United States Nuclear Regulatory Commission Official Hearing Exhibit
In the Matter of: DUKE ENERGY FLORIDA, LLC
(Levy Nuclear Plant, Units 1 and 2)
Commission Mandatory Hearing
Docket #: 05200029 | 05200030
Exhibit #: NRC-007A-MA-CM01 Identified: 07/28/2016
Admitted: 07/28/2016 Withdrawn:
Rejected: Stricken:
Other:

Final Safety Evaluation Report for Combined Licenses for Levy Nuclear Plant Units 1 and 2

U. S. Nuclear Regulatory Commission
Office of New Reactors
Washington, DC 20555-0001

May 2016

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 Levy Nuclear Plant (LNP) Units 1 and 2. The applicant, Duke Energy Florida, LLC, was formerly identified as Duke Energy Florida, Inc., and Progress Energy Florida, Inc.

By letter dated July 28, 2008, 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 LNP Units 1 and 2, and would be located at a greenfield site in Levy County, Florida. The applicant submitted its final update to the COL application, Revision 9, on April 6, 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.

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

---

[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 7, 2011 and April 18, 2016. These reports are included as Appendix F to this FSER.

[2] An environmental review was also performed of the COL application, and its evaluation and conclusions are documented in NUREG-1941, "Final Environmental Impact Statement for Combined Licenses for Levy Nuclear Plant Units 1 and 2," dated April 2012.

5) issuance of the license will not be inimical to the common defense and security or to the health and safety of the public.

CONTENTS

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).

APPENDICES

FIGURES

TABLES

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 Levy Nuclear Plant (LNP) Units 1 and 2 site. The applicant, Duke Energy Florida, LLC (DEF), was formerly identified as Duke Energy Florida, Inc., and Progress Energy Florida, Inc (PEF). In a letter dated April 15, 2013, PEF notified the NRC that its name was changing to Duke Energy Florida, Inc., effective April 29, 2013. The Revision 8 update of the COL application, submitted December 7, 2015, identifies the applicant as DEF. 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-1941, "Final Environmental Impact Statement for Combined Licenses (COLs) for Levy Nuclear Plant Units 1 and 2," was issued in April 2012, and can be accessed through the Agencywide Documents Access and Management System (ADAMS) at accession nos. ML12100A063, ML12100A068 and ML12100A070.[4]

By letter dated July 28, 2008, the applicant submitted its initial application to the NRC for COLs for two AP1000 advanced passive pressurized-water reactors (PWRs) (ADAMS Accession No. ML082260277) to be located at the LNP site. The application identified the two units as LNP Units 1 and 2. The LNP site is located in Levy County, Florida, in a large rural area southwest of Gainesville and west of Ocala and approximately 15.5 kilometers (9.6 miles) northeast of the Crystal River Energy Complex, an energy facility also owned by DEF.

---

[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.

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 9, on April 6, 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 reports dated December 7, 2011 and April 18, 2016. Appendix F includes a copy of these reports 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
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
RCC	roller compacted concreter
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

1.0 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 11 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); and
- 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 letter dated July 28, 2008, as supplemented by several letters, Florida Power Corporation, doing business as Progress Energy Florida, Inc. (PEF), submitted its application to the U.S. Nuclear Regulatory Commission (NRC or the Commission) for a 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 would be identified as Levy Nuclear Plant (LNP), Units 1 and 2, and would be located approximately 9.6 miles northeast of the Crystal River Energy Complex in Levy County, Florida.

The COL applicant is Duke Energy Florida, LLC (DEF). Subsequent to a corporate merger between Progress Energy, Inc., formerly the ultimate corporate parent of PEF, and Duke Energy, and subsequent to a corporate reorganization, Duke Energy Florida, Inc., submitted an updated Revision 7 of the COL application on August 28, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession Number ML14258A955).[1] In the Revision 8 update to the COL application dated December 7, 2015, Part 1, the applicant stated that it filed amended articles of conversion and organization to change its corporate name to Duke Energy Florida, LLC, effective August 1, 2015 (ADAMS Accession Number ML15349A100). Unless otherwise noted, this FSER (also referred to as the safety evaluation report (SER) in later sections of this document) is based on Revision 9 of the LNP COL application.

[1] The applicant, Duke Energy Florida, LLC, was formerly identified as Duke Energy Florida, Inc. and Progress Energy Florida, Inc. In a letter dated April 15, 2013, Progress Energy Florida, Inc., notified the NRC that its name was changing to Duke Energy Florida, Inc., effective April 29, 2013. The name change and a 2012 corporate merger between Duke Energy and Progress Energy are described in Section 1.5.1 of the SER. Because a portion of the review described in this chapter was completed prior to the name change, the NRC staff did not change references to "Progress Energy Florida" or "PEF" to "Duke Energy Florida" or "DEF" in this chapter.

As indicated in the applicant's April 6, 2016, Revision 9 submission, the applicant incorporates by reference 10 CFR Part 52, Appendix D, "Design Certification Rule for the AP1000 Design," and the Westinghouse Electric Corporation's (Westinghouse's) Design Control Document (DCD) Revision 19.

The AP1000 nuclear reactor design is a PWR with a power rating of 3400 megawatts thermal (MWt) and 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 LNP 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 LNP 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 DEF 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 LNP Plant-Specific Technical Specifications (PTS) and Bases. Specifically, Section A addresses completion of bracketed information. Section B provides a complete copy of the LNP PTS and Bases.

- **Part 5 Emergency Plan**

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

- **Part 6 Limited Work Authorization (Revision 1)**

Part 6 of the COL application, Revision 0, included a site redress plan and environmental report related to a Limited Work Authorization (LWA) request to perform certain safety-related construction activities. Subsequently, the applicant withdrew its LWA request. As such, Part 6 of the COL application is not used.

- **Part 7 Departure and Exemption Requests**

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 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 Safeguards/Security Plans**

These plans are categorized as security safeguards information and are withheld from public disclosure pursuant to 10 CFR 73.21, “Protection of Safeguards Information: Performance requirements,” and 10 CFR 73.22, “Protection of Safeguards Information: Specific requirements.”

- LNP Safeguards/Security Plan, which consists of the Physical Security Plan, the Training and Qualification Plan, and the Safeguards Contingency Plan. These security plans are submitted to the NRC as a separate licensing documents in order to fulfill the requirements of 10 CFR 52.79(a)(35) and 10 CFR 52.79(a)(36).

- Special Nuclear Material (SNM) Physical Protection Program Description

- **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:

- Portions of the COL application Part 5 – Emergency Plan

- LNP Units 1 and 2 Cyber Security Plan, as required by 10 CFR 73.54, “Protection of Digital Computer and Communication Systems and Networks.”

- Mitigative Strategies Description and Plans, as required by 10 CFR 52.80(d)

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

Part 10 includes LNP proposed license conditions and inspections, tests, analyses, and acceptance criteria (ITAAC) information in accordance with 10 CFR 52.80. A table identifying the proposed license conditions appears in Appendix A of this FSER.

The contents of the environmental protection plan (and associated license conditions) are not evaluated in this SER. Part 10 of the application incorporated by reference the AP1000 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 includes information submitted by the applicant in support of the LNP COL application. Specifically, these sections include

- New Nuclear Plant Development 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 DEF.

- Cyber Security Plan: The SUNSI version of the cyber security plan is provided in Part 9 of the application.

- Mitigative Strategies Descriptions and Plans: The SUNSI version of the Mitigative Strategies Descriptions and Plans is provided in Part 9 of the application

- Special Nuclear Material (SNM) Material Control and Accounting Program Description

- New Fuel Shipping Plan

- Supplemental Information in Support of 10 CFR Part 70 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, environmental report, and mitigative strategies plan required by 10 CFR 50.54(hh)(2).

- 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 (DC)).

- 10 CFR 52.85, “Administrative review of applications; hearings,” provides requirements for administrative reviews and hearings.

- 10 CFR 52.87, “Referral to the Advisory Committee on Reactor Safeguards (ACRS),” provides for referral to the ACRS.

The NRC 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"
- "Standard Review Plan on Foreign Ownership, Control, or Domination"

In addition, the staff considered the format and content guidance in RG 1.206[2] for the COL application.

1.2.2 Finality of Referenced NRC Approvals

In accordance with 10 CFR 52.83, if the application for a COL references a DC rule, the scope and nature of matters resolved in the DC for the application and any COL issued are governed by 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), programmatic elements that are the responsibility of the COL, and departures from the certified design).

The certified AP1000 design currently incorporated by reference in the LNP COL application is in 10 CFR Part 52, Appendix D, and is based on the AP1000 DCD as amended through Revision 19. The results of the NRC staff's technical evaluation of the AP1000 DCA application are documented in NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," and its supplements. Referencing the Revision 19 certified design in 10 CFR Part 52, Appendix D resolves Confirmatory Item LNP 1.2-1 from the advanced safety evaluation (ASE).

The contents of the AP1000 COL application are specified by 10 CFR 52.79(a), which requires the submission of information within the FSAR that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components (SSCs) of the facility as a whole. For a COL application that references a DC, 10 CFR 52.79(d) requires the DCD to be included or incorporated by reference into the FSAR. A COL application that references a certified design must also include the information and analysis required to be submitted within the scope of the COL application, but which is outside the scope of the DCD. This set of information addresses plant- and site-specific information and includes all COL action or information items; design information replacing CDI; and

[2] Appendix D, Section IV.A.2.a to 10 CFR Part 52 requires the COL application to include a plant-specific DCD that includes the same type of information and uses the same organization and numbering as the generic DCD. The generic DCD used RG 1.70, Revision 3, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," 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 LNP COL application and the guidance of RG 1.206.

programmatic information that was not reviewed 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 as supplemented by the COL application, or completely in the COL application. Following this confirmation, the staff's review of the COL application is limited to the COL-specific 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," dated 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 cyber security plans for the AP1000 COL applicants are essentially identical with the exception of title names being different. Other areas where this approach was used include 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 NRC 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 LNP Units 1 and 2 COL application has been designated as an SCOL application in the AP1000 design center[3].

DEF, 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 LNP 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:

[3] In a letter dated April 28, 2009, 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 safety evaluation (SE) with open items. As part of the transition, the NRC staff concluded that the BLN evaluation material identified as Standard (STD COL, STD SUP, STD DEP and Interfaces for Standard Design) in the BLN SE was directly applicable to the VEGP review. As a result, standard content material from the SE for the RCOL (VEGP) application and referenced in the LNP SE includes evaluation material from the SE for the BLN COL application.

- STD – standard (STD) information that is identical in each COL referencing the AP1000.
- LNP – plant-specific information that is specific to this application.
- DEP – represents a departure (DEP) from the DCD.
- COL – represents a COL information item identified in the DCD.
- SUP – represents information that supplements (SUP) information in the DCD.
- CDI – represents design information replacing CDI included in the DCD but not addressed within the scope of the DCD review.

The following text is added to the technical evaluation sections in this SER 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:
>
> - The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 LNP 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 LNP application. These evaluations are in the SER sections that reference the standard content.

The staff compared the VEGP COL FSAR, Revision 2 to the LNP COL FSAR at the time of the development of the ASE. The ASE included confirmatory items. Subsequent to the issuance of the ASE, the LNP applicant updated the standard portions of its application to be consistent with the VEGP COL application to close the standard content confirmatory items. Following this update, the staff performed a complete comparison of the standard content appearing in the VEGP COL FSAR, Revision 5 to the LNP COL FSAR, Revision 4. The staff confirmed that responses to standard content confirmatory items were endorsed by LNP applicant and that the changes discussed in the standard confirmatory items were made in the LNP COL FSAR. The staff reviewed DEF and PEF changes to standard content as discussed above.

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 DEF pursuant to 10 CFR 50.33, "Contents of applications; general information," is provided in Section 1.5.1 of this SER.

- **Part 2 Final Safety Analysis Report**

The staff's evaluation of information in the LNP COL FSAR is provided in the corresponding sections of this SER.

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 LNP 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 staff's evaluation of information in an environmental report submitted pursuant to the requirements of 10 CFR 51.50(c) is provided in the Environmental Impact Statement.

- **Part 4 Technical Specifications**

Chapter 16 of this SER includes the staff's evaluation of the LNP Units 1 and 2 PTS and Bases (specifically completion of bracketed text).

- **Part 5** **Emergency Plan**

Chapter 13 of this SER includes the staff's evaluation of the LNP Emergency Plan, 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 evaluation.

- **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. 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.[4]	1.5.4
LNP DEP 1.8-1. Departure correcting an inconsistency in regulatory citation in an interface description	1.5.4
LNP DEP 3.2-1. Departure adding downspouts and downspout screens to the condensate return portion of the Passive Core Cooling System.[4]	21.1
LNP DEP 3.7-1. Departure to address use of site-specific horizontal seismic response spectra for the design of drilled shafts that support the seismic Category II portions of the Annex and Turbine Buildings.	3.7
LNP DEP 3.11-1. Departure revising the environmental zone numbers for Spent Fuel Pool Level instruments.	3.11
LNP DEP 6.2-1. Departure revising the ITAAC Acceptance Criteria for the in-containment PXS compartment vents to reflect the current plant configuration.[4]	21.4
LNP DEP 6.3-1. Departure to quantify the term "indefinitely" as used in the DCD for maintenance of safe shutdown conditions using the PRHR HX during non-LOCA accidents.	21.1
LNP 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."[4]	21.2

[4] These departures include revisions to either AP1000 Tier 1 information or generic TS and correspond to exemptions requested by the applicant.

Description of Departure	Location of Evaluation in this Report
LNP 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.[4]	21.3
LNP 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, "Standard Criteria for Safety Systems for Nuclear Power Generating Stations," in accordance with 10 CFR 50.55a(h), "Protection and safety systems."[4]	21.5
STD DEP 8.3-1. Departure for Class 1E voltage regulating transformer current limiting features.	8.3.2

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 Special Nuclear Material (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 Technical Specification (TS) Surveillance Requirement (SR) 3.5.4.7 related to Containment Cooling Changes in regard to Passive Core Cooling System Condensate Return	21.1
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)[5]	1.5.4

[5] Part 7 of the LNP 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.

- **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 SER and is withheld from the public in accordance with 10 CFR 73.21 and 10 CFR 73.22. A non-sensitive summary of the staff's evaluation of those plans is provided in Section 13.6 of this SER.

The staff's evaluation of the Special Nuclear Material (SNM) Physical Protection Plan is documented in Section 1.5.5.1 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 or fires is provided in Appendix 19A of this SER. 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 LNP Units 1 and 2 Cyber Security Plan is provided in Section 13.8 of this SER.

- **Part 10 Proposed Combined License Conditions and 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 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 LNP Units 1 and 2 COL application. Specifically, these enclosures include:

- Nuclear Development Quality Assurance Manual (NDQAM) – The NDQAM is the top-level policy document that establishes the QA policy and assigns major functional responsibilities for nuclear development activities conducted by or for DEF.

- 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) – The SUNSI version of this enclosure is provided in Part 9 of the application.

- Cyber Security Plan – The SUNSI version of the Cyber Security Plan is provided in Part 9 of the application.

- SNM Material Control and Accounting (MC&A) Program

- New Fuel Shipping Plan

- Supplemental Information in Support of 10 CFR Part 70, SNM License Application

Organization of the SER

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 SER 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 LNP 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 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 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 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 LNP COL FSAR Chapter 1

1.4.1 Introduction

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

- 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.

- Specific information relating to qualifications of the applicant, construction impacts, and regulatory considerations that applies throughout the balance of the application (e.g., conformance with the acceptance criteria in NUREG-0800).

This section of the SER 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 Chapter 1 of the LNP COL FSAR 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 SER section, but in subsequent sections as referenced below.

<u>FSAR Section 1.1, Introduction</u>

Section 1.1 of the LNP COL FSAR, Revision 9, incorporates by reference Section 1.1, "Introduction," of the AP1000 DCD, Revision 19 with the following supplements. In a letter dated April 19, 2011, the applicant endorsed a VEGP letter dated November 11, 2010, that added a discussion of incorporation of the proprietary information and safeguards information referenced in the AP1000 DCD.

- STD SUP 1.1-1

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

- LNP SUP 1.1-2

The applicant clarified that the FSAR was submitted to NRC by DEF 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."

- LNP COL 2.1-1

The applicant provided additional information in LNP COL 2.1-1 to address COL Information Item 2.1-1 (COL Action Item 2.1.1-1). Specifically, LNP Units 1 and 2 are to be located in Levy County, Florida approximately 9.6 miles northeast of the Crystal River Energy Complex. This is a brief introductory summary of the plant location. An expanded discussion of LNP COL 2.1-1 is included in LNP COL FSAR Section 2.1.

- LNP COL 1.1-1

The applicant provided the anticipated schedule for construction and operation of LNP Units 1 and 2 in LNP COL FSAR Table 1.1-203. The applicant committed to provide a site-specific construction plan and startup schedule after issuance of the COL.

- STD SUP 1.1-6

The applicant identified that, while the LNP 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 LNP COL FSAR Section 1.8 and Part 7 of the LNP 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 LNP 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 and 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 LNP COL FSAR. Proprietary and sensitive material was provided in Part 9 of the COL application.

- LNP SUP 1.1-5

The applicant provided additional information to identify acronyms and abbreviations used in the LNP COL FSAR that are in addition to the acronyms identified in the AP1000 DCD.

<u>FSAR Section 1.2, General Plant Description</u>

Section 1.2 of the LNP COL FSAR, Revision 9, incorporates by reference Section 1.2, "General Plant Description," of the AP1000 DCD, Revision 19 with the following departures and supplements:

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

The applicant provided additional information on the site plan for LNP Units 1 and 2 summarizing the principal structures and facilities, parking areas, and roads. The location and orientation of the power block complex are also described. These COL information items are expanded in other sections of the LNP COL FSAR.[6]

FSAR Section 1.3, Comparisons with Similar Facility Designs

Section 1.3 of the LNP COL FSAR, Revision 9, incorporates by reference Section 1.3, "Comparisons with Similar Facility Designs," of the AP1000 DCD, Revision 19 with no departures or supplements.

