir.

The site seismic input conforms to the Central and Eastern United States Seismic Source Characterization (NUREG-2115). The units are located on a uniform, hard-rock site which is consistent with that described in the AP1000 Design Certification Document (DCD). In order to accommodate differing site seismic conditions, the certified design includes two seismic spectra: the Certified Seismic Design Response Spectra and the Hard Rock High Frequency Response Spectra. However, the Lee site Ground Motion Response Spectrum exceeds both of these at higher frequencies. The exceedances require a DCD departure.

Accordingly, in compliance with requirements included in the AP1000 DCD, the applicant and Westinghouse performed evaluations for Seismic Category I structures, systems and components (SSCs) to demonstrate that the high frequency exceedances of the DCD spectra are non-damaging. At six specific locations which have been identified in the DCD for this purpose, in-structure spectra from these evaluations were compared to in-structure spectra from the DCD. Where the Lee spectra exceeded the DCD spectra, more detailed analysis was performed to demonstrate that the exceedances will not result in damage to the SSCs. Information from additional analyses was used to identify equipment for which high frequency amplification is important. In those cases, completed equipment qualification test response spectra were reviewed and found to envelope the Lee plant-specific response spectra. The applicant stated they will ensure that all future test response spectra for installed equipment are also higher than the plant-specific response spectra.

The staff conducted a review of these evaluations and concluded that the certified AP1000 design of Seismic Category I SSCs is acceptable at the Lee hard rock site. We agree with this conclusion.

For some Seismic Category II structures, the site foundation response spectra exceed the DCD spectra at higher frequencies. The applicant committed to design the Seismic Category II structures to the higher of the DCD or the Lee plant-specific spectra.

The applicant evaluated the potential for site water holding ponds to be the cause of reservoir-induced seismicity. The staff agreed that these ponds, including Pond C, pose negligible induced seismic risk to the Lee site.

The foundation for Lee Unit 1 will include a layer of concrete which was placed originally for the Cherokee Nuclear Station before its construction was terminated. The foundation for Lee Unit 2 will be on the natural hard rock of the site. Any resulting differences in the nuclear island foundation input response spectra have been addressed by the applicant and found acceptable by the staff.

Thorough mapping of the underlying rock formation was conducted prior to placement of concrete for the planned Cherokee Nuclear Station and similar work will be completed in accordance with a license condition to include the entire site. These explorations, which will be examined independently by the staff, and the associated laboratory testing comply with applicable regulatory guidance and satisfy the requirements of 10 CFR 100.23(c).

The waste water and liquid radioactive waste systems will discharge through a diffuser into the Ninety-Nine Islands Reservoir in accordance with a National Pollutant Discharge Elimination System operating permit. The discharge uses a stainless steel pipe located inside a guard pipe in order to ensure against leakage into the groundwater. Any leakage into the guard pipe is monitored as part of the Radiation Protection Program.

An emergency response facilities departure from the AP1000 DCD provides for a common Technical Support Center (TSC) for the two units. This is estimated to increase the travel time between the TSC and the Control Room from two minutes for the single-unit TSC location to about five minutes for the common location. This increase is considered acceptable, based on the communication and data links which are provided, and based on the fact that it allows each unit's Operational Support Center to be located adjacent to the Control Room where the TSC would have been located.

The applicant is also seeking a location exception for the Emergency Operations Facility (EOF). It would be located in Charlotte, North Carolina, approximately 40 miles from the site, in an existing EOF facility which is currently supporting the Catawba, McGuire, and Oconee Nuclear Stations. Lee would be the fourth site supported from this facility. The staff has proposed a license condition requiring that the ability of the facility to simultaneously support an emergency condition at Lee and one of the other three sites be demonstrated prior to fuel loading. The distance from the Lee site to the common EOF is not excessive, and Duke Energy will provide a closer offsite assembly area in a facility approximately 15 miles from the Lee site. We recommend approval of this location exception based on the advantages provided by use of a common EOF facility with the resources necessary to support more than a single site.

The applicant's analysis shows that conservatively determined water elevations due to local intense precipitation and the probable maximum flooding levels do not exceed the elevation of safety-related SSCs. The staff performed an independent analysis of flooding due to potential dam failures, including failure of a proposed dam not addressed in the application. This analysis showed that a conservatively estimated flood elevation due to dam failure would remain 7.5 feet below the site grade.