FSAR Section 1.4, Identification of Agents and Contractors

Section 1.4 of the LNP COL FSAR, Revision 9, incorporates by reference Section 1.4, "Identification of Agents and Contractors," of the AP1000 DCD, Revision 19 with the following departures and/or supplements:

- LNP SUP 1.4-1

The applicant provided additional information to identify DEF as the COL applicant for LNP Units 1 and 2. Additionally, the applicant identified DEF as the owner and operator of LNP Units 1 and 2. DEF is a subsidiary of Progress Energy, Inc., which is a subsidiary of Duke Energy Corporation.

- LNP SUP 1.4-3

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

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

- CH2M Hill, Inc.
- Sargent & Lundy, LLC
- WorleyParsons Resources and Energy
- Westinghouse Electric Company, LLC

FSAR Section 1.5, Requirements for Further Technical Information

Section 1.5 of the LNP COL FSAR, Revision 9, incorporates by reference Section 1.5, "Requirements for Further Technical Information," of the AP1000 DCD, Revision 19 with no departures or supplements. This section of the 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 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.

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

FSAR Section 1.6, Material Referenced

Section 1.6 of the LNP COL FSAR, Revision 9, incorporates by reference Section 1.6, "Material Referenced," of the AP1000 DCD, Revision 19 with the following supplements:

- STD SUP 1.6-1

The applicant identified Table 1.6-201 as providing a list of the technical documents incorporated by reference in the LNP COL FSAR in addition to those technical documents incorporated by reference in the AP1000 DCD.

- LNP SUP 1.6-1

The applicant identified supplemental portions of Table 1.6-201 as site-specific and identified them as LNP SUP 1.6-1.

FSAR Section 1.7, Drawings and Other Detailed Information

Section 1.7 of the LNP COL FSAR, Revision 9, incorporates by reference Section 1.7, "Drawings and Other Detailed Information," of the AP1000 DCD, Revision 19, with the following supplements:

- LNP SUP 1.7-1

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

FSAR Section 1.8, Interfaces for Standard Design

Section 1.8 of the LNP COL FSAR, Revision 9, incorporates by reference Section 1.8, "Interfaces for Standard Design," of the AP1000 DCD, Revision 19 with the following departures and/or supplements:

- LNP DEP 1.8-1

The applicant provided a departure to address an error in the DCD Table 1.8-1 listing of plant interfaces where Item 13.1 incorrectly references Appendix O of 10 CFR Part 50. This departure is evaluated in Section 1.4 of this document.

- LNP SUP 1.8-1

The applicant identified departures in LNP COL FSAR Table 1.8-201, "Summary of FSAR Departures from the DCD." The departures are listed above in Table 1-1.

- LNP SUP 1.8-2

The applicant provided a list of the COL information items in the AP1000 DCD. In LNP COL FSAR Table 1.8-202, DEF provides the sections of the application addressing these issues.

The table further identifies each AP1000 COL item as an “applicant” item, a “holder” item, or both. An applicant item is completely addressed in the application. DEF’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.

- LNP SUP 1.8-3

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

<u>FSAR Section 1.9, Compliance with Regulatory Criteria</u>

Section 1.9 of the LNP COL FSAR, Revision 9, incorporates by reference Section 1.9, “Compliance with Regulatory Criteria,” of the AP1000 DCD, Revision 19 with the following supplements:

- STD COL 1.9-1 and LNP COL 1.9-1

The applicant provided additional information related to NRC RGs cited in the LNP COL FSAR. Table 1.9-201 identifies the RG revision and provides LNP COL FSAR cross-references. In addition, Appendix 1AA, “Conformance with Regulatory Guides,” was developed by the applicant to supplement the detailed discussion presented in Appendix 1A, “Conformance with Regulatory Guides,” of the referenced DCD. Specifically, Appendix 1AA delineates conformance of design aspects as stated in the DCD and conformance with programmatic and/or operational issues as presented in the LNP COL FSAR. In certain RGs, design aspects were beyond the scope of the DCD and are presented in the LNP COL FSAR.

- STD COL 1.9-2 and LNP COL 1.9-2

The applicant provided additional information related to operational experience. LNP COL FSAR Table 1.9-204 provides a list of Bulletins and Generic Letters (GLs), the appropriate LNP 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 related to review of Unresolved Safety Issues and Generic Safety Issues (GSIs). Specifically, 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 and LNP SUP 1.9-1

The applicant provided additional information related to conformance with NUREG-0800. Specifically LNP 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 DCD as presented in the LNP COL FSAR.

- STD SUP 1.9-2

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

- STD SUP 1.9-3

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."

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

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

- STD SUP 1.10-1

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 addresses the review of an evaluation of potential hazards to the SSCs important to safety of the 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 limiting conditions for operation (LCOs) are not exceeded as a result of construction activities at a multi-unit site.

- LNP SUP 1.10-1

The applicant identified that the power blocks for LNP Units 1 and 2 have a minimum separation of at least 900 feet between plant centerlines. The standard portion of the application discusses the primary consideration in setting this separation distance as 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 Appendix B of Part 10 of the LNP COL application 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 the Commission regulations for the introductory information in LNP COL FSAR Chapter 1 are given in Section 1.0 of NUREG-0800.

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

- 10 CFR 50.43(e), "Additional standards and provisions affecting class 103 licenses and certifications for commercial power," 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, which 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) requiring 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), "Issuance of combined licenses," 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 within 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 the applicant's 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.

In conformance with the regulatory acceptance criteria in RG 1.206 the applicant provided an evaluation for conformance with guidance in RGs in effect 6 months prior to the submittal of the COL application.

1.4.4 Technical Evaluation

The NRC staff reviewed Section 1 of the LNP 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.[7] The NRC 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 NRC staff's evaluation of the information incorporated by reference in the LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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.

[7] See Section 1.2.2, "Finality of Referenced NRC Approvals" 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.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the LNP 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 LNP designation (e.g., LNP Confirmatory Item 1.4-1).

The staff reviewed the information in the LNP COL FSAR:

<u>LNP COL FSAR Sections 1.1, 1.2, and 1.3</u>

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

In LNP COL FSAR Section 1.1, LNP COL 1.1-1 states that a site-specific construction plan and startup schedule will be provided after issuance of the COL. This is identified as **LNP Commitment Number 1.4-1**.

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

> *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**.*
>
> <u>*Resolution of Standard Content Confirmatory Item 1.4-1*</u>
>
> *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.*

In a letter dated June 3, 2014, the applicant notified NRC that the engineering, procurement, and construction (EPC) contract for LNP Units 1 and 2 had been terminated, and proposed language revising related information in FSAR Chapters 1 and 13. The staff verified that applicant incorporated these changes in Revision 7 of the LNP Units 1 and 2 COL application. In RAI Letter No. 123 dated October 17, 2014, the staff issued RAI 01.05-1 to confirm that, even without an EPC contract, the applicant had access to nonpublic information as stated above in the FSAR. The applicant responded in a letter dated January 22, 2015, describing another agreement in place in which Westinghouse grants DEF the right to the nonpublic information for the life of the project, as stated in the FSAR.

<u>LNP COL FSAR Section 1.4</u>

- LNP SUP 1.4-1

This evaluation is limited to DEF'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 SER.

LNP COL FSAR Section 1.4 states that DEF will own and operate LNP Units 1 and 2. Part 1 of the COL application, Section 1.1.3, states that DEF, the applicant for the LNP 1 and 2 COLs, is primarily engaged in the generation, transmission, distribution, and sale of electricity in portions of central and north Florida. DEF serves approximately 1.7 million customers in a territory encompassing over 20,000 square miles, including the cities of Saint Petersburg, Clearwater, and areas surrounding Orlando. DEF owns and operates the Crystal River plant (permanent shutdown/retired). DEF is a regulated public utility, and is subject to the regulatory provisions of the Florida Public Service Commission, the NRC and the Federal Energy Regulatory Commission. In addition, the FSAR (Sections 1.4 and 13.1.1) states that Duke Energy Corporation, the ultimate corporate parent of DEF, has over 40 years of experience in the design, construction and operation of nuclear power stations, and currently has twelve nuclear operating units.

Because DEF has demonstrated its ability to build and operate a nuclear unit, the staff finds that DEF is qualified to hold a 10 CFR Part 52 license. The staff notes that Section 17.5 of the LNP COL FSAR discusses the QA program to be implemented at the receipt of the COL. The staff's evaluation of Section 17.5 of the LNP COL FSAR is in Section 17.5 of this SER. Based on DEF's experience with building and operating a nuclear power plant and the staff's evaluation of DEF's QA program, the staff finds that DEF is technically qualified to hold a 10 CFR Part 52 license in accordance with 10 CFR 52.97(a)(1)(iv).

<u>LNP COL FSAR Section 1.5</u>

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.

<u>LNP COL FSAR Section 1.6</u>

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

LNP 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.

LNP COL FSAR Section 1.8

- LNP SUP 1.8-1

As discussed in SER Section 1.4.2, the applicant identifies departures in LNP COL FSAR Table 1.8-201 from the referenced AP1000 DCD and proposed additional departures. Section 1.3 of this SER provides a cross-reference to where these departures are discussed in this SER.

- LNP SUP 1.8-2

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

> *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.*

- LNP SUP 1.8-3

AP1000 DCD Table 1.8-1 presents interface items for the AP1000. This section of the DCD identifies certain interfaces with the standard design that have to be addressed in accordance with 10 CFR 52.47(a)(1)(vii).[8] As required by 10 CFR 52.79(d)(2), the COL application must demonstrate how these interface items have been met. In the LNP COL FSAR, the applicant did not explicitly identify how these interface items have been met. In a letter dated August 31, 2009, the applicant provided LNP COL FSAR Table 1.8-203, which explicitly identifies the FSAR location of information addressing the interface items identified in

---

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

Section 1.8 of the AP1000 DCD. The staff's review of the identified FSAR locations confirmed that interface items are adequately addressed in the LNP COL FSAR. The technical discussions related to specific interface requirements are addressed in related sections of this SER (e.g., SER Sections 8.2.4 and 11.3.2).

- LNP DEP 1.8-1

This Tier 2 departure, appearing in the FSAR Table 1.8-203 listing of AP1000 plant interfaces, corrects an error in 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 Appendix O of 10 CFR Part 50 to 10 CFR 52.137(a)(11). In issuing the final rule for 10 CFR Part 52 in the *Federal Register* (see 72 FR 49352; August 28, 2007), the requirement relating to providing this interface information was moved from Appendix O of 10 CFR Part 50 to a new location in 10 CFR 52.137 (see 72 FR 49391; August 28, 2007). 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.

<u>LNP COL FSAR Section 1.9</u>

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

The following portion of this technical evaluation section is reproduced from Section 1.4.4 of the VEGP FSER[9]:

> <u>*AP1000 COL Information Item*</u>
>
> - *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.*

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

<u>RAI 1-5</u>

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.

<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.

<u>Resolution of Standard Content Confirmatory Item 1.4-2</u>

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.

<u>Resolution of Standard Content Open Item 1.4-2</u>

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 above. The staff considers Open Item 1.4-2 resolved for VEGP.

Evaluation of Site-Specific Information Related to Standard Content (LNP COL 1.9-1)

In comparing VEGP COL FSAR Table 1.9-201 and Appendix 1AA to the respective tables in the LNP COL FSAR, the staff notes that there are several differences. These differences are associated with site-specific information and are reflected in the LNP COL FSAR by a "LNP COL 1.9-1" designation. The staff reviewed the site-specific differences in Table 1.9-201 and Appendix 1AA and has determined that the LNP 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 Section 1.4.4 of the VEGP SER:

- *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 steamline 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.*

- LNP SUP 1.9-1

LNP COL FSAR Table 1.9-202 contains both site-specific and standard information about the application's conformance with NUREG-0800. The technical discussions related to the specific acceptance criteria of NUREG-0800 are addressed in the related sections of the LNP COL FSAR and addressed in Chapters 2 through 19 of this SER, as needed.

- STD SUP 1.9-2

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

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

> - *STD SUP 1.9-3*
>
> *This COL supplemental item is addressed as VEGP SUP 8.1-2* [LNP SUP 8.1-3] *in SER Section 8.1*.

<u>LNP COL FSAR Section 1.10</u>

In this section of the application, the applicant provides 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).

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

- *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.

<u>Resolution of Standard Content Open Item 1.4-4</u>

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.

- LNP SUP 1.10-1

The supplemental information states that the power blocks for LNP Units 1 and 2 have a minimum separation of at least 900 feet between plant centerlines and notes that SSCs important to safety are described in LNP COL FSAR Chapter 3 and the LCOs for LNP Units 1 and 2 are identified in Part 4 of the COL application. In the standard portion of LNP COL FSAR Section 1.10, there is a discussion that the primary consideration in setting the 900-foot 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 resolution of Open Item 1.4-3.

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

<u>Resolution of Standard Content Open Item 1.4-3</u>

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 applicant and was discussed at a public meeting on August 26, 2010.

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 structures, 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:

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 to 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**.*

<u>Resolution of Standard Content Confirmatory Item 1.4-2</u>

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.

<u>License Conditions</u>

- *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

For the reasons discussed in the technical evaluation section above, the following FSAR commitment is identified as the responsibility of the licensee:

- **LNP Commitment Number 1.4-1 -** A site-specific construction plan and startup schedule will be provided after issuance of the COL.

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 principal review matters, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP 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:

Merger of Progress Energy with Duke Energy Corporation

On July 2, 2012, a merger occurred between Duke Energy Corporation (Duke) and Progress Energy Inc., the holding company of PEF. On February 6, 2013, PEF filed amended articles of incorporation with the Florida Department of State to change its corporate name to Duke Energy Florida, Inc. This name change became effective on April 29, 2013. Through this merger and subsequent name changes, Duke became the ultimate holding company of Progress Energy Inc. Progress Energy, Inc. continues to be the parent of Florida Progress Corporation, which is the direct parent of DEF. Following the July 2012 merger, Duke, the holding company and ultimate parent of DEF, is now the largest electric power holding company in the United States with more than $100 billion in total assets.

Duke Energy Florida, LLC

DEF, a subsidiary of Duke, is primarily engaged in the generation, transmission, distribution, and sale of electricity in portions of central and north Florida. DEF serves approximately 1.7 million customers in a territory encompassing over 20,000 square miles, including the cities of St. Petersburg, Clearwater, and areas surrounding Orlando. The address of the applicant is Duke Energy Florida, LLC, 299 First Avenue North, St. Petersburg, FL 33701. DEF is a corporation organized and existing under the laws of the State of Florida. DEF owns and operates Crystal River Nuclear Plant Unit 3, now in permanent shutdown mode, located near Crystal River, Florida, on a site that also includes four coal-fired generating units.

REGULATORY EVALUATION:

DEF's request (formerly a request by PEF) for the NRC to issue two COLs pursuant to Section 103 of the Atomic Energy Act of 1954, as amended, is subject to, among other things, the requirements of 10 CFR Part 52, Subpart C; 10 CFR Part 50; and 10 CFR Part 140. This SER reviews the following areas: financial qualifications, decommissioning funding assurance, antitrust, foreign ownership control or domination, and nuclear insurance and indemnity.

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.

Construction Cost Estimate:

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 has been 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 or an explanation given as to any departure therefrom.

In accordance with 10 CFR 50.33(f) and 10 CFR Part 50, Appendix C, DEF has estimated the construction costs for the two proposed units at the LNP site (LNP Units 1 and 2), which is provided in Part 9 of the LNP COL application. The costs are based upon a construction period for the project beginning in the third quarter of 2016 and ending with Unit 1 commercial operation in the third quarter of 2023, and Unit 2 commercial operation in the first quarter of 2025.

In its application, DEF described the basis for the foregoing cost estimate. DEF stated that the estimate was derived from the current LNP Total Project Cost analysis developed using cost estimates based on the best available information from internal and external sources for all aspects of plant costs. The estimate is consistent with the Florida Public Service Commission (FPSC) filing submitted on April 30, 2012, by DEF. LNP is expected to operate at an estimated gross electrical power output of approximately 2234 MWe (1117 MWe per unit).

The NRC staff reviews studies from independent sources and collects projected construction cost estimates from all COL applications, as they are submitted, for comparison and reasonableness.[10] According to these sources, the cost of constructing a plant comparable to LNP Units 1 and 2 ranges from approximately $3,221/kilowatt electric (kWe) to $5,072/kWe

[10] The staff's consideration of the cost information submitted by the applicant focused on the estimated production plant cost and cost of fuel. Cost estimates provided by the applicant were presumed to be true and accurate under 10 CFR 50.9, "Completeness and accuracy of information," and no further assessment of that estimate was performed.

(Massachusetts Institute of Technology [MIT] Study) installed.[11] Based, in part, on information provided by the applicant, staff independently calculated DEF's overnight cost per unit to be approximately $6,461,500,000. This is above the range derived from the studies developed from independent sources, and is also greater than construction cost estimates reviewed to date for comparable plants. In addition, based on estimated electrical power output of 1117 MWe per unit as reported by the applicant, staff independently calculated the construction cost of each LNP unit to be approximately $5,785/kWe. This value is derived by dividing the overnight cost per unit by the MWe output per unit. This value is also above the maximum construction cost per unit kilowatt electric cited above. Accordingly, based on data from independent sources and staff's analyses, the NRC staff finds DEF's overnight cost estimate to be reasonable.

Sources of Construction Funds:

Pursuant to 10 CFR Part 50, Appendix C, Section 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.

According to the COL application, in 2006, Florida enacted legislation that included cost recovery mechanisms supportive of nuclear plant investment. In 2007, the FPSC approved a new rule that allowed PEF to recover prudently incurred siting and preconstruction costs, and allowance for funds used during construction (AFUDC) on an annual basis through the capacity cost-recovery clause. The nuclear cost recovery rule also allows recovery of costs should a project be abandoned once the utility receives a final order granting a Determination of Need.

According to the COL application, DEF expects to finance this project through a combination of debt and equity in a manner that will support its investment grade credit ratings. The equity will come from DEF's retained earnings and equity contributions from Duke, as needed to maintain appropriate capital structures. Accordingly, the staff concludes that both DEF and Duke have sufficient financing capacity to fund this project from a number of sources: internally generated operating cash flows, commercial paper and bank facilities, and access to long-term debt and equity capital markets.

<u>Financial Qualifications</u>

Pursuant to 10 CFR Part 50, Appendix C, Section 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

[11] The 2009 update to the MIT interdisciplinary study entitled "The Future of Nuclear Power."

> all pertinent notes thereto and certification by a public accountant should be furnished.

Duke Energy Florida, LLC

DEF provided, at the time of application, financial statements filed with the Securities and Exchange Commission (SEC). Following the Duke Energy Corporation and Progress Energy 2012 merger, Duke Energy Corporation filed with the SEC a combined Form 10-Q which reflected financial information for DEF. Combined financial statements for Duke and DEF can be found at the SEC web site or by link through Duke's web site at:

http://www.duke-energy.com/investors/financials-sec-filings.asp?company=all

Prior to the 2012 merger and 2013 name change, PEF submitted, pursuant to 10 CFR Part 50, Appendix C, Section I.A.3, annual financial statements. Additionally, updated financial information was submitted to the NRC on December 7, 2015 (ADAMS Accession Nos. ML15349A770 and ML15349A100). The NRC staff performed an independent review of the applicant's December 7, 2015, financial information submittals and did not identify anything in DEF's, Duke's, or Progress Energy's, financial statements, submitted or otherwise, that warranted further inquiry.

In consideration of the foregoing, the NRC staff finds that the applicant has demonstrated it possesses or has reasonable assurance of obtaining the funds necessary to cover estimated construction costs and related fuel cycle costs. Therefore, the NRC staff finds that the applicant is financially qualified to construct the facilities.

Operating License

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

> If the application is for a combined license under subpart C of 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.

As previously discussed, Duke is a holding company that owns regulated and non-regulated subsidiaries, including DEF. DEF, the applicant for the proposed LNP 1 and 2 COLs, is primarily engaged in the generation, transmission, distribution, and sale of electricity in portions of central and north Florida. DEF is a regulated public utility, and is subject to the regulatory provisions of the Florida Public Service Commission, the NRC and the Federal Energy Regulatory Commission. DEF recovers the cost of electricity through rates established by the Federal Energy Regulatory Commission or the Florida Public Service Commission.

In consideration of the foregoing, the NRC staff finds that DEF is an electric utility and exempt from providing financial qualification information related to operating cost recovery. Because it is an electric utility, DEF is 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 § 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)).

Decommissioning Funding Estimate:

LNP is a two-unit PWR site that is incorporating by reference the Westinghouse AP1000 certified design, as documented in the referenced DCD including any supplemental material.

In its December 7, 2015, submittal to the NRC, DEF stated that it will provide decommissioning funding assurance in an amount of $373.4 million (2007 dollars) per unit. The NRC staff independently calculated the minimum funding acceptable under 10 CFR 50.75(c), and found the applicant’s amounts to be consistent with staff’s calculation and therefore acceptable.

Decommissioning Funding Mechanism:

DEF stated in the application that it would use an external sinking fund as the method to provide decommissioning funding assurance. Under 10 CFR 50.75(e)(1)(ii), an external sinking fund may be used as an exclusive method by a:

> . . . licensee that recovers, either directly or indirectly, the estimated total cost of decommissioning through rates established by 'cost of service' or similar ratemaking regulation.

The NRC staff will verify the acceptability of the decommissioning funding mechanism and prospective financial instrument in the future consistent with the schedule set forth in 10 CFR 50.75(e)(3) for the submission of reports by a holder of the COL.

Therefore, at this time, the NRC staff finds that DEF has complied with the applicable decommissioning funding assurance requirements.

ANTITRUST REVIEW:

The Energy Policy Act of 2005 (EPAct) removed the antitrust review authority in Section 105.c of the Atomic Energy Act of 1954 (AEA), as amended, regarding license applications for production or utilization facilities submitted under Sections 103 or 104.b 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 COL application.

FOREIGN OWNERSHIP, CONTROL, OR DOMINATION:

Section 103 of the AEA prohibits the Commission from issuing a license for a nuclear power plant under Section 103 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 50.38, "Ineligibility of certain applicants," is the regulatory provision that implements this statutory prohibition.

DEF is a corporation organized and existing under the laws of the State of Florida and is a wholly-owned subsidiary of Florida Progress Corporation. Progress Energy and Duke Energy Corporation merged on July 2, 2012. Following the merger, Duke Energy Corporation became the ultimate holding company of Progress Energy, Inc., and Progress Energy, Inc., continues to be the parent of Florida Progress Corporation, which is the direct parent of DEF. Duke Energy Corporation is the ultimate parent of DEF.

By letter dated December 7, 2015 (ADAMS Accession ML15349A100, Duke Energy Corporation notified the NRC of its corporate name change from a "corporation" to an "LLC", as well as changes to its Board of Directors, executive officers, and senior nuclear leadership team. The COL application includes the names and addresses of the directors and officers of Duke Energy Corporation and indicates that all are United States citizens. According to the COL application, neither Duke Energy Corporation, Progress Energy, Inc., Florida Progress Corporation, nor DEF are owned, controlled, or dominated by any alien, foreign corporation, or foreign government. The COL application was originally filed by PEF on its own behalf and not as an agent or representative of any other person.

As described above and in the application there is a Board of Directors for Duke Energy. There is also a separate Board of Directors for DEF. The business of DEF is conducted by its own Board of Directors, although for internal governance purposes, the Duke Energy Corporation Board of Directors also has approval authority over certain types of transactions. All members of the senior management and the Board of Directors for Duke Energy Corporation and for DEF are United States citizens. Staff conducted an independent analysis, including open-source research and verification of the information provided in the application and found no evidence of foreign ownership, control, or domination.

Based on its review, the NRC staff does not know or have reason to believe that Duke Energy Corporation or any of its subsidiaries, including DEF, are foreign owned, controlled, or dominated.

NUCLEAR INSURANCE AND INDEMNITY:

This section of the SER addresses the applicant's offsite and onsite insurance requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) 140.11(a)(4) and 10 CFR 50.54(w), as well as the requirements of 10 CFR 140.21, "Licensee guarantees of payment of deferred premiums," and 10 CFR 140.20, "Indemnity agreements and liens."

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 require, in part, each holder of a license issued pursuant to 10 CFR Part 50 or 10 CFR Part 52 to have and maintain financial protection. Under these regulations, DEF is required to provide satisfactory documentation that it has obtained financial protection required by 10 CFR 140.13, "Amount of financial protection required of certain holders of construction permits and combined licenses under 10 CFR Part 52," 10 CFR 140.11(a)(4), at least the amount of financial protection required by 10 CFR 50.54(w), and evidence that it maintains a guarantee of payment of deferred premiums pursuant to 10 CFR 140.21. In addition, as required by 10 CFR 140.20, DEF will enter into an agreement of indemnity with the NRC.

The requirements in 10 CFR 140.13 provide the amount of financial protection required by a license holder, who also holds a license under 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," during the period of construction and before the Commission makes the finding under 10 CFR 52.103(g) (i.e., a finding that a nuclear reactor is authorized to initially load fuel and operate). Because the Part 70 license will be issued with the COL, DEF 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. By letter dated May 1, 2015 (ADAMS Accession No. ML15126A181), DEF's insurance broker, Marsh USA, Inc., provided proof of insurance coverage from American Nuclear Insurers in the amount of $1,000,000. On January 19, 2016, DEF supplemented this submittal with a certificate of insurance that reflects updated insurance coverage for 2016. DEF's $1,000,000 insurance policy will remain in effect until the 52.103(g) finding. Therefore, the staff concludes that the proof of financial protection provided by DEF will satisfy the requirements in 10 CFR 140.13.

The staff notes that although licensees of large operating reactors under Parts 50 and 52 must have and maintain financial protection upon NRC action authorizing operation, the timing provisions for reporting under 10 CFR Part 140 and 10 CFR Part 50 are not the same as for Part 52 licenses; these regulations do not specifically address the Part 52 process. Thus, under the requirements in 10 CFR 140.11(a)(4), 10 CFR 50.54(w), and 10 CFR 140.21, the NRC staff notes that coverage for offsite and onsite insurance, and the guarantee of payment of deferred premiums, are only required for reactors authorized to load fuel and operate. Under the Part 52 COL process, this is the time period beginning once the 52.103(g) finding has been made by the Commission, which also authorizes a licensee to load fuel and operate. Therefore, these requirements will be deferred until the date that the 52.103(g) finding has been made by the Commission. This time period is consistent with the time period under Part 50 for which an operating license has been granted. As such, the staff proposes the following license conditions to meet the requirements in 10 CFR 140.11(a)(4), 10 CFR 50.54(w), and 10 CFR 140.21.

The staff proposes the following license condition to address the deferred reporting of 10 CFR 140.11(a)(4) requirements for primary and secondary financial protection, and the deferred reporting of 10 CFR 50.54(w) requirements for onsite financial protection:

> License Condition (1-1) - Before the scheduled date for initial fuel load, DEF 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 financial protection (insurance) required of licensees pursuant to 10 CFR Part 140 and 10 CFR 50.54(w).

With the license condition as described above, the staff finds that DEF will satisfy the requirement of 10 CFR 140.11(a)(4) and 10 CFR 50.54(w).

The staff proposes the following license condition to address the deferred reporting of 10 CFR 140.21 for guarantee of payment of deferred premiums:

> 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*, the licensees shall provide evidence to the NRC that they 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) 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.

With the license condition as described above, the staff concludes that DEF will satisfy the requirement in 10 CFR 140.21. Thereafter, the licensee shall provide evidence of the guaranteed payment of deferred premiums in accordance with the timing provisions specified in 10 CFR 140.21.

For these two license conditions, the staff notified the applicant of the above-proposed language, and the applicant accepted the license conditions (ADAMS Accession No. ML16084A126).

As required by 10 CFR 140.20, the Commission will enter into an indemnity agreement with DEF concurrent with the issuance of a license (issued under 10 CFR Part 70) authorizing the licensee to possess and store special nuclear material at the site of the nuclear reactor after issuance of an operating license. This agreement will also address indemnity as described in 10 CFR 140.92, “Appendix B – Form of indemnity agreement with licensees furnishing insurance policies as proof of financial protection,” between the period when the 10 CFR Part 70 license is issued and the time the 52.103(g) finding has been made by the Commission.

CONCLUSION:

Based on the foregoing, and the updated information provided to the NRC on December 7, 2015, the NRC staff finds that there is reasonable assurance that DEF is financially qualified to engage in the proposed activities regarding LNP Units 1 and 2, as described in the application, and that there are no decommissioning funding assurance, foreign ownership, control, or domination, or nuclear insurance and indemnity issues.

1.5.2 Nuclear Waste Policy Act

Section 302(b) of the Nuclear Waste Policy Act of 1982, as amended, states, “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.”

In a letter dated February 4, 2009, the applicant stated that on December 18, 2008, it signed contracts with the U.S. Department of Energy (DOE) establishing the terms and conditions applicable to the DOE’s responsibility for disposal of spent nuclear fuel and high-level radioactive waste generated at the proposed LNP Units 1 and 2. The DOE contract numbers referenced in the letter are DE-CR01-09RW09019 for LNP Unit 1 and DE-CR01-09RW09020 for LNP Unit 2.

Because Progress Energy has entered into contracts with the DOE for the disposal of high-level radioactive waste and spent nuclear fuel for LNP Units 1 and 2, the staff considers that the applicable requirements of Section 302(b) of the Nuclear Waste Policy Act of 1982 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 Section 657 of the *Energy Policy Act of 2005*, the NRC consulted with the Department of Homeland Security (DHS) with respect to the PEF COL application for LNP Units 1 and 2. Between February 17, 2009, and February 19, 2009, DHS conducted a site visit and was accompanied by NRC staff (ADAMS Accession No. ML091950039). On August 31, 2009, NRC issued a DHS consultation report regarding the DHS site visit with the applicant (ADAMS Accession No. ML091960397). The DHS report concludes that the applicant and the NRC staff have satisfied 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 10 CFR 50.43(a), on December 15, 2011, the NRC notified the Public Service Commission of Florida of the LNP COL application (ADAMS Accession No. ML112521258). In addition, in November and December 2008, the NRC published notices of the application in *The Newscaster/Nature Coast News*, the *Ocala Star Banner*, the *Levy County Journal*, and the *Citrus County Chronicle*. In accordance with Section 182c., the staff also published a notice of the 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 completion of notifications to regulatory agencies and the public notices described above, the staff concludes that, for the purposes of issuing COLs for LNP 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 Numbering in the Application and Exemption Associated with Special Nuclear Material Control and Accounting Program

Evaluation of Departures and Exemption Associated with Numbering in the Application

In STD DEP 1.1-1, the applicant renumbered LNP COL FSAR sections to include content consistent with RG 1.206 and NUREG-0800. The departure and the exemption associated with the numbering scheme of the FSAR are closely related. The departure provided in Part 7 of the COL application provides the specific sections of the LNP COL FSAR that deviate from the DCD numbering scheme.

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 Part 7, "Departures and Exemptions," of the LNP COL application, the applicant states 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 NRC review. The applicant states that the subject deviations are considered 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 DC rule, then the Commission may grant the exemption if the exemption complies with the appropriate change provision in the referenced DC 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 DC 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 DC 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 DC 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 DC information (e.g., Tier 1 and Tier 2) that is incorporated by reference in the DCRs," (72 FR 49444; August 28, 2007). 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.

Because there is not an applicable change provision in the referenced DC, 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) to be acceptable for the review of the exemption related to STD DEP 1.1-1.

Pursuant to the exemption described above, the NRC staff has reviewed the exemption related to STD DEP 1.1-1 to determine whether it meets the requirements in 10 CFR 52.7. This exemption would allow the applicant to provide an FSAR with numbering and topics more closely related to NUREG-0800 and RG 1.206, and 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 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 LNP COL FSAR to be acceptable.

Exemption Associated with Special Nuclear Material Control and Accounting Program

In a letter dated April 19, 2011, 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; 10 CFR 74.41; or 10 CFR 74.51[12]. 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 as required under 10 CFR Part 74 Subparts C through E and to request Commission approval prior to implementing program changes. 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 applicant stated that 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 applicant stated that 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 Parts 70 and 74, respectively. Therefore, by demonstrating that the exemption criteria in 10 CFR 50.12 are satisfied, 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 applicant stated that the subject exemption would allow nuclear reactors licensed under 10 CFR Part 52 to be explicitly excepted 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 *Federal Register* 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 Parts 70 and 74). In particular, the exception for nuclear reactors licensed under

[12] 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.

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.

In addition, pursuant to 10 CFR 74.7, 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 NRC 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 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 PEF's letter transmitting Revision 2 of the COL application, dated October 4, 2010, and in Part 1, "General and Financial Information," of the application, PEF 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 COL encompass those necessary to support granting 10 CFR Parts 30, 40, and 70 licenses. In this respect, the 10 CFR Part 52 COLs for LNP 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 LNP COL as would relate to authorization pursuant to the regulations in 10 CFR Parts 30, 40, and 70[13]:

> Subject to the conditions and requirements incorporated herein, the Commission hereby licenses DEF:
>
> (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

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

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 LNP 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”)

- Physical Protection 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 a letter dated April 19, 2011, the applicant proposed to revise the LNP COL FSAR Table 13.4-201 to add information (milestones and requirements) related to the SNM MC&A program. In addition, as documented in the Table 1-3, the LNP applicant endorsed VEGP standard content letters related to this subject.

Table 1-3. LNP COL Applicant Endorsements of VEGP COL Standard Content Letters

VEGP Letter Date	VEGP Letter ADAMS Accession Number	LNP Endorsement Letter Date	LNP Letter ADAMS Accession Number
July 9, 2010	ML101940025	September 23, 2011	ML102740219
July 29, 2009	ML092120064	December 7, 2009	ML093450351
October 15, 2010	ML102920120	April 19, 2011	ML11111A125
November 23, 2010	ML103300034	April 19, 2011	ML11111A125
March 3, 2011	ML110660153	April 19, 2011	ML11111A125
March 16, 2011	ML110800088	April 19, 2011	ML11111A125
March 16, 2011	ML110770137	April 19, 2011	ML11111A125
May 6, 2011	ML11129A155	July 28, 2011	ML11213A096
June 22, 2011	ML11175A169	July 28, 2011	ML11213A096

These letters identify the portions of the LNP COL application that satisfy the basis for meeting the requirements of 10 CFR Parts 30, 40, 70, and 74. In addition, in a letter dated April 19, 2011, 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. This exemption request is addressed in Section 1.5.4 of this SER.

Additionally, in a letter dated November 20, 2014, submitted in response to RAI Letter No. 120 dated July 2, 2014, the applicant provided a revised physical protection program for SNM possessed onsite prior to establishment of a protected area per 10 CFR 73.55 to meet the requirements of 10 CFR 73.67.

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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 confirmed that the April 19, 2011, LNP submittal endorses the SNM MC&A Program description submitted by Southern Nuclear Operating Company (SNC) in a letter dated November 23, 2010.

- The staff confirmed that the July 28, 2011, LNP submittal endorses the VEGP New Fuel Shipping Plan submitted by SNC in a letter dated May 6, 2011.

- The staff confirmed that the supplemental information in support of 10 CFR Part 70 SNM license application found in Part 11 of the VEGP COL application is identical to the material found in Part 11 of the LNP COL application.

- The staff verified that 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 incorporation of the LNP SNM MC&A Program description, the SNM physical protection plan (SNMPPP), and the new fuel shipping plan into the LNP COL application is **LNP Confirmatory Item 1.5-1**.

<u>Resolution of LNP Confirmatory Item 1.5-1</u>

Confirmatory Item LNP 1.5-1 is an applicant commitment to revise the LNP COL application to include the LNP SNM MC&A Program description, the SNMPPP, and the new fuel shipping plan. For the SNM MC&A Program description and the new fuel shipping plan, the staff verified that the LNP COL application was appropriately revised. For the SNMPPP, the applicant submitted a revised plan, as described above. The staff review of the revised plan appears below, following the review of the standard content material. As a result, Confirmatory Item LNP 1.5-1 is now closed.

The staff has completed its review and found the evaluation performed for the standard content to be directly applicable to the LNP COL application, with the site-specific exceptions noted. 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 1.5.5 of the VEGP SER:

> *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.

<u>Review of Parts 30 and 40 Materials</u>

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**.*

<u>Resolution of Standard Content Confirmatory Item 1.5-1</u>

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 [Part 11 of the LNP COL 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.

<u>*MC&A Program for SNM (Fuel)*</u>

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 1.5-3 stating that no SNM onsite will exceed a 10-percent uranium-235 isotope enrichment level, the applicant updated its COL application to include Part 11F, "Supplemental Information of 10 CFR Part 70 Special Nuclear Material License Application" acknowledging that LNP 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 Section 1.5.5 of the VEGP SER:

> *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.*

<u>Resolution of Standard Content Confirmatory Item 1.5-2</u>

Confirmatory Item 1.5-2 is an applicant commitment to revise 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) [Part 11 of the LNP COL application] *of its application were appropriately revised. As a result, Confirmatory Item 1.5-2 is now closed.*

The following portion of this technical evaluation section is reproduced from Section 1.5.5 of the VEGP SER. Portions of the standard content review addressing SNM physical protection superseded by the staff's review of additional site-specific information have been deleted from the standard content review appearing below. The staff review of the additional site-specific information, including a revised SNMPPP appears below following the review of the standard content material.

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

[Standard content deleted as noted above]

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.

[Standard content deleted as noted above]

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 E) are appropriately updated.

[Standard content deleted as noted above]

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.

[Standard content deleted as noted above]

<u>Non-Fuel SNM</u>

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.

<u>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</u>

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: LNP COL FSAR Section 1.4.1 has a similar discussion regarding the applicant's operation of its other nuclear power plants. The staff's evaluation of the technical qualifications of the applicant appears in Section 13.1 of this SER. As discussed in Section 1.4 of the SER, the staff also concludes the applicant is technically qualified to engage in the proposed activities associated with this license based on the applicant's on-going experience with the safe operation of its other nuclear power plants. In addition, Section 1.5.1 of this report finds that the financial qualifications for the LNP COL application are acceptable.

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

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).

<u>Emergency Plan (SNM, Fuel, and Non-Fuel)</u>

The applicant will be storing the new fuel in the new fuel racks (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 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 site PSP.

1.5.5.1.2 Summary of Application

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 5.66, “Access Authorization Program for Nuclear Power Plants,” Revision 1, July 2009 (Official Use Only – Security-Related Information)

- 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).

- 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. Regulatory Guide 5.59 (1983).”

1.5.5.1.4 Technical Evaluation

The staff performed a technical evaluation of the LNP Units I and 2 COL application 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. In addition, the staff requested information related to how the applicant addressed the post September 11, 2001, security order measures for SNM of low strategic significance (nonpublic). The staff sent a letter conveying the order measures on July 2, 2014, (ADAMS Accession No. ML141813240) and the safeguards information containing orders were sent under separate cover (Safeguards Lan Electronic Safe (SLES) Accession No. NS113122). A technical evaluation of how the order measures were addressed was also performed. The applicant submitted a letter on November 20, 2014, that 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. ML14325A657).

1.5.5.1.4.1. Fixed Site General Performance Objectives

The applicable physical protection 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 performance objectives for fixed sites.

The physical protection requirements of 10 CFR 73.67(a)(1), stated, “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 LNP 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 states in the “Implementation Milestone” column of FSAR Table 13.4-201 under Item 15 (ADAMS Accession No. ML14258A229) its commitment to meet the requirements of 10 CFR 73.67 “[p]rior to initial receipt of special nuclear material.” Establishment of the physical protection system is outlined in the SNMPPP, Revision 1, dated September 2014 (SLES Accession No. NS113156 (nonpublic)). Specifically, Section 4.4.1 “Establishment of the Physical Protection System,” describes 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, Section 4.4.2, “Maintenance of the Physical protection System," of the SNMPPP contains an explanation of the maintenance that will be applied to the physical protection system.

The DEF application states that 10 CFR 73.67 will be fully implemented before SNM is onsite. Also, the application outlined establishment and maintenance elements for the physical protection system. The establishment of physical protection elements is sufficient because before the physical protection infrastructure will be considered operational: 1) the lighting necessary for human detection through visual observation will be tested and confirmed as adequate, 2) visual assessment systems will be tested as functioning as necessary to support security operations, 3) alarm stations will be validated as having the ability to adequately support physical security activities for the protection of the SNM of low strategic significance, 4) communication technologies that are to be relied upon to enable the physical security strategy to operate effectively will be tested and confirmed to allow for intelligible voice interfaces, 5) the means of access control will be tested for its performance to support the physical security strategy, and 6) the physical barriers that provide containment of the SNM of low strategic significance will be inspected to ensure a comprehensive impediment to personnel entry is in place. The development of a maintenance program for the six physical protection elements established is committed to in the application. In addition, the application states that the maintenance program will have periodicity of maintenance configured for each of the six physical protection system elements that is commensurate with each of the elements' intended function. Therefore, the staff finds that 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 states in the "Implementation Milestone" column of FSAR Table 13.4-201 under Item 15 its commitment to meet the requirements of 10 CFR 73.67, "[p]rior to initial receipt of special nuclear material." In addition, the SNMPPP describes in Section 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 explaining how the physical protection system will work to meet the requirement. In addition, the SNMPPP within Section 5.3.2, "Monitoring SNM (New Fuel Assemblies)," describes adversary scenarios applied to SNM reactor fuel and explains how the physical protection system will work to meet this requirement as well.

The DEF application states that 10 CFR 73.67 will be fully implemented before SNM is delivered. In addition, its SNMPPP describes how the possibilities for unauthorized removal are minimized in ways consistent with the consequences of such actions. The application describes potential adversarial scenarios for all activities involving SNM of low strategic significance and highlights how the six physical protection system elements work in a coordinated fashion to adequately minimize the risk of theft of the materials. Therefore, the staff finds that 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 SNM in ways 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 states in the "Implementation Milestone" column of FSAR Table 13.4-201 under Item 15 its commitment to meet the requirements of 10 CFR 73.67, "[p]rior to initial receipt of special nuclear material." 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 SNM.

The DEF application states that 10 CFR 73.67 will be fully implemented before SNM is delivered. In addition, its SNMPPP describes the detection, assessment and response attributes of the physical protection system that would facilitate the location and recovery of missing SNM. The application explicitly points out how the planned-for detection and assessment physical protection system elements function to provide adequate detection and assessment of malevolent activities in order to initiate a specific response that would enable the location and recovery of SNM of low strategic significance. Scenarios that depict adversary actions, operation of physical security system elements, and security force response activities provide assurance that the requirement of 10 CFR 73.67(a)(1)(ii) would be met. Therefore, the staff finds that 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 SNM) 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 the "Implementation Milestone" column of FSAR Table 13.4-201 under Item 15 its commitment to meet the requirements of 10 CFR 73.67, "[p]rior to initial receipt of special nuclear material." In addition, its SNMPPP in Sections 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 SNM.

The DEF application states that 10 CFR 73.67 will be fully implemented before SNM is delivered. In addition, their SNMPPP describes the physical protection strategies for early detection and assessment to address unauthorized access or activities by an external adversary within the controlled access area containing SNM. These physical protections strategies are consistent with staff guidance in RG 5.59. 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 SNM) 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 states in the "Implementation Milestone" column of FSAR Table 13.4-201 under Item 15 its commitment to meet the requirements of 10 CFR 73.67, "[p]rior to initial receipt of special nuclear material." In addition, its SNMPPP in Sections 5.3.1 and 5.3.2 describes how the physical protection system provides early detection of removal of SNM by an external adversary from a controlled access area.

The DEF application states that 10 CFR 73.67 will be fully implemented before SNM is delivered. In addition, its SNMPPP describes the physical protection strategies for early detection and assessment to address removal of SNM by an external adversary from a controlled access area. These physical protections strategies are consistent with staff guidance in RG 5.59. Therefore, the staff finds that the requirement of 10 CFR 73.67(a)(2)(ii) (to have a physical protection system that provides early detection of removal of SNM 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 states in the "Implementation Milestone" column of FSAR Table 13.4-201 under Item 22 its commitment to meet the requirements of 10 CFR Part 74, "Material Control and Accounting of Special Nuclear Material" "[p]rior to receipt of special nuclear material" as a license condition. Also, the applicant states in Part 11D of the COL application, "Special Nuclear Material (SNM) Material Control and Accounting Program Description," (ADAMS Accession No. ML14258A226) that the applicant will establish, a "…SNM control and accounting system… including…internal control, physical inventory, and shipment of SNM."

In addition, the applicant describes in its SNMPPP in Sections 5.1.1, "Receipt of Non-Fuel SNM," 5.1.2, "Receipt of SNM - Fuel Assemblies/Fuel Components," (pertaining to fuel SNM), and 5.8, "Internal Transfers," material control and accounting (MC&A) measures specific to the non-fuel and fuel SNM, respectively.

The DEF application states that the appropriate provisions of 10 CFR Part 74 will be fully implemented before SNM is received. The application also states that: 1) notification will be made to the shipper upon receipt of the SNM of low strategic significance; 2) an investigation will be initiated as required per 10 CFR 73.67 and 10 CFR74.11 if the shipment is not received as scheduled; 3) the NRC Operations Center will be notified within an hour after assessing that a shipment has not arrived and/or within an hour of SNM of low strategic significance recovery; 4) the licensee will conduct an inspection of tamper seal devices on containers of SNM of low strategic significance after accessing the shipment conveyance that has been received at the nuclear reactor facility; and 5) the licensee will verify that the shipment is consistent with the shipment's manifest in regard to identification markings and numbers of SNM containers. The applicant has described in their SNMPPP how specific MC&A measures apply to meet this general performance objective; therefore, the staff finds that the requirement of 10 CFR 73.67(a)2(iii) (to assure proper placement and transfer of custody of SNM) 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 states in the "Implementation Milestone" column of FSAR Table 13.4-201 under Item 15 its commitment to meet the requirements of 10 CFR 73.67, "[p]rior to initial receipt of special nuclear material." In addition, its SNMPPP in Section 5.10, "Contingency Response," 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 the local law enforcement (LLEA) agency, etc.) was pointed out in the SNMPPP by citing Section 8, "LOCAL LAW ENFORCEMENT LIASON," of the reactor PSP, Revision 4, dated June 3, 2011 (SLES Accession No. NS108206), and Sections 5.6, "LOCAL LAW ENFORCEMENT AGENCIES (LLEA)," 5.7, "STATE RESPONSE ACTIONS," and 5.8, "FEDERAL RESPONSE ACTIONS," of the reactor Contingency Plan, Revision 4, dated June 3, 2011 (SLES Accession No. NS108206).

The DEF application states that 10 CFR 73.67 will be fully implemented before SNM is delivered. In addition, its SNMPPP describes the early detection, assessment, and response physical protection strategies that would facilitate recovery of missing or stolen SNM. Specifically, the applicant described in the SNMPPP detection, assessment, communication, and response scenarios associated with all locations of SNM of low strategic significance. In addition, the response protocols described are consistent with both RG 5.59 and the response measure criteria in the post September 11, 2001, SNM of low strategic significance security order. Therefore, the staff finds that 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 SNM 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 three 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 three 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 three years. . ." "Copies of the superseded material must be retained for three 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 SNMPPP describes the required retention parameters for the SNMPPP and changes to it. Therefore, the staff finds that 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 states in the "Implementation Milestone" column of FSAR Table 13.4-201 under Item 15 its commitment to meet the requirements of 10 CFR 73.67, "[p]rior to initial receipt of special nuclear material."

Additionally, the staff proposes to impose the following license condition, based on License Condition 6, as listed in Part 10 of the COL application:

> 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.

In the application, the applicant has stated that the requirements of 10 CFR 73.67 will be implemented before SNM is received. Also, a license condition has been applied to ensure the NRC staff is aware of the scheduled date for implementation of the requirements of 10 CFR 73.67. Therefore, the requirement to either implement the SNMPPP within 30 days after NRC approval of it, or as designated by the NRC in writing, will be met, by the required schedule for implementation of operational programs.

1.5.5.1.4.3. Fixed Site 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," include 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 three 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 three 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 LNP 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 states in the "Implementation Milestone" column of FSAR Table 13.4-201 under Item 15 its commitment to meet the requirements of 10 CFR 73.67, "[p]rior to initial receipt of special nuclear material." In addition, its SNMPPP in Sections 5.2, "Storage," and 5.8, "Internal Transfers," and Figures 1 through 13 describes the physical characteristics of the controlled access area. The description of the controlled access area depicted in the SNMPPP includes temporary and permanent controlled access areas to enable protection during receipt and long-term storage of SNM, respectively. In addition, the described physical characteristics of the controlled access area are consistent with the recommended penetration resistance features explained in RG 5.59. Furthermore, as described in the application, both the fuel SNM and non-fuel SNM of low strategic significance will always be protected within a controlled access area. The non-fuel SNM is described as only being removed from the its controlled access area and into its functioning location after the protected area of the nuclear reactor has been established per 10 CFR 73.55(e)(8), which is an acceptable practice because when the SNM is located inside a protected area, it is provided adequate protection.

The DEF application states that 10 CFR 73.67 will be fully implemented before SNM is delivered. In addition, the SNMPPP describes the characteristics of their planned-for controlled access area; therefore, the staff finds that 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 the "Implementation Milestone" column of FSAR Table 13.4-201 under Item 15 its commitment to meet the requirements of 10 CFR 73.67, "[p]rior to initial receipt of special nuclear material." In addition, its SNMPPP in Sections 5.3, "Monitoring," 5.3.1, "Monitoring SNM (Non fuel SNM…," and 5.3.2, "Monitoring SNM (New Fuel Assemblies)," 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.

The DEF application states that 10 CFR 73.67 will be fully implemented before SNM is received. In addition, its SNMPPP describes the detection processes that would result in recognition of unauthorized penetrations or activities in the locations of SNM and the controlled access area. Specifically, the applicant described in its SNMPPP the detection techniques and assessment methods that would result in a high probability of detection and accurate assessment of malevolent acts or potentially malevolent indications. In addition, administrative controls were described in the SNMPPP that would reduce the risk of not detecting a malevolent act or indications of potential malevolent acts to an acceptable level. 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 states in the "Implementation Milestone" column of FSAR Table 13.4-201 under Item 15 its commitment to meet the requirements of 10 CFR 73.67, "[p]rior to initial receipt of special nuclear material." In addition, its SNMPPP in Sections 5.3.1, "Monitoring SNM (Non fuel SNM…," 5.3.2, "Monitoring SNM (New Fuel Assemblies)," and 5.10, "Contingency Response," 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 PSP, Revision 4, dated June 3, 2011 (SLES Accession No. NS108206), and Sections 5.6, 5.7 and 5.8 of the reactor Contingency Plan, Revision 4, dated June 3, 2011 (SLES Accession No. NS108206).

The DEF application states that 10 CFR 73.67 will be fully implemented before SNM is received. In addition, its SNMPPP and other information referenced in the SNMPPP describe the detection, assessment, and response measures for the physical protection of the material. The applicant provided details in the SNMPPP of the protocols of detection, assessment, communications, and response that would work to adequately protect the SNM. In addition, those protocols or both onsite and offsite response actions were committed to be developed and implemented via written procedures. Therefore, the staff finds that 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 three 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 three years after each change."

The applicant states in the "Implementation Milestone" column of FSAR Table 13.4-201 under Item 15 its commitment to meet the requirements of 10 CFR 73.67, "[p]rior to initial receipt of special nuclear material." In addition, Sections 4.1, "Procedures," 5.3.1, "Monitoring SNM (Non fuel SNM…," 5.3.2, "Monitoring SNM (New Fuel Assemblies)," 5.7, "Audits and Records," and 5.10, "Contingency Response," of the SNMPPP describe the framework of and details to the development of response procedures. In addition, Section 5.7 "Audits and Records," of the SNMPPP notes the retention for 3 years of response procedures and changes thereof.

The DEF application states that 10 CFR 73.67 will be fully implemented before SNM is delivered. In addition, its SNMPPP describes the framework of the response procedures, the details on the development of response procedures, and the retention actions of 3 years of the response procedures; therefore, the staff finds that 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 LNP Units I 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, "Shipment," of its SNMPPP. The SNMPPP states that a SNM-qualified licensed shipper, other than DEF, 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 DEF 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 DEF 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 SNMPPP states that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance and that DEF will confirm that the licensed shipper has physical protection measures in place to meet each general performance objective; therefore, the staff finds that 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 its SNMPPP. The SNMPPP states that a SNM-qualified licensed shipper, other than DEF, 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 DEF 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 DEF 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 SNM consistent with the potential consequences of such actions.

The DEF application states that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance, and that DEF will confirm that the licensed shipper has physical protection measures in place to meet each general performance objective; therefore, the staff finds that the requirement of 10 CFR 73.67(a)(1)(i) to minimize the possibilities for unauthorized removal of SNM 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. The SNMPPP states that a SNM-qualified licensed shipper, other than DEF, 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 DEF 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 DEF 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 SNM.

The SNMPPP states that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance and that DEF will confirm that the licensed shipper has physical protection measures in place to meet each general performance objective; therefore, the staff finds that 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 intends 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 DEF, 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 DEF 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 DEF 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 SNM.

The DEF application states that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance and that DEF will confirm that the licensed shipper has physical protection measures in place to meet each general performance objective; therefore, the staff finds that the requirement of 10 CFR 73.67(2)(i) to provide, "[e]arly 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 described 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 DEF, 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 DEF 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 DEF 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 SNM by an external adversary from a controlled access area.

The SNMPPP states that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance, and that DEF will confirm that the licensed shipper has physical protection measures in place to meet each general performance objective; therefore, the staff finds that the requirement of 10 CFR 73.67(2)(ii) to provide, "[e]arly 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 described 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 DEF, 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 DEF 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, DEF has described the process for receiving and placing SNM in Sections 5.1.1, "Receipt of Non-Fuel SNM," and 5.1.2 (for fuel SNM) of its SNMPPP. 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 DEF 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, 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 DEF 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 SNM.

The DEF application states that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance, and that DEF will confirm that the licensed shipper has physical protection measures in place to meet each general performance objective. In addition, DEF 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 that the requirement of 10 CFR 73.67(2)(iii) to assure proper placement and transfer of custody of SNM 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 described how it intended to meet the in-transit physical protection requirements of 10 CFR 73.67(g) in Section 6 of their SNMPPP. The SNMPPP states that a SNM-qualified licensed shipper, other than DEF, 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 DEF 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 DEF 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 SNM and then notify the appropriate response forces of its removal in order to facilitate its recovery.

The DEF application states that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance, and that DEF will confirm that the licensed shipper has physical protection measures in place to meet each general performance objective; therefore, the staff finds that the requirement of 10 CFR 73.67(a)(2)(iv) (to respond to indications of an unauthorized removal of SNM 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 three 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 three 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 three years. . .” “Copies of the superseded material must be retained for three years after each change.”

The applicant described 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 DEF, 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 DEF 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.

The DEF application states that 10 CFR 73.67 will be fully implemented before SNM is received. In addition, its 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 DEF application states that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance, and that DEF will confirm that the licensed shipper has provisions in place to meet 10 CFR 73.67(c)(1); therefore, the staff finds that the requirements of 10 CFR 73.67(c)(1), as described above, 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 intends 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 DEF, 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 DEF 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.

The DEF application states that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance and that DEF will confirm that the licensed shipper has provisions in place to meet 10 CFR 73.67(c)(1); therefore, the staff finds that the requirement of 10 CFR 73.67(c)(1) (to "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") 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 three 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 three 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 LNP 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 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 DEF, 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 DEF 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. Because DEF 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), subsequently 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 DEF application stated that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance, and that DEF 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 that 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 its SNMPPP. The SNMPPP states that a SNM-qualified licensed shipper, other than DEF, will be used to transport SNM of low strategic significance both to and from the site. In addition, Section 6 of the SNMPPP states that DEF 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 DEF 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 DEF application states that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance, and that DEF 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 that 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. The SNMPPP states that a SNM-qualified licensed shipper, other than DEF, 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 DEF 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 DEF application states that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance, and that DEF 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 that 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 its SNMPPP. Specifically, Sections 5.1.1.4 (for non-fuel SNM) and 5.1.2.4 (for fuel SNM) state that the integrity of both shipping containers and tamper-seals will be checked.

The DEF application states that shipment containers and tamper-seals applied to those containers would be checked upon receipt; therefore, the staff finds that 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. . ."

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

The DEF application states that shipper would be notified in accordance with 10 CFR 74.15 for both non-fuel and fuel SNM; therefore, the staff finds that 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 its SNMPPP. The SNMPPP states that a SNM-qualified licensed shipper, other than DEF, 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 DEF 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. Because DEF 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)(2)(iii), the staff finds that this requirement would be met.

The DEF application states that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance, and that DEF will confirm that the licensed shipper has provisions in place to meet10 CFR 73.67(g)(2)(iii). Therefore, the staff finds that 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 three 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 three years after each change."

The applicant included a description of how it intends 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 DEF, 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 DEF 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 DEF application states that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance, and that DEF will confirm that the licensed shipper has provisions in place to meet 10 CFR 73.67(g)(3)(i); therefore, the staff finds that the requirement of 10 CFR 73.67(g)(3)(i) to, "[e]stablish and maintain response procedures …," as described above, 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 intends 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 DEF, 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 DEF 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 states that DEF will use an SNM licensed shipper and that DEF will verify that the shipper will be able to meet the requirement.

The DEF application states that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance, and that DEF will confirm that the licensed shipper has provisions in place to meet10 CFR 73.67(g)(3)(ii). Therefore, the staff finds that 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 intends 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 DEF, 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 DEF 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. DEF has committed to meeting the requirement in Sections 5.1.1.1 (for non-fuel SNM) and 5.1.2.1 (for fuel SNM) of its SNMPPP. Also, DEF noted that a procedure would be developed for notification processes in Section 4.1 of the SNMPPP.

The DEF application states that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance, and that DEF will confirm that the licensed shipper has provisions in place to meet 10 CFR 73.67(g)(3)(iii). In addition, DEF has committed to meeting the 10 CFR 73.67(g)(3)(iii) trace investigation/notification requirement. Therefore, the staff finds that 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 three 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 three years after each change."

How the requirements of 10 CFR 73.67(c) would be met by the applicant are described above in Section 1.5.5.1.4.2 of this SER. Also, the applicant included a description of how it intends 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 DEF, 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 Duke Energy 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 DEF application states that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance and that DEF 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 above in Section 1.5.5.1.4.6 of this SER. Therefore, the staff finds that the requirement of 10 CFR 73.67(g)(4), as described above, 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 three 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 three years after each change."

The applicant included a description of how it intends 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 DEF, 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 DEF 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 DEF application states that arrangements with a SNM-qualified licensed shipper would be made for the transport of SNM of low strategic significance, and that DEF will confirm that the licensed shipper has provisions in place to meet 10 CFR 73.67(c) requirements, as specified in SNMPPP Section 6.1. How the requirements of 10 CFR 73.67(c), (g)(2) and (g)(3) would be met by the applicant are described above in this SER. Therefore, the staff finds that the requirement of 10 CFR 73.67(g)(5), as described above, 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 intends to meet the in-transit physical protection requirements of 10 CFR 73.67(g) in Section 6 of its SNMPPP. Specifically, Sections 5.1.1.1 (for non-fuel SNM) and 5.1.2.1 (for fuel SNM) of the SNMPPP, state that the shipper would be notified upon receipt of SNM. In addition, the development of procedures for "[r]eceiving and shipping SNM" is described in Section 4.1 of the SNMPPP.

The staff finds that because DEF 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 "[r]eceiving 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. Section 13.5.2.2.8 of the FSAR

The applicant noted in the letter to the NRC dated September 18, 2014, that Section 13.5.2.2.8 of the FSAR would be modified to include the fact that the SNMPPP covers non-fuel SNM of low strategic significance (ADAMS Accession No. ML14267A029).

This inclusion presents, 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 Levy site. The staff verified that FSAR Section 13.5.2.2.8 as proposed by the applicant.

Therefore, the staff finds that the requirement to apply 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 Order 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 should be addressed. Section 1 of the SNMPPP discusses the analysis that justified only Parts C and D of the order needed to be addressed. The analysis provided by the applicant describes the details of the assessment as to whether or not the nuclear reactor would have a critical target area, as defined in the security order text. Therefore, the staff finds that 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.

The staff finds that, because the applicant committed to development of response implementing procedures that would be subject to NRC inspection, the order requirement of Part 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. of the order in Section 5.10, “Contingency Response,” of the SNMPPP.

The staff finds that, because the applicant described the response attributes that aligned with Part C.2. of the order, the order requirement of Part 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. The applicant stated that uranium hexafluoride would not be brought on the nuclear power reactor site and was not associated with the license application whatsoever.

The staff finds that, because the applicant described the conditions associated with uranium hexafluoride with the Levy site, the order requirement of Part D.1. 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,” of the SNMPPP. 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.

The applicant described an acceptable means to reduce storage of hazardous material on-site to the minimal necessary in order to avoid disrupting operations. Therefore, the staff finds that, because the applicant described a strategy to address Part D.2. of the order and committed to development of a procedure to implement that strategy, the order requirement of Part D.2. 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. The applicant committed to supplementing the Emergency Action Levels and their thresholds in response to a range of credible or imminent threats. The staff reviewed the applicant's description of the Emergency Action Level actions to be accomplished and found that the order measure was addressed in an acceptable manner.

The staff finds that, because the applicant described how the requirement of Part D.3. of the order would be addressed, the order requirement of Part D.3. 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. Specifically, the applicant committed to the evaluation of computer and communication networks for vulnerabilities, including modem access vulnerabilities, and to address them as necessary.

Therefore, the staff finds that, because the applicant described how the requirement of Part D.4. of the order would be addressed, the order requirement of Part D.4. 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. Specifically, the applicant coordinated with off-site fire departments and developed a response plan to notify those departments if and when necessary to facilitate fire suppression efforts.

Therefore, the staff finds that, because the applicant described how the requirement of Part D.5. of the order would be addressed, the order requirement of Part D.5. 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. Specifically, the applicant identified two local medical care facilities available for utilization given such a need was requested.

Therefore, the staff finds that, because the applicant described how the requirement of Part D.6. of the order would be addressed, the order requirement of Part D.6. would be met.

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

The applicant discussed Part D.7. of the order in Section 5.7, "Audits and Records," of the SNMPPP. Specifically, the applicant addressed establishing limited access to plant information that could possibly aid an adversary in planning and conducting an attack. The information would be protected as proprietary type information per 10 CFR 2.390, "Public inspections, exemptions, requests for withholding."

The staff finds that, because the applicant described how the requirement of Part D.7. of the order would be addressed, the order requirement of Part D.7. 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. Section 14.1 of the PSP, Revision 4, dated June 3, 2011 (SLES Accession No. NS108206), identifies the RG 5.66, "Access Authorization Program for Nuclear Power Plants," as 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 described escort methodologies developed for the Levy power reactors.

The applicant stated that RG 5.66 would be applied to meet Part 3 of the order and the staff recognizes that in doing so a more stringent access authorization process would be utilized than that described in Part 3 of the order. Therefore, the staff finds that the order requirements of Part 3, which include fingerprinting and other access authorization provisions, would be met.

1.5.5.2 Conclusion and Post Combined License Activities

Based on the above, the NRC staff finds that the information 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 fuel and non-fuel SNM (Fission Chambers), conforms to the applicable guidance in NUREG-1520 and NUREG-0800 and, therefore, is acceptable.

With respect to the applicable physical protection requirements specified in 10 CFR 73.67 and the post September 11, 2001, security order for the possession, use, and transport 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 applicable requirements and the guidance 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 DEF:

(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. The 10 CFR Part 37 rule affects any licensee that possesses an aggregated Category 1 or Category 2 quantity of radioactive material, any licensee that transports these materials using ground transportation, and any licensee that transports small quantities of irradiated reactor fuel. The compliance date for the 10 CFR Part 37 regulation was March 19, 2014.

By letter dated January 2, 2014 (ADAMS Accession No. ML14002A334), the NRC issued RAI 01.05-2 for the LNP Units 1 and 2 COL application. RAI 01.05-2 requested the applicant to provide descriptions in the FSAR, (e.g., Chapter 13), to address how the applicant, prior to taking possession of an aggregated Category 1 or Category 2 quantity of radioactive material, will implement the requirements of 10 CFR Part 37, by establishing, implementing, and maintaining a security program for LNP Units 1 and 2. By letter dated February 11, 2014 (ADAMS Accession No. ML14043A399), the applicant provided a response to RAI 01.05-2. Upon further review by the staff, it was determined that the regulations of 10 CFR Part 37 do not require COL applicants to address 10 CFR Part 37. After COL issuance, a COL licensee becomes subject to the requirements of this regulation upon taking possession of an aggregated Category 1 or Category 2 quantity of radioactive material. Therefore, the NRC withdrew RAI 01.05-2 as stated in a letter dated April 28, 2014 (ADAMS Accession No. ML14094A244). By letter dated May 27, 2014 (ADAMS Accession No. ML14149A263), the applicant withdrew its response to RAI 01.05-2. Since the initial RAI response proposed changes to be incorporated into a future revision of LNP Units 1 and 2 COL application, and the applicant subsequently rescinded the proposed changes, the staff verified that subsequent submittal of the updated COL application did not include the rescinded changes.

2.0 SITE CHARACTERISTICS

Chapter 2, "Site Characteristics," of the 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.

2.0.1 Introduction

The site characteristics are reviewed by the Nuclear Regulatory Commission (NRC) staff to determine whether the applicant has accurately described the site characteristics and site parameters 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 NRC staff to reach a conclusion on all safety matters related to siting of the Levy Nuclear Plant (LNP) Units 1 and 2. Because this combined license (COL) application references a 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

Section 2.0 of the LNP COL FSAR, Revision 9, incorporates by reference Chapter 2 of the AP1000 DCD, Revision 19. AP1000 DCD Chapter 2 includes Section 2.0 of the LNP COL FSAR.

In addition, in LNP COL FSAR Section 2.0, the applicant provided the following:

Supplemental Information

- LNP Supplemental (SUP) 2.0-1

The applicant provided supplemental information in LNP COL FSAR Section 2.0, "Site Characteristics," which describes the site characteristics of LNP. The applicant also provided LNP COL FSAR Table 2.0-201, which provides a comparison of the LNP site characteristics and the AP1000 DCD Site Parameters in AP1000 DCD Tier 1 Table 5.0-1 and DCD Tier 2 Table 2-1.

2.0.3 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" (September 2004), and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the site characteristics are given in Section 2.0 of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition).”

The applicable regulatory requirements for site characteristics are as follows:

- 10 CFR 52.79(a)(1)(i) - (vi) provides 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 from Section 2.0 of NUREG-0800 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 and Chapter 3 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.

2.0.4 Technical Evaluation

The NRC staff reviewed Section 2.0 of the LNP 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 site characteristics. The results of the NRC staff’s evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the LNP COL FSAR:

---

[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.

Supplemental Information

- LNP SUP 2.0-1

The NRC staff reviewed supplemental information LNP SUP 2.0-1 in LNP COL FSAR Section 2.0 describing the site characteristics of LNP Units 1 and 2. The AP1000 DCD site parameters in DCD Tier 2 Table 2-1 are compared to the site-specific site characteristics in LNP COL FSAR Table 2.0-201. In addition, control room atmospheric dispersion factors for accident dose analysis are presented in LNP COL FSAR Table 2.0-202.

The NRC staff reviewed and compared the site-specific characteristics included in LNP COL FSAR Table 2.0-201 against AP1000 DCD Tier 2 Table 2-1 and DCD Tier 1 Table 5.0-1. 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 Safety Evaluation Report (SER). 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 SER. The staff's evaluation of seismic and soil site characteristics is addressed in Section 2.5 of this SER. The staff's evaluation of site characteristics associated with missiles is addressed in Section 3.5 of this SER.

The site-specific characteristics listed in LNP COL FSAR Table 2.0-201 are enveloped by the AP1000 DCD site parameter values addressed in DCD Tier 2 Table 2-1 and DCD Tier 1 Table 5.0-1.

2.0.5 Post Combined License Activities

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

2.0.6 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 site characteristics, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

As set forth above, the NRC staff reviewed the application to ensure that sufficient information was presented in LNP SUP 2.0-1 to demonstrate that the characteristics of the site fall within the site parameters specified in the DC. The applicant has demonstrated that the requirements of 10 CFR 52.79(d)(1) have been met.

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*

Section 2.1 of the LNP COL FSAR, Revision 9, incorporates by reference Section 2.1 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Section 2.1, the applicant provided the following:

Tier 2 Departure

- STD DEP 1.1-1

The applicant proposed the following Tier 2 standard (STD) departure (DEP) from the AP1000 DCD. Part 7 of the LNP application identifies instances where the FSAR sections are renumbered to include content consistent with Regulatory Guide (RG) 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," as well as NUREG-0800 rather than following the AP1000 DCD numbering. In addition, LNP Part 7 requests an exemption from the requirement to use the same organization and numbering as the AP1000 DCD. In LNP COL FSAR Section 2.1, "Geography and Demography," Section 2.1.1 of the AP1000 DCD is renumbered as Section 2.1.4.

AP1000 COL Information Item

- LNP COL 2.1-1

The applicant provided additional information in LNP COL 2.1-1 to resolve COL Information Item 2.1-1 (COL Action Item 2.1.1-1), which addresses the provision of site-specific information related to site location and description, including political subdivisions, natural and man-made features, population, highways, railways, waterways, and other significant features of the area.

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 the Commission regulations for the site location and description are given in Section 2.1.1 of NUREG-0800.

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); (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. In particular, 10 CFR 100.20(a), and 10 CFR 100.21 require that population density and use characteristics of the site environs, including the exclusion area, low-population zone, and population center distance, be considered in determining the acceptability of a site for a stationary power reactor.

The related acceptance criteria from Section 2.1.1 of NUREG-0800 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 SER Sections 2.1.2 and 2.1.3, and Chapter 15) that the applicant has met the requirements of 10 CFR 50.34(a)(1) and 10 CFR Part 100.

The regulatory requirement associated with the Tier 2 departure request is as follows:

- 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.

2.1.1.4 *Technical Evaluation*

The NRC staff reviewed Section 2.1 of the LNP 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 site location and description. The results of the NRC staff’s evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the LNP COL FSAR:

Tier 2 Departure

- STD DEP 1.1-1

The applicant’s evaluation in accordance with Item B.5 of Section VIII of Appendix D to 10 CFR Part 52 determined that this departure did not require prior NRC approval. The NRC 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. A detailed evaluation of STD DEP 1.1-1 and the associated exemption can be found in Section 1.5.4 of this SER.

AP1000 COL Information Item

- LNP COL 2.1-1

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

The NRC staff, using publicly available maps, has independently verified the latitude and longitude supplied by the applicant. The NRC staff then converted this latitude and longitude to

Universal Transverse Mercator (UTM) coordinates for the proposed LNP Units 1 and 2 and used the calculated values to verify the UTM coordinates provided in the FSAR.

The NRC staff reviewed the site area map provided in the 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 NRC staff's review of the information in the LNP COL FSAR, and also the NRC staff's confirmatory review of the political subdivisions, and prominent natural and manmade features of the area as described in publically available documentation, the NRC 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 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 location and description, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

As set forth above, the applicant has presented and substantiated information to establish the site location and description. The staff has reviewed LNP COL 2.1-1, and for the reasons given in Section 2.1.1.4, 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 13.3 and Chapters 11 and 15 of this SER, 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.

The staff also concluded that STD DEP 1.1-1 meets the requirements for departures in 10 CFR Part 52, Appendix D, Section VIII, Item B5 and is, therefore, acceptable.

2.1.2 Exclusion Area Authority and Control

2.1.2.1 *Introduction*

The applicant's descriptions of exclusion area authority and control, which are used to verify the applicant's legal authority to determine and control activities within the designated exclusion area, are sufficient to enable the NRC staff to assess the acceptability of the reactor site. This review covers the following specific areas: (1) the applicant establishes its legal authority to determine all activities within the designated exclusion area, (2) the applicant establishes authority and control in excluding or removing personnel and property in the event of an emergency, (3) the applicant establishes 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) the applicant provides additional information requirements as prescribed within the "Contents of Application" sections of the applicable Subparts to 10 CFR Part 52.

2.1.2.2 *Summary of Application*

Section 2.1 of the LNP COL FSAR, Revision 9, incorporates by reference Section 2.1 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Section 2.1.2, the applicant provided the following:

AP1000 COL Information Item

- LNP COL 2.1-1

The applicant provided additional information in LNP COL 2.1-1 to resolve COL Information Item 2.1-1 (COL Action Item 2.1.2-1), which addresses the provision of site-specific information related to exclusion area authority and control, including size of the area, exclusion area authority and control, 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 the Commission regulations for the exclusion area authority and control are given in Section 2.1.2 of NUREG-0800.

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 these regulations relate to the inclusion in the 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 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 Section 2.1.2 of NUREG-0800 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, 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 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 NRC staff reviewed Section 2.1.2 of the LNP 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 exclusion area authority and control. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the LNP COL FSAR:

AP1000 COL Information Item

- LNP COL 2.1-1

The NRC staff reviewed LNP COL 2.1-1 related to the exclusion area authority and control, including size of the area, exclusion area authority and control, and activities that may be permitted within the designated exclusion area included in Section 2.1.2 of the LNP COL FSAR. COL Information Item in Section 2.1.1 of the AP1000 DCD 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, unauthorized commercial activities, or recreational activities within the Units 1 and 2 exclusion area. No public highways or active railroads not owned and controlled by the applicant traverse the exclusion area. There are no residents in the exclusion area. No unrestricted areas within the site boundary area are accessible to members of the public. The acceptance criteria for NUREG-0800, Section 2.1.2 state that, “Absolute ownership of all lands, including mineral rights, is considered to carry with it the required authority to determine all activities on this land and is acceptable.” The NRC staff verified that the applicant owns all of the land in the exclusion area and site boundary, including mineral rights.

The NRC staff also verified for consistency that the exclusion area boundary (EAB) is the same as being considered for the radiological consequences in Chapter 15 and Section 13.3 of the FSAR by the applicant. 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 control all activities within the designated exclusion area.

The NRC staff used publically available maps, satellite pictures, and the area map provided in the Unit 1 and 2 FSAR to verify that no publicly used transportation mode crosses the EAB; therefore, arrangements for the control of traffic in the event of an emergency are not required.

The NRC staff, using maps, satellite pictures and the area map provided in the Unit 1 and 2 FSAR verified that no public roads cross the exclusion area; therefore, neither relocation nor abandonment of roads is needed.

2.1.2.5 *Post Combined License Activities*

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

2.1.2.6 *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 the exclusion area authority and control, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff’s technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

As set forth above, the applicant has provided and substantiated information concerning its legal authority and control of all activities within the designated exclusion area. The staff has reviewed LNP 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. 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 methods by which the relocation or abandonment of public roads that lie within the proposed exclusion area can be accomplished, if necessary, 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 exclusion and removal of persons and property, and has established acceptable methods for control of the designated exclusion area.

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 appropriate protective measures could 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 Regulatory Position C.4 of RG 4.7, "General Site Suitability Criteria for Nuclear Power Stations"; 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*

Section 2.1 of the LNP COL FSAR, Revision 9, incorporates by reference Section 2.1 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Section 2.1.3, the applicant provided the following:

AP1000 COL Information Item

- LNP COL 2.1-1

The applicant provided additional information in LNP COL 2.1-1 to resolve COL Information Item 2.1-1 (COL Action Item 2.1.3-1), which addresses the provision of site-specific information related to population distribution for the site environs.

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 the Commission regulations for population distribution are given in Section 2.1.3 of NUREG-0800.

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), 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 SAR 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 Section 2.1.3 of RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," Revision 3, and in accordance with DG-1145; (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 Regulatory Position C.4 of RG 4.7, the applicant must give special attention to the consideration of alternative sites with lower population densities.

2.1.3.4 *Technical Evaluation*

The NRC staff reviewed Section 2.1.3 of the LNP 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 population distribution. The results of the NRC staff's evaluation of this information that is incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the LNP COL FSAR:

AP1000 COL Information Item

- LNP COL 2.1-1

The NRC staff reviewed LNP COL 2.1-1 related to the population distribution around the site environs included in Section 2.1.3 of the LNP COL FSAR. COL information item in Section 2.1.1 of the AP1000 DCD 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 environs, as presented in the applicant's FSAR, to determine whether the exclusion area, LPZ, and population center distance for the proposed LNP site comply with the requirements of 10 CFR Part 100. The staff also evaluated whether, consistent with Regulatory Position C.4 of RG 4.7 with regard to population density, 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 LPZ consist of two circles each with a radius of 3 miles and centered on each of the LNP reactor buildings. The staff verified the applicant's population data against U.S. Census Bureau data. Transient population estimates were based on evaluations of seasonal transient business, hotels, motels, recreation, schools, hospitals, nursing homes, correctional facilities, festivals, and migrant worker populations. The staff reviewed and verified the projected population data provided by the applicant, including the weighted transient population for 2005, 2010, 2015, 2020, 2030, 2040, 2050, 2060, 2070, and 2080. Based on this information, the staff finds that the applicant's estimate of the population and population projections are reasonable.

The nearest population center to the LNP site with more than 25,000 residents is the city of Ocala, Florida, with a 2000 population of 45,622. The closest point of Ocala's corporate limit to the LNP site was determined to be approximately 30.1 miles to the east-northeast of the site. This distance is over ten times the distance from the center of Units 1 and 2 to the closest LPZ boundary. This distance meets the requirement that the population center distance is at least one and one-third times the distance from the reactor to the outer boundary of the LPZ as stipulated in 10 CFR 100.21(b). The NRC staff's review of population centers closer than Ocala did not identify any population centers that were projected to reach a population of 25,000 prior to the projected end of plant life. The distance factors described in NUREG-0800, Section 2.1.1 and RG 4.7 Section C.4 are met. Therefore, the NRC staff concludes that the proposed site meets the population center distance requirement in accordance with 10 CFR 100.21.

Regulatory Position C.4 of RG 4.7, Revision 2 states that the population density, including the weighted transient population projected at the time of initial site approval and 5 years thereafter should not exceed 500 persons per square mile averaged over any radial distance out to 20 miles (cumulative population at a distance divided by the area at that distance).

The NRC staff evaluated the site population density provided by the applicant in FSAR Table 2.1.3-207 against the guidance in Regulatory Position C.4 of RG 4.7, Revision 2. Table 2.1.3-207 indicates that the population density for the years 2000 through the year 2020 is between 97 and 146 persons per square mile. Therefore, the population density would not exceed 500 persons per square mile averaged over a radial distance of up to 20 miles (cumulative population at a distance divided by the area at that distance). The NRC staff independently verified these estimates by reviewing U.S. Census Bureau data and concludes that the population density is consistent with the demographic factors of RG 4.7, Revision 2. Therefore, the NRC staff concludes that the LNP application is consistent with Regulatory Position C.4 of RG 4.7, 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 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 population distribution, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

As set forth above, the applicant has provided an acceptable description of current and projected population densities in and around the site. The staff has reviewed LNP COL 2.1-1, and for the reasons given above, concludes that the population data meets the requirements of 10 CFR 50.34(a)(1), 10 CFR 52.79(a)(1), and 10 CFR 100.20(a) and (b). The staff found that the applicant provided an acceptable description and safety assessment of the site, which includes present and projected population densities that are consistent with Regulatory Position C.4 of RG 4.7, and the applicant properly specified the LPZ and population center distance. In addition, the staff has reviewed and confirmed, by comparison with independently obtained U.S. Census Bureau population data, that the applicant's estimates of the present and projected populations surrounding the site, including transients, are accurate.

2.2 Nearby Industrial, Transportation, and Military Facilities

2.2.1 Locations and Routes

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 requirements prescribed in the "Contents of Application" sections of the applicable subparts to 10 CFR Part 52.

The NRC staff's review of LNP COL FSAR Section 2.2.1, "Locations and Routes," and Section 2.2.2, "Descriptions," is addressed in this SER section.

2.2.1.2 *Summary of Application*

Section 2.2 of the LNP COL FSAR, Revision 9, incorporates by reference Section 2.2 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Section 2.2, the applicant provided the following:

Tier 2 Departure

- STD DEP 1.1-1

The applicant proposed the following Tier 2 departure from the AP1000 DCD. Part 7 of the LNP application identifies instances where the FSAR sections are renumbered to include content consistent with RG 1.206, as well as NUREG-0800. In addition, LNP Part 7 requests an exemption from the requirement to use the same organization and numbering as the AP1000 DCD. In LNP COL FSAR Section 2.2, "Nearby Industrial, Transportation, and Military Facilities," Section 2.2.1 of the AP1000 DCD is renumbered as Section 2.2.4.

AP1000 COL Information Item

- LNP COL 2.2-1

The applicant provided additional information in LNP COL 2.2-1 to resolve COL Information Item 2.2-1 (COL Action 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.

2.2.1.3 *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 the Commission 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:

- 10 CFR 100.20(b), which requires 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.

- 10 CFR 52.79(a)(1)(iv), as it relates to the factors to be considered in the evaluation of sites, which require 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 reactor site criteria in 10 CFR Part 100.

- 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 Commission 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 Sections 2.2.1 and 2.2.2 of NUREG-0800 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 Section III of Sections 2.2.1-and 2.2.2 of NUREG-0800.

- 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.

The regulatory requirement associated with the Tier 2 departure request is as follows:

- 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.

2.2.1.4 *Technical Evaluation*

The NRC staff reviewed Section 2.2 of the LNP 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 nearby industrial, transportation, and military facilities. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the LNP COL FSAR:

Tier 2 Departure

- STD DEP 1.1-1

The applicant's evaluation in accordance with Item B.5 of Section VIII of Appendix D to 10 CFR Part 52 determined that this departure did not require prior NRC approval. The NRC 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. A detailed evaluation of STD DEP 1.1-1 and the associated exemption can be found in Section 1.5.4 of this SER.

AP1000 COL Information Item

- LNP COL 2.2-1

The NRC staff reviewed LNP 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 Section 2.2 of the LNP COL FSAR. COL information item in AP1000 DCD Section 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 NRC staff reviewed the LNP COL FSAR using the guidance described in Sections 2.2.1 and 2.2.2 of NUREG-0800.

This SER section identifies and provides the information that would help in evaluating potential effects on the safe operation of the nuclear facility by industrial, transportation, mining, and military installations in the LNP area. The evaluation of potential effects on the safe operation of the nuclear facility is described in SER Section 2.2.3.

Locations and Routes

The applicant identified and provided information regarding potential external hazard facilities and operations within a 5 mile radius of the LNP site.

The NRC staff confirmed that no major industrial activities are located within the 8-kilometer (km) (5-mile) radius of the LNP site (FSAR Figure 2.2.1-202)

The NRC staff verified that no active quarrying or mining facilities are located within the 8-km (5-mile) radius of the LNP site. Sixteen active mining or quarrying facilities are located within a 40-km (25-mile) radius of the LNP site (FSAR Figure 2.2.1-203).

Plum Creek Timberlands, L.P., is planning a mining operation, Titan Mines–Phase 2, within 8 km (5 mile) of the LNP site, approximately 1.6 km (1 mile) west of U.S. Highway 19 (FSAR Figure 2.2.1-203). The NRC staff verified with the Titan Mines Manager that all blasting will be done by a licensed contractor and that no explosives will be stored onsite. Only the explosives for one shoot will be brought to the site each day that a shoot is scheduled.

In addition to Orlando and Tampa, which are located beyond the 80-km (50-mile) radius, Gainesville and Ocala are two major transportation hubs for central Florida that are located within the region (FSAR Figure 2.2.1-201). Gainesville and Ocala are served by rail lines, as well as major interstates and highways that serve local and interstate traffic. These highways and interstates are described in LNP COL FSAR Section 2.2.2.5.

The NRC staff verified that no airports or private airstrips are located within the 8-km (5-mile) radius of the LNP site (FSAR Figure 2.2.1-204). J.R's private airstrip and the Crystal River Power Plant Heliport are located within a 16-km (10-mile) radius of the site.

Military Facilities

The NRC staff verified that no active military facilities are within 8 km (5 mile) of the LNP site. Florida National Guard, Company B, 3rd Battalion, 20th Special Forces Group and the 690th Military Police Company National Guard are the only significant military facilities located within an 80-km (50-mile) radius of the LNP site. Florida National Guard, Company B, 3rd Battalion, 20th Special Forces is located in Brooksville, Florida and is 67.6 km (42 mile) from the site. The 690th Military Police Company National Guard is located in Crystal River, Florida, adjacent to the Crystal River Airport, and is 24.5 km (15.2 mile) from the site.

Railroads

The NRC staff verified that no active railroads are located within the 8-km (5-mile) radius of the LNP site. Two railroad lines, an abandoned track, and an active line are located within 16 km (10 mile) of the LNP site.

Manufacturing and Storage of Hazardous Materials

The NRC staff verified that no manufacturing facilities that use or store hazardous products are located within the 8-km (5-mile) radius of the LNP site (FSAR Figure 2.2.2-201). A Tier 2 facility (the Town of Inglis water treatment plant [WTP]) is located approximately 4.8 km (3 mile) from the LNP site and stores/uses hazardous chemicals. Tier 2 facilities are those that store or manufacture hazardous materials. LNP COL FSAR Table 2.2.2-202 presents the chemicals and the quantities stored/used at the Town of Inglis WTP.

The NRC staff verified the following information. Florida Public Utilities is located on the east side of U.S. Highway 19, approximately 5.5 km (3.4 mile) south of the LNP site. This facility, located in the Town of Inglis, provides propane gas and has three tanks on site. One tank has a storage capacity of 113,563 liters (30,000 gallons) and each of the other two tanks can store 68,137 liters (18,000 gallons). No other volatile materials are located at this facility.

Pipelines

The NRC staff verified that underground natural gas pipelines are located within the 8-km (5-mile) radius of the LNP site on the north side of U.S. Highway 19 alongside the remaining rail bed from the abandoned railroad track. The pipelines run parallel to U.S. Highway 19, approximately 1769 meters (m) (5803 feet [ft.]) to the west-northwest of the LNP site. Florida Gas Transmission Company (FGT) plans to construct a 24.5-km (15.2-mile) loop, which would extend approximately 24 km (15 mi) along the eastern side of the existing pipeline. In a letter dated July 14, 2011, the applicant provided additional information related to a FGT expansion project, which placed a 36-inch pipeline into service on April 1, 2011.

The 20.3-centimeter (cm) (8-inch [in.]), 76.2-cm (30-in.), and 91.4-cm (36-in.) natural gas pipelines are owned by FGT. The 20.3-cm (8-in.) pipeline is buried to a minimum of 0.9 m (3 ft.) below ground surface (bgs), and is 2123 m (6966 ft.) west of the LNP site. The pipeline has a maximum pressure of 912 pounds per square inch (psi). The 76.2-cm (30-in.) pipeline is buried a minimum of 0.9 m (3 ft.) bgs. The pipeline has a maximum pressure of 1200 psi and is located 1769 m (5803 ft) west of the LNP site. The 91.4-cm (36-in.) pipeline is buried a minimum of 0.9 m (3 ft.) bgs. The pipeline has a maximum pressure of 1333 psi and is located 1757 m (5763 ft.) west-northwest of the LNP site. There are no plans to carry any other product in the pipeline except for natural gas. The locations of the 20.3-cm (8-in.), 76.2-cm (30-in.), and 91.4-cm (36-in.) pipelines with respect to the safety-related structures of the LNP are shown in LNP COL FSAR Figure 2.2.2-202.

Description of Waterways

The NRC staff verified that five waterways are located within the 8-km (5-mile) radius of the LNP site. The waterways include Ten Mile Creek, which connects to Cow Creek and the Gulf of Mexico, Spring Run Creek, which extends to the Gulf of Mexico, Lake Rousseau, the Cross Florida Barge Canal (CFBC), and Withlacoochee River. Lake Rousseau's main channel is 4.3 to 5.2 m (14 to 17 ft) deep, the CFBC is 3.7 m (12 ft) deep, and Withlacoochee River is 3 m (10 ft) deep.

Recreational boating within the 8-km (5-mile) radius is likely to be associated with Cow Creek, Lake Rousseau, the CFBC, and Withlacoochee River. The CFBC was renamed the Marjorie Harris Carr Cross Florida Greenway and is now used for recreational boating (see LNP COL FSAR Figure 2.1.3-204). The Inglis Mine utilizes the section of the barge canal to the west of U.S. Highway 19. The Inglis Mine has a slip on the northern side of the CFBC that is used for periodic shipments of limestone. The Inglis Mine is located outside of the 8-km (5-mile) radius of the LNP site (LNP COL FSAR Figure 2.2.1-203).

Description of Highways

The NRC staff verified that the major highway located near the LNP site leading to Gainesville and Ocala is U.S. Highway 19/98 (State Route [SR] 55). LNP COL FSAR Figure 2.2.1-201 illustrates the transportation routes in the region of the LNP site. Interstate 75 (I-75) is the closest interstate, which is located approximately 45 km (28 mile) to the east of the LNP site. At its nearest point, U.S. Highway 19/98 (SR 55) is located approximately 1974 m (6477 ft) from the center of the LNP site (FSAR Figure 2.2.2-201). The average annual daily traffic (AADT) counts at the four closest monitoring points within the 8-km (5-mile) radius of the LNP site range from 1600 (Site 340086–SR 121, 0.32 km [0.2 mile] northeast of SR 55) to 8600 (Site 340069-SR 55 at the southern city limits of Inglis) vehicles per day. This highway is mainly used for local traffic and local commodity deliveries.

Description of Railways

The NRC staff verified that two railroad lines are located within 16 km (10 mile) of the LNP site. The lines include an abandoned track with only the rail bed remaining, which is located northeast of the site and north of SR 336, and an active railroad line operated by CSX Transportation, Inc. (CSX), which is located southeast of the LNP site. The CSX line runs from the city of Crystal River northeast to the city of Dunnellon. The applicant stated that in accordance with NRC RG 1.206, further analysis of the CSX rail segment was not required since it is outside of the 8-km (5-mile) radius of the LNP site. RG 1.206, Section C.1.2.2, footnote 2, states that applicants should consider all facilities and activities within 5 miles (8.05 km) of the nuclear site. NUREG-0800, Section 2.2.1-2.2.2, item III.2, states that the staff's review should include all identified facilities and activities within 8 kilometers (5 miles) of the plant. The staff confirmed that no railroad passes within 5 miles of the LNP site. The staff finds that not performing additional analysis of the CSX rail segment is acceptable because it meets the criteria described in NUREG-0800 and the guidance of RG 1.206.

Description of Airports

The NRC staff verified that no airports are within the 8-km (5-mile) radius of the LNP site (LNP COL FSAR Figure 2.2.1-204). J.R.'s private airstrip is 10.1 km (6.3 mile) from the LNP site, and the Crystal River Power Plant Heliport is 14.5 km (9 mile) from the site. Nine public airports and 48 private airports or airstrips are located outside the 16-km (10-mile) radius, but within the 80-km (50-mile) radius of the LNP site, but these locations have limited facilities. No further analysis was performed by the applicant on the private airports or airstrips. The nine public airports and their respective distances to the LNP site are listed in LNP COL FSAR Section 2.2.2.7. LNP COL FSAR Table 2.2.2-203 provides a summary of operations data for these public airports. The table includes distance to the LNP site, daily operation traffic, runway information types of aircraft using the facility, aircraft based on the field, and flying patterns associated with each airport.

Approximately 50 aircraft are based at the Crystal River Airport (43 single-engine, 5 multi-engine, 1 helicopter, and 1 glider airplane), with approximately 100 aircraft operations per day (49 percent local general aviation [49 flights]; 49 percent transient general aviation [49 flights]; 1 percent air taxi aviation [1 flight]; and less than 1 percent military [1 flight]). Future plans for the airport include a 1524-m (5000-ft) extension of the east-west runway to be completed within the next 4 to 5 years. This improvement is designed to make aircraft landings safer and will not increase traffic. No aircraft accidents or collisions have occurred at Crystal River Airport that have resulted in fatalities or that have been considered serious accidents. Only minor landing mishaps that did not result in property damage have been reported by airport operations.

Approximately 52 aircraft are based at Marion County Dunnellon Airport (42 single-engine, 5 multi-engine, and 5 ultra lights), with approximately 41 aircraft operations per day (80 percent local general aviation [33 flights] and 20 percent transient general aviation [8 flights]). Future plans for the airport include rehabilitation of the two existing runways to accommodate slightly larger general aviation and corporate aircraft. An increase in traffic is not expected. Two accidents occurred in the past 3 years at Marion County Dunnellon Airport.

Approximately 36 aircraft are based at Williston Municipal Airport (27 single-engine, 3 multi-engine, 2 jet planes, 2 helicopters, and 2 ultra lights), with approximately 45 aircraft operations per day (30 percent local general aviation [14 flights] and 70 percent transient general aviation [31 flights]). Skydiving activities also originate from the Williston Municipal Airport. Williston Municipal Airport will be constructing new hanger storage and anticipates a 20 percent growth in operations. No aircraft accidents or collisions have occurred at Williston Municipal Airport that have resulted in fatalities or that have been considered serious accidents. Only minor landing mishaps that did not result in property damage have been reported by airport operations.

The closest large-scale public airport to the LNP site is the Ocala International Airport (LNP COL FSAR Figure 2.2.1-204). Ocala International Airport maintains 155 aircrafts used for general aviation with approximately 110,000 operations annually. No plans to expand the runways are projected for the near future at Ocala International Airport; however, within in the next 10 to 15 years, the airport plans to expand. Consistent with the guidance in NUREG-0800, Section 3.5.1.6 and RG 1.106, Section C.1.2.2.2.7, and due to Ocala's distance from the LNP site, Ocala International Airport operations would have to increase more than 500% before the applicant would have to provide an additional analysis regarding the probability of an airplane crash affecting safety related structures or systems at the LNP site.

George T. Lewis Airport, also known as the Cedar Key Airport, is located on an island 1.6 km (1 mile) west of Cedar Key and is owned by Levy County. The airport is public, does not have service staff, and has very light operations. George T. Lewis Airport has no aircraft types or operations data and has no plans to expand. The main function of this airport is to serve the resort and recreation activities at Cedar Key.

The Hernando County Airport maintains166 total aircraft with approximately 72,500 annual operations (125 single-engine, 16 twin-engine, 8 jets, 15 helicopters, and 2 ultra lights). Currently, the airport is extending one of the runways. No major accidents have been reported.

Approximately 135 aircrafts are based at the Gainesville Regional Airport, with 93,502 annual operations. Helicopters for the Gainesville Police and Alachua County Sheriff's Department are also housed at this airport, in addition to operating a flight school. Additional growth for the airport will be associated with the Eclipse 500.

LNP COL FSAR Table 2.2.2–203 describes the types of aircraft and flying patterns for aircraft-associated airports within the region. According to the Federal Aviation Administration (FAA), there are no temporary flight restrictions (TFR) within 32 km (20 mile) of the LNP site.

The applicant addressed and evaluated potential aircraft hazards following the approach and methodology 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. One of the factors the applicant used to assess the probability of aircraft accidents resulting in radiological consequences greater than the 10 CFR Part 100 exposure guidelines, was that there were no Federal airways within 2 miles of the LNP site.

In a letter dated March 6, 2009, the staff requested additional information (RAI) related to Federal airways within the 2 mile radius of the LNP site and requested that the applicant address the potential hazards. The applicant response to this RAI, dated April 6, 2009, noted a total of five Federal airways within the 2 mile limit of the LNP site.

The applicant submitted a supplemental response to this RAI, dated July 29, 2009. This supplement provided an analysis of the potential hazards from these airways and revised the LNP COL FSAR Sections 2.2.2.7 and 3.5.1.6. The applicant also replaced Table 2.2.1-204 and added new Table 3.5-201. The staff found the applicant's analysis showing the large and small aircraft crash probabilities, to be acceptable.

The staff reviewed this evaluation, the methodology and acceptance criteria and determined that the application is consistent with the acceptance criteria in NUREG–0800 Section 3.5.1.6.

The staff verified that the proposed markup changes in the applicant's RAI response are acceptable. This RAI is closed.

Projections of Industrial Growth

The staff verified that the LNP site is located in the southern part of Levy County immediately east of U.S. Highway 19/98 (SR 55). The site is primarily timber and currently undeveloped. The Goethe State Forest is located to the northeast, and the surrounding area is undeveloped agricultural land or sparsely populated rural residential land use. Some commercial automotive service, parts, storage, and gas stations are located within 8 km (5 mi) of the site. These facilities are primarily located along U.S. Highway 19 and County Route 40. Because Levy County is primarily rural; the majority of the industrial development within an 80-km (50-mile) radius of the LNP site is located in the urbanized areas of Marion and Citrus counties. Personal communication with the Levy County Planning Department indicates that no industrial growth is planned within an 8-km (5-mile) radius of the project site. Industrial development within a 16-km (10-mile) radius of the LNP site is primarily concentrated in Inglis along County Route 40 and U.S. Highway 19, and is limited to metal fabrication, automotive repair shops, and several mining operations. Mines within the 16-km (10-mile) radius of the LNP site include the Inglis Mine, located north of the CFBC; Holcim (US), Inc., located south of the CFBC; and Crystal River Quarry located in the community of Red Level. Gulf Rock Mine is located northwest of the LNP site and is inactive (LNP COL FSAR Figure 2.2.1-203).

The LNP site is located in the southern portion of Levy County. Citrus County is located to the south and Marion County is located to the east. LNP COL FSAR Table 2.2.2-204 lists the largest employers in Citrus, Levy, and Marion counties. The largest employers are within the utilities, education, and healthcare sectors.

2.2.1.5 *Post Combined License Activities*

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

2.2.1.6 *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 nearby industrial, transportation, and military facilities, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL 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 has reviewed LNP 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 LNP COL FSAR, as well as information that the staff independently obtained, the staff has 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 SER.

The staff also concluded that STD DEP 1.1-1 meets the requirements for departures in 10 CFR Part 52, Appendix D, Section VIII, Item B5 and is, therefore, acceptable.

2.2.2 Descriptions

The NRC staff's review of the LNP COL FSAR Section 2.2.2, "Descriptions," is addressed in SER Section 2.2.1.

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.

The scope of the review also includes the evaluation of man-made site hazards that have been identified as design-basis accidents with respect to safety-related structures, systems, and components (SSCs).

2.2.3.2 *Summary of Application*

Section 2.2 of the LNP COL FSAR, Revision 9, incorporates by reference Section 2.2 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Section 2.2, the applicant provided the following:

AP1000 COL Information Item

- LNP COL 2.2-1

The applicant provided additional information in LNP COL 2.2-1 to resolve COL Information Item 2.2-1 (COL Action 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 the Commission regulations for the evaluation of potential accidents are given in Section 2.2.3 of NUREG-0800.

The applicable regulatory requirements for evaluation of potential accidents are:

- 10 CFR 100.20(b), which requires that the nature and proximity of man-made 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.

- 10 CFR 52.79(a)(1)(iv), as it relates to the factors to be considered in the evaluation of sites, which require the location and description of industrial, military, or transportation facilities and routes, and the requirements of 10 CFR 52.79(a)(1)(vi) as they relate to compliance with 10 CFR Part 100.

The related acceptance criteria from Section 2.2.3 of NUREG-0800 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 NRC 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(December 2001).

2.2.3.4 *Technical Evaluation*

The NRC staff reviewed Section 2.2 of the LNP 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 evaluation of potential accidents. The results of the NRC staff’s evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the LNP COL FSAR:

AP1000 COL Information Item

- LNP COL 2.2-1

The NRC staff reviewed the LNP COL 2.2-1 related to information about industrial, military, and transportation facilities and routes used 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 Section 2.2.3 of the LNP COL FSAR. COL information item in Section 2.2 of the AP1000 DCD 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 due to 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 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 1 psi are considered to cause no significant damage.

The applicant determined a minimum safe standoff distance of 1658 ft. for truck transport using conservative assumptions and RG 1.91 methodology. By comparison, the distance to the closest highway is 6477 ft. from the nearest safety-related structure. The NRC staff performed independent calculations, which confirmed the applicant's results. Therefore, the NRC staff concludes the applicant's assumptions and methodology are acceptable, because they follow the guidance described in RG 1.91.

The applicant reported that, except for minor barge traffic on the CFBC, to and from the Inglis Mine (approximately 6 miles from the LNP), the local waterways are not navigable for commercial shipping and therefore, are not considered for hazard evaluations. The NRC staff finds this determination acceptable, recognizing that the CFBC may be used for the delivery of components during the construction of LNP.

In a letter dated July 14, 2011, the applicant provided additional information related to a FGT expansion project, which placed a 36-inch pipeline into service on April 1, 2011. The nearest and largest nearby natural gas pipeline runs parallel to U.S. Highway 19, approximately 5763 ft. to the west-northwest of LNP as shown on FSAR Figure 2.2.2-202. The 36-inch diameter pipeline is buried at a depth of 3 ft. with a maximum operating pressure of 1333 psi. Isolation of the line is obtained with isolation valves up to 19.4 miles apart.

In LNP COL FSAR Section 2.2.3.2.3, 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 FSAR Section 2.2.3.2.3, a delayed flammable cloud ignition is discounted on the basis of insufficient gas concentrations at the LNP site. However, resolving this issue for an onsite hazard does not preclude ignition at a location between the pipeline and the LNP site. Therefore, the overpressure hazard from either immediate or delayed ignition of the vapor cloud is not resolved. In a letter dated March 6, 2009, the NRC staff requested clarification of the applicant's statement that unconfined vapor explosions of a natural gas/air mixture are not credible.

In a letter dated April 6, 2009, the applicant provided a revision to LNP COL FSAR Section 2.2.3.2.3 to clarify the overpressure analysis, and clarified the basis for the statement that unconfined vapor explosions of a natural gas/air mixture are not credible. In the July 14, 2011, letter related to the 36-inch pipeline, the applicant affirmed the overpressure analysis and associated technical basis described in the April 6, 2009, letter and in FSAR section 2.2.3.2.3.

The NRC staff verified the analysis and determined the clarification requested is acceptable, because it follows the guidance described in RG 1.91. This RAI is closed.

Toxic Chemicals

The applicant stated, there is no rail or major barge traffic within 8 km (5 miles) of the LNP site. The road transportation corridors within 8 km (5 miles) of the LNP site include the following routes. U.S. Highway 19/98, located 1.9 km (1.2 miles) west of the LNP site, is mainly used for local traffic and local commodity deliveries only. Four county roads are shown on Figure 2.2.2-201: County Road 40, 4.5 km (2.8 miles) south; County Road 40A, 4.8 km (3.0 miles) southwest; SR 336, 6.8 km (4.2 miles) east-northeast; and County Road 337, 7.7 km (4.8 miles) northeast of the LNP site. None of these roadways are assumed to carry regular heavy truck traffic. Due to the lack of major industries in the area, significant commodity traffic on U.S. Highway 19/98 is expected to be minimal, with the preferred route for north-south commodity flow to be via I-75, which is 45.1 km (28 miles) east of the LNP site. Therefore, there are no adverse effects to LNP likely due to the transportation of toxic materials. The NRC staff, after independently reviewing available information on the internet from local, State and Federal agencies, concluded that the applicant's determination is adequate.

The applicant stated further that stationary hazardous chemical sources within 8 km (5 miles) of the LNP site are limited to the Inglis WTP located 4.8 km (3 mi) from the LNP site. As listed in Table 2.2.2-202, the quantities stored at the plant are small and are not significant sources of airborne contamination even in the event of an accidental failure of the storage containers. Therefore, there are no offsite sources of toxic chemicals within 8 km (5 miles) of the LNP site that could pose a threat to LNP.

In response to RAI 2.2.1-2.2.2-1 pertaining to the onsite storage of chemicals, the applicant stated that the chemicals stored on site are bounded by the standard chemicals identified in DCD Table 6.4-1. These chemicals were assessed by Westinghouse as part of the main control room habitability hazard analysis. The Westinghouse analysis found the chemicals listed in AP1000 DCD Table 6.4-1 not to present a hazard to the control room operators or to safety-related SSCs.

The applicant identified no site specific onsite toxic chemicals other than the standard onsite toxic chemicals identified in LNP COL FSAR Table 6.4-201. The NRC staff finds the chemicals listed in LNP COL FSAR Table 6.4-201 to be acceptable because they follow the guidance described in RG 1.78.

Fires

Fires originating from accidents at any facilities or transportation routes identified above do not have the potential to affect the safe operation of LNP because the distances between potential accident locations and LNP are greater than 1.6 km (1 mile). The closest potential source of a significant fire is the 91.4-cm (36-in.) natural gas line at 1757 m (5763 ft.) from the LNP site. An evaluation of the heat flux from a prolonged fire at the gas line results in a calculated heat flux less than the maximum solar heat flux on the surface of the earth (approximately 300 British thermal units per hour per square foot) at about 883.9 m (2900 ft) from the pipeline. In addition, the LNP main control room heating, ventilation, and air conditioning (HVAC) system 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 LNP operators. The NRC staff reviewed the above information and concluded that the applicant's determination is acceptable because it follows the guidance described in RG 1.78 and RG 1.106.

Collision with the Intake Structure

This section is not applicable, as the LNP intake structure is not located on a navigable waterway with commercial traffic.

Liquid Spills

There is no safety-related equipment located at the intake structure. The CFBC is now used for recreational boating. In addition, the Inglis Mine utilizes a section of the CFBC to the west of U.S. Highway 19 periodically for minor barge shipments of limestone. Neither category of water traffic is considered likely to possess or transport liquids that may be corrosive, cryogenic, or coagulant. Accidental release of minor quantities of oil could be associated with marine engine operation but would be effectively diluted by the water in the CFBC and Gulf of Mexico.

Therefore, in the unlikely event of an accidental spill of oil or liquids that may be corrosive, cryogenic, or coagulant in nature, the CFBC would provide ample dilution before any such liquids reach the CWS. Even if the operation of the CWS were adversely affected by an accidental spill, there would be no impact on the ability of the plant to safely shutdown since the passive core cooling system would not be affected by degradation of the CWS.

The NRC staff reviewed this information and finds it acceptable because the CWS of the AP1000 design has no safety related function.

2.2.3.5 *Post Combined License Activities*

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

2.2.3.6 *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 evaluation of potential accidents, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

As discussed above, the applicant has presented and substantiated information to identify potential hazards in the site vicinity. The staff has reviewed the information provided and concludes that the applicant has provided sufficient information with respect to the 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 LNP COL FSAR as well as information that the staff independently evaluated, the staff has concluded that potentially hazardous activities on site and in the vicinity of the LNP site have been identified. This addresses and resolves COL Information Item 2.2-1. In conclusion, the applicant has provided sufficient information to satisfy the requirements of 10 CFR Parts 50, 52, and 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 the NRC regulations, the NRC staff evaluates 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 reviews 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 has 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 Section 2.3 of the LNP COL FSAR (which references Revision 19 to 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*

Section 2.3.1, "Regional Climatology," of the LNP COL FSAR addresses averages and extremes of climatic conditions and regional meteorological phenomena that could affect the safe design and siting of the plant, including information describing the general climate of the region, seasonal and annual frequencies of severe weather phenomena, and other meteorological conditions to be used for design- and operating-basis considerations.

2.3.1.2 *Summary of Application*

Section 2.3 of the LNP COL FSAR, Revision 9, incorporates by reference Section 2.3 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Section 2.3, the applicant provided the following:

AP1000 COL Information Item

- LNP COL 2.3-1

The applicant provided additional information in LNP COL 2.3-1 to address COL Information Item 2.3-1 (COL Action Item 2.3.1-1). LNP COL 2.3-1 addresses site-specific information related to regional climatology.

In addition, this LNP 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 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 regional climatology are given in Section 2.3.1 of NUREG-0800.

The applicable regulatory requirements for identifying regional meteorology are:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 52.79(a)(1)(iii), as it relates to identifying the more 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 regional meteorological characteristics of the site.

The climatological and meteorological information assembled in compliance with the above regulatory requirements are necessary to determine a proposed facility's compliance with the following requirements in Appendix A of 10 CFR Part 50:

- GDC 2, Design Bases for Protection Against Natural Phenomena, which requires that structures, systems and components important to safety 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.

- GDC 4, Environmental and Dynamic Effects Design Bases, which requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, included loss-of-coolant accidents.

The related acceptance criteria from Section 2.3.1 of NUREG-0800 are as follows:

- 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).
- 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 tornado parameters should be consistent with Regulatory Guide (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 should be based on appropriate standards, with suitable corrections for local conditions.
- Consistent with RG 1.27, "Ultimate Heat Sink for Nuclear Power Plants," Revision 2, 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. (Not applicable to a passive containment system design that does not utilize a cooling tower or cooling pond).
- The weight of the 100-year return period snowpack should be based on data recorded at nearby representative climatic stations or obtained from appropriate standards with suitable corrections for local conditions. The weight of the 48-hour probably maximum winter precipitation (PMWP) should be determined in accordance with reports published by NOAA's Hydrometeorological Design Studies Center.
- Ambient temperature and humidity statistics 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 United States Environmental Protection Agency (EPA) studies.
- All other meteorological and air quality conditions identified by the applicant as design and operating bases should be documented and substantiated.

The information should be consistent with acceptable practices, data from NOAA, industry standards, and NRC regulatory guides.

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," was issued subsequent to the publication of NUREG-0800, Section 2.3.1. The ISG clarifies the staff's position that the applicant should identify 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 NRC staff reviewed Section 2.3.1 of the LNP COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL 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 regional climatology. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the LNP COL FSAR:

AP1000 COL Information Item

- LNP COL 2.3-1

The NRC staff reviewed LNP COL 2.3-1 related to the provision of regional climatology included in Section 2.3.1 of the LNP COL FSAR. The COL information item 2.3-1 in Section 2.3.6.1 of the AP1000 DCD, states:

> Combined License applicants referencing the AP1000 certified design will address site-specific information related to regional climatology.

Evaluation of the information provided in LNP COL 2.3-1 is discussed below.

2.3.1.4.1 General Climate

The applicant's description of the general climate of the proposed LNP site is based on references, which include the National Climatic Data Center (NCDC) Local Climatic Data (LCD) Annual Summaries for Gainesville, Jacksonville, Orlando, Tallahassee, and Tampa, Florida. Airflow, temperature and humidity, and precipitation patterns for these five locations were presented in LNP COL FSAR Table 2.3.1-202. The applicant identified the LNP site as being located in Florida's North Central state climate division as specified by the NCDC.

The NRC staff compared the applicant's general climate description to a similar NCDC narrative description of the climate of Florida (NCDC, Climates of the States #60)[2] and has confirmed its accuracy and completeness; thus, the staff accepts the applicant's description of the general climate.

---

[2] http://cdo.ncdc.noaa.gov/climatenormals/clim60/states/Clim_FL_01.pdf Accessed 11/17/2008

2.3.1.4.2 Regional Meteorological Conditions for Design and Operating Basis

2.3.1.4.2.1 Thunderstorms, Hail, and Lightning

The following discussion on thunderstorms, hail, and lightning is intended to provide a general understanding of the severe weather phenomena 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 thunderstorms have been observed on an average of 67.5 to 81.3 days per year. Thunderstorms have occurred most frequently during the months of June, July, and August. Consistent with NUREG-0800, Section 2.3.1, the applicant compiled this information from the 2006 LCDs for Gainesville, Jacksonville, Orlando, Tallahassee, and Tampa, Florida from the NCDC.

Using both 2006 and 2007 LCDs for Gainesville, Jacksonville, Orlando, Tallahassee, and Tampa, Florida from the NCDC, the staff confirmed that thunderstorms have been observed on an average of 67.5 to 81.3 days per year. The staff agrees with the applicant that thunderstorms have occurred most frequently in the months of June, July, and August at the five observation locations.

The applicant stated that 45 hail events were reported in Levy County from January 1, 1950 through November 30, 2008. Hail stone diameters greater than 0.75 inches were recorded. Consistent with the guidance provided in NUREG-0800, Section 2.3.1, the applicant compiled this information from the NCDC. The applicant noted that the number of reported hail events has increased significantly over time, primarily as a result of increased reporting efficiency and confirmation skill. This increase in hail reports is also likely due to the increased number of targets because of urbanization. This is because there are more targets damaged by hail in urban areas than in a rural area. Using the same database, the staff was able to confirm the applicant’s value of 45 hail events for Levy County, Florida during the same time frame.

The applicant stated that there are 12.52 flashes to earth per year per square kilometer on average, based on the data from Gainesville, Jacksonville, Orlando, Tallahassee, and Tampa. The staff independently evaluated this estimate based on LCDs from the same weather reporting stations from the NCDC and a method attributed to the Electric Power Research Institute (8.1 – 9.7 flashes to earth per square kilometer), a 10-year flash density map from Vaisala[3] (8 – 10 flashes to earth per square kilometer), and a 1999 paper by G. Huffines and R.E. Orville, titled “Lightning Ground Flash Density and Thunderstorm Duration in the Continental United States: 1989-96” (> 11 flashes to earth per square kilometer). Thus, the staff concludes that the applicant has provided a reasonable estimate of the frequency of lightning flashes.

Based on a mean frequency of 12.52 flashes to earth per year per square kilometer and an exclusion area for the proposed Units 1 and 2 of 5.64 square-kilometers, the applicant predicted

[3] http://www.lightningsafety.noaa.gov/stats/08_Vaisala_NLDN_Poster.pdf accessed 9/27/2010

that 70.6 lightning flashes per year can be expected within the exclusion area of the two proposed units. Using the methodology provided in Annex L of the National Fire Protection Association (NFPA), Standard for the Installation of Lightning Protection Systems, 2008 Edition, the staff has confirmed the applicant's calculation and finds it to be a reasonable estimate.

Consistent with the guidance provided in NUREG-0800, Section 2.3.1, the applicant has provided the necessary information regarding thunderstorms, hail, and lightning. As previously discussed, the staff has independently confirmed the descriptions provided by the applicant and accepts them as correct and adequate.

2.3.1.4.2.2 Tornadoes and Severe Winds

The applicant used a 57.25-year period of tornado reports (01/01/1950 through 03/31/2007) from the NCDC to determine the number of reported tornadoes in the vicinity of the proposed LNP COL site. During this period there have been 2911 total tornadoes (50.8 tornadoes per year) in Florida and 336 reported tornadoes in the 10 counties surrounding the proposed LNP site. The 10 surrounding counties include Levy, Dixie, Gilchrist, Alachua, Marion, Lake, Sumter, Citrus, Hernando, and Pasco. Using the same tornado database, the staff independently confirmed the tornado statistics, as presented in LNP COL FSAR Tables 2.3.1-203 through 2.3.1-205, as correct.

Following the methodology presented in WASH-1300, "Technical Basis for Interim Regional Tornado Criteria," issued May 1974, and the past tornado reports in the 10 counties surrounding the proposed LNP site, the applicant used the following formula to calculate the probability that a tornado will strike a particular location during any one year period:

$$P_s = \bar{n}\left(\frac{a}{A}\right)$$

where:

P_s = mean tornado strike probability per year

$\bar{n}$ = average number of tornadoes per year in the area being considered

a = average individual tornado area

A = total area being considered

The applicant calculated the probability of a tornado strike in the vicinity of the proposed LNP site of 4.39×10^{-4} per year, or, put differently, a recurrence interval of once every 2280 years. The staff verified the applicant's probabilistic calculation, using the same tornado database, "U.S. Storm Event Database, Tornadoes," from the NCDC. It should be noted that the applicant used a 1-degree square to determine the total area being considered for the tornado strike probability. This method does not follow the methodology presented in WASH-1300, which defines *A* as the “total area in which the tornado frequency has been determined.” However, the total area of the 10-counties used for the tornado analysis is roughly twice that of the 1-degree square box. The applicant’s method results in a higher, and consequently more conservative, estimation of the tornado strike probability due to the use of a smaller area in the denominator of the above equation. The staff also compared the tornado strike probability against the 2-degree box value in Appendix A to NUREG/CR-4461, “Tornado Climatology of the Contiguous United States,” Revision 2. The staff found that the applicant has presented a conservative estimate and accepts the tornado strike probability as presented.

The applicant chose tornado site characteristics based on Draft RG 1143 (DG-1143), “Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants.” This draft RG provides design-basis tornado characteristics for three tornado intensity regions throughout the United States, each with a 10^{-7} per year probability of occurrence. The proposed COL site is located in tornado intensity Region II; however, the applicant has chosen to include the maximum tornado wind speed intended for tornado intensity Region I. This is a conservative assumption and is, therefore, acceptable to the staff. The applicant proposed the following tornado site characteristics, which are listed in LNP COL FSAR Table 2.0-201:

Maximum Wind Speed	300 miles per hour

Because the applicant has correctly identified those design-basis tornado site characteristics presented in DG-1143, the staff concludes that the applicant has chosen acceptable tornado site characteristics. DG-1143 was the draft Revision 1 version to RG 1.76 and is acceptable to the staff because the design-basis tornado characteristics presented are more conservative than those presented in RG 1.76, Revision 1. This is because RG 1.76, Revision 1 relies on the Enhanced-Fujita (EF) scale to relate the degree of damage from a tornado to the tornado maximum wind speed. The EF scale effectively lowered the maximum wind speed associated with tornados, thus making RG 1.76, Revision 1 values less than the values in DG-1143. The applicant stated that the latest NRC position on design basis tornadoes is based on the information in NUREG/CR-4461 Revision 1. The staff notes that the current position of the NRC on design basis tornadoes is based on NUREG/CR-4461, Revision 2, which was published in February 2007. The applicant’s tornado wind speed site characteristic value bounds the value provided in NUREG/CR-4461, Revision 2 and RG 1.76, Revision 1, and is therefore acceptable to the staff.

Section 3.3.1 of the AP1000 DCD states that the operating basis wind speed site parameter value of 145 miles per hour (mph) (3-second gust) is based on an annual probability of occurrence of 0.02 (i.e., 50-year return period). Higher winds with an annual probability of occurrence of 0.01 (i.e., 100-year return period) were used in the design of seismic Category I structures by using an importance factor of 1.15. This is equivalent to designing the seismic Category I structures to a wind speed of 155 mph by using a 1.07 scaling factor from Table C6-7 of American Society of Civil Engineers/Structural Engineering Institute (ASCE/SEI) 7-05, “Minimum Design Loads for Buildings and Other Structures,” to convert a 50-year return period gust wind speed to a 100-year return period gust wind speed.

In an August 10, 2009, letter to the NRC, the applicant voluntarily submitted a supplemental response to RAI 2.3.1-8. In this letter, the applicant updated previous estimates of the LNP site characteristic basic wind speeds. The supplemental response to RAI 2.3.1-8 estimates that the LNP site characteristic basic wind speeds for the 50-year and 100-year return periods are 120 mph and 128 mph, respectively. The applicant followed the guidance provided in NUREG-0800, Section 2.3.1 by determining these site characteristic values using Figure 6.1 from ASCE/SEI 7-05. The staff independently verified that the applicant has followed an acceptable methodology and therefore accepts the applicant's values as correct. In RAI 2.3.1-17, the staff requested clarification on four points related to the tornado and severe wind speeds described in LNP COL FSAR Section 2.3.1.2.2. The applicant provided a clarification of each point in their RAI response and made a commitment to change and clarify the wording in the LNP COL FSAR. The staff reviewed the changes proposed in the RAI response and finds them to be acceptable. Therefore, the staff considers RAI 2.3.1-17 to be resolved. The commitment to update the FSAR with these clarifications is being tracked as **Confirmatory Item 2.3.1-1**.

<u>Resolution of Confirmatory Item 2.3.1-1</u>

Confirmatory Item 2.3.1-1 is an applicant commitment to update section 2.3.1 of its FSAR. The staff verified that LNP COL FSAR Section 2.3.1 was appropriately updated. As a result, Confirmatory Item 2.3.1-1 is now closed.

In RAI 2.3.1-18, the staff asked the applicant to explain the discrepancy between the 100-year return period site characteristic basic wind speed of 128 mph and the identification of a 100-year return basic wind speed of 139 mph. Using the Engineering Weather Data (EWD) compact disc, published by NOAA, the applicant updated LNP COL FSAR Section 2.3.1.2.2 to state that the maximum published 3-second gust wind speed for the region, based on severe wind events reported at the surrounding stations, is 130 mph. The applicant assumed this value represented a 50-year return period 3-second gust and converted it to a 100-year return period 3-second gust value of 139 mph using the 1.07 scaling factor from ASCE/SEI 7-05. The staff has found that the 50-year recurrence, 3-second gust basic wind speed reported on the EWD CD is based on data from ASCE 7-95, "Minimum Design Loads for Buildings and Other Structures." The 50-year recurrence basic wind speeds were updated three years later in ASCE 7-98, "Minimum Design Loads for Buildings and Other Structures," and were subsequently lowered to the basic wind speeds that are found in ASCE 7-05. The basic wind speeds presented in ASCE 7-05 were updated "based on a new and more complete analysis of hurricane wind speeds." A complete discussion on the reasons for this change can be found in Section C6.5.4, "Basic Wind Speed," of ASCE 7-05. The staff considers the 100-year return period site characteristic basic wind speed of 128 mph to be appropriate for the LNP site because it is based on the more recent analysis of hurricane winds presented in ASCE 7-05. The applicant provided clarifying language in their RAI response and made a commitment to change and clarify the wording in the FSAR. The staff reviewed the changes proposed and based on the above discussion, finds them to be acceptable. Therefore, the staff considers RAI 2.3.1-18 to be resolved. The commitment to update the FSAR with these clarifications is being tracked as **Confirmatory Item 2.3.1-2**.

Resolution of Confirmatory Item 2.3.1-2

Confirmatory Item 2.3.1-2 is an applicant commitment to update section 2.3.1 of its FSAR. The staff verified that LNP COL FSAR Section 2.3.1 was appropriately updated. As a result, Confirmatory Item 2.3.1-2 is now closed.

In RAI 2.3.1-20, the staff asked the applicant to describe how the Levy County COLA satisfies the Combined License Information requirement of AP1000 DCD Section 3.5.4 in consideration of RG 1.221, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants." The applicant responded by committing to update LNP COL FSAR Subsection 3.3.2.1 and by adding new Subsection 3.5.2 and Table 3.5-202. These modifications to the FSAR, using the figures and tables in RG 1.221, include the hurricane generated missile velocities based on a maximum hurricane wind speed of 195 mph at the LNP site. The staff reviewed the changes proposed and finds them to be acceptable. Therefore, the staff considers RAI 2.3.1-20 to be resolved. The staff's evaluation of the wind loading and structural engineering aspects of RAI 2.3.1-20 is in Section 3.3.2.4 of this SER.

2.3.1.4.2.3 Heavy Snow and Severe Glaze Storms

The applicant stated that trace amounts of snowfall do occur in Florida, but measurable snowfalls are typically less than a quarter of an inch and are extremely rare. The record snowfall for the region was at Jacksonville, Florida, which received 1.5 inches of snow in February of 1958. The NRC staff issued DC/COL-ISG-007, which clarifies the NRC 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. The ISG revises the previously issued NRC 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 (in inches of water) for a 48-hour period that is physically possible over a 25.9-square-kilometer (km) (10-square-mile (mi)) area at a particular geographical location during those months with the historically highest snowpack.

The staff evaluated the normal winter precipitation event and the extreme frozen and liquid winter precipitation events in accordance with the ISG. Due to the location of the proposed units along the Gulf Coast, large snow and ice events are rare. The normal and extreme winter precipitation loads for the LNP COL were determined to be significantly less than the AP1000 DCD site parameter value of 75 lb/ft^2. The staff agrees with the applicant that the normal and extreme winter precipitation roof loads are not significant; therefore, the staff accepts the applicant's discussion as correct and adequate.

2.3.1.4.2.4 Hurricanes

The applicant discussed a history of hurricanes impacting both the Atlantic and Gulf of Mexico coastlines of Florida between 1899 and 2007. The applicant stated that Florida has been impacted by 150 hurricanes and tropical storms. Of the 150 storms, 85 were tropical storms, 19 were Category 1, 19 were Category 2, 19 were Category 3, 6 were Category 4, and 2 were Category 5 hurricanes. The applicant also stated