Other site-specific considerations, including meteorology and industrial hazards, have been addressed by the applicant, evaluated by the staff and determined to be bounded by the AP1000 DCD and RCOLA design requirements. The consequences of accidental releases of radioactive liquid effluents are within regulatory requirements, and there are no nearby intakes for potable water supplies.

SUMMARY

The applicant and the staff have addressed the plant-specific requirements necessary for approval of the SCOLA. This includes DCD departures concerning the site-specific seismic input to the plant and the location of the TSC. Subject to our recommendation that the generic issues identified by the applicant should first be approved in accordance with the DCRA, the SCOLA for Lee Units 1 and 2, including the departures and the EOF location exception, should be approved.

Sincerely,

/RA/

John W. Stetkar
Chairman

REFERENCES

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

April 18, 2016

The Honorable Stephen G. Burns
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: EXEMPTIONS TO THE AP1000 CERTIFIED DESIGN INCLUDED IN THE LEVY NUCLEAR PLANT UNITS 1 AND 2 COMBINED LICENSE APPLICATION

Dear Chairman Burns,

During the 633rd meeting of the Advisory Committee on Reactor Safeguards (ACRS), April 7-9, 2016, we reviewed five exemption requests for the Westinghouse Electric Company (WEC) AP1000 certified design which Duke Energy Florida, LLC (Duke Energy) has included in the combined license application (COLA) for the Levy Nuclear Plant (Levy) Units 1 and 2. We also reviewed the NRC staff's related Advanced Safety Evaluation Report (ASER), Chapter 21. The exemptions include changes that are grouped into six departures from the AP1000 Design Control Document (DCD), Revision 19. Our AP1000 Subcommittee held a meeting on April 5, 2016, to review the departures and the staff's ASER. The Subcommittee also met with Duke Energy, WEC, and the staff on April 9 and September 17, 2014, to review the development of the changes that are needed to achieve the intended design functions for passive residual heat removal (PRHR). These changes are included in the exemption concerning condensate return and PRHR.

During the meeting, we had the benefit of discussions with representatives of the staff, Duke Energy, and WEC, and we had input from members of the public. We also had the benefit of the referenced documents. This report fulfills the requirement of 10 CFR 52.87 that the ACRS report on those portions of the application which concern safety.

CONCLUSIONS AND RECOMMENDATION

1. Five exemptions to the AP1000 certified design have been included in the Levy combined license application. The five exemptions are needed to enable the certified design to perform intended functions and should be approved.

2. The causes for the exemptions have been identified and addressed for the AP1000 certification.

3. Generic lessons learned, relative to the reactor design process leading to certification, should be identified and further evaluated.

BACKGROUND

By letter dated July 28, 2008, Progress Energy Florida, Inc., now Duke Energy, submitted a COLA for Levy Units 1 and 2 to the NRC. On December 7, 2011, we issued a letter report to the Commission recommending approval following implementation of the stated recommendations. Subsequently, changes needed to achieve the intended design functions for PRHR were identified. Development of these changes was undertaken by WEC, with oversight from Duke Energy, and these changes are now required to be included in the COLA, pursuant to Interim Staff Guidance DC/COL-ISG-011. These departures are common to all COLAs referencing the AP1000 design, and similar changes will be necessary for AP1000 combined license holders.

Ongoing detailed design of the AP1000 units, and investigation into the extent of the condition that created the need for the PRHR-related changes, identified other needed changes requiring approval of exemptions in four additional areas. Duke Energy noted the areas requiring departures from the certified AP1000 design during our review of its William States Lee III Nuclear Station (Lee) Units 1 and 2 COLA in 2015. These were listed as follows in our letter, dated December 14, 2015, concerning the Lee COLA:

- Condensate return and PRHR
- Main control room operator dose
- Main control room heat load
- Plant monitoring system flux doubling to comply with IEEE 603
- Hydrogen vent in containment

DISCUSSION

The five exemptions and associated departures from the AP1000 certified design are needed to implement intended functions of the certified design. Each is distinct and separate from the others. The changes will be made for the common purpose of correcting errors and omissions in the certified design, which have been identified during licensing and detailed design development subsequent to certification. Therefore, we also reviewed elements that are common to the departures; in particular, the implementation of the quality assurance program requirements in 10 CFR Part 50, Appendix B during design. Finally, we also reviewed the staff's assessment of the effect of the departures on the previously completed probabilistic risk assessment.

Condensate Return and Passive Residual Heat Removal

The AP1000 design provides for closed-loop cooldown and passive heat removal under accident conditions not involving loss of coolant. Reactor coolant circulates naturally through a PRHR heat exchanger located within the in-containment refueling water storage tank (IRWST). The PRHR heat exchanger converts IRWST water to steam, and the subsequent condensation of this steam on the containment vessel interior surface passively transfers residual heat by conduction through the containment wall to the outside air. This closed-loop cooling requires that sufficient condensed water be returned to the IRWST to ensure the inventory needed to maintain the cooldown status and to continue the PRHR process for as long as necessary.

Features in the containment that are required to direct condensate back to the IRWST are described in AP1000 DCD, Revision 19. The rate of condensation varies with time, and the return of condensate to the IRWST is subject to some loss. A constant loss rate of 10 percent was assumed in the DCD analysis. Based on this assumption, DCD, Revision 19 states that (a) acceptance criteria associated with the Chapter 15 design basis safety analyses remain satisfied indefinitely, and (b) cooldown to 420°F can be achieved in 36 hours and maintained indefinitely, based on Chapter 19 assumptions and acceptance criteria.

Duke Energy has proposed for its Levy COLA an exemption seeking approval of two departures that concern cases (a) and (b) above. These departures involve physical changes in containment to increase condensate return. Downspouts, collection points, and connecting piping have been added to the polar crane girder and the internal stiffener, and many attachment plates on the containment inner surface have been eliminated. Additional testing was performed to estimate better the condensate collection on surfaces and losses at discontinuities such as attachment plates and to provide an improved basis for the estimation of condensate losses.

Based on testing and the additional features provided to return sufficient condensate back to the IRWST, a loss rate of 18 percent of the water that condenses on the containment vessel inner surface has now been assumed for cases (a) and (b) above. Water that condenses on other surfaces within containment is assumed to be entirely lost to the IRWST.

Analyses by WEC and the staff of PRHR performance were extensive. WEC used WGOTHIC and LOFTRAN with some confirmatory analyses using RELAP. Adiabatic and heat-loss models of the reactor coolant system, and the potential loss of subcooling in the reactor coolant system on heat transfer in the PRHR heat exchanger, were examined. The staff's confirmatory calculations used MELCOR and RELAP, and their results agreed well with the WEC calculations. The analyses included both the most limiting Chapter 15 non-loss-of-coolant-accident transient that credits the PRHR heat exchanger, which is the loss of normal feedwater coincident with the loss of AC power to the plant auxiliaries, and the safe shutdown analysis in Chapter 19. Based on these analyses, the duration for case (a) was extended to 72 hours, and the duration for case (b) was revised from an indefinite period to at least 14 days. Also, criteria for activation of the backup automatic depressurization system in order to establish open loop PRHR were updated.

Main Control Room Operator Dose

WEC identified several discrepancies in the certified design analyses supporting the determination of main control room (MCR) operator dose following a design basis accident (DBA). Specifically, (1) the analyses did not account for the direct dose from the MCR emergency ventilation system filter, (2) the normal ventilation system radiation monitor setpoints were not based upon all DBA release scenarios, and (3) the methodology used to estimate MCR dose contribution from direct radiation and skyshine was not up-to-date.

This exemption includes changes which add shielding for the ventilation filter, reduce the allowable secondary coolant iodine activity, update the radiation dose analyses, and revise the normal ventilation system radiation monitor logic and setpoints. The result of the changes provides a revised MCR dose for the DBA, which slightly increases the margin to the 5 rem limit.

Main Control Room Heat Load

Duke Energy identified that heat sources in the MCR had increased with detailed design development and now exceed those assumed in the certified design. Also, the design had not considered an event in which the MCR could be isolated and dependent on the emergency ventilation system, while offsite power remained available and powering certain MCR equipment. This event results in significantly higher heat loads than are considered in the certified design.

The exemption includes changes that add automatic, two-stage de-energization of select non-safety MCR heat loads. This load shed retains power for plant controls and parameter indications at the operators' normal work stations. Also, changes were made to establish limits, with surveillance requirements, for the initial MCR conditions and to ensure operation of the electrical load shedding functions.

With these changes, analysis projects that operators may remain in the MCR indefinitely, consistent with NUREG-0700 limits, following its isolation and resulting dependence on the emergency ventilation system.

Plant Monitoring System Compliance with IEEE 603

The source range neutron flux logic is a control system feature of the plant monitoring system that isolates dilute water sources to the reactor coolant system, in order to protect against inadvertent criticality due to boron dilution during shutdown conditions. Under some plant conditions, it is necessary to manually block or bypass the operation of this feature.

Operating bypasses are addressed in IEEE Standard 603-1991, and this standard is applicable to COLAs referencing the AP1000 certified design. WEC identified that, due to an omission, the certified design did not meet the requirements of the standard because this protection function could be blocked and would not be reset automatically when plant conditions require it. The exemption includes a change that will revise the plant monitoring system logic to comply with the standard and with regulatory requirements.

Hydrogen Vent Inspection, Tests, Analyses, and Acceptance Criteria (ITAAC)

WEC identified that changes in structural details internal to the containment have occurred which are inconsistent with the certified design ITAAC for one of the compartments, relative to the venting of any hydrogen accumulation in the compartment following a severe accident. The departure change to the ITAAC recognizes the possibility of a standing hydrogen flame that is closer to the containment boundary than allowed by the current ITAAC. Although the possible standing flame is closer to the containment boundary, results from analyses indicate that the higher temperatures would not compromise the structural integrity of the containment wall or of the equipment hatch cover and seals, and therefore, is acceptable.

NRC Staff Review

On March 7, 2016, the ASER for the five exemptions included in the Levy COLA was transmitted to the ACRS for review. It documents the staff's very thorough and technically complete review of the changes as they were developed over the past three years. The staff has identified that each of the exemptions is necessary in order to perform the intended functions, and therefore, meet the underlying purposes of the AP1000 certification rule.

The concluding statement in ASER Section 21.0 is "The staff finds that the cumulative risk impact of these design changes and departures is negligible." The changes are necessary to perform the intended functions that were the basis for the DCD risk calculation. However, the risk has not been calculated for the condition without the changes. While it is clear that there has been no increase in risk, it should not be concluded that the actual reduction in risk achieved by these changes is negligible.

Design Certification Quality Assurance Program

Detailed development of a certified design, involving the increasing engagement of combined license holders and applicants, should be expected to identify needed design and analysis changes. However, there are lessons to be learned from the Levy COLA experience.

Following initial discussions with our Subcommittee in 2014, WEC, Duke Energy, and the staff performed thorough evaluations, including the quality assurance program implementation. The results were reflected in the April 2016 Committee presentations. We conclude that the causes of the errors and omissions that made these exemptions necessary were addressed and programmatic changes applicable to the AP1000 certification were made where necessary.

We recommend that staff evaluate on a generic basis whether there are any lessons learned, relative to ongoing and future oversight of the quality assurance program implementation during development of designs seeking certification under 10 CFR Part 52. Prospective combined license applicants may not be in a position to provide such oversight during this phase, and they may find it difficult to do so following certification when customer oversight can be more effective. We would appreciate the opportunity to meet with the staff on this generic matter at an appropriate time.

Conclusion

The five exemptions, which include six departures from the AP1000 certified design that will be included in the Levy Units 1 and 2 COLA, effectively address errors and omissions in the current certification and should be approved. As indicated in our letter on the Lee Units 1 and 2 COLA, dated December 14, 2015, other combined license applicants referencing the AP1000 certified design will also include the exemptions in accordance with the design centered review approach described in that letter. Current combined license holders will submit license amendments to incorporate these, or similar, changes.

Sincerely,

/RA/

Dennis C. Bley
Chairman

REFERENCES

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3. Duke Energy Florida, Levy Nuclear Plant, Units 1 and 2, "Departure from AP1000 DCD Revision 19 to Address Compliance with IEEE 603-1991," September 1, 2015 (ML15247A153).

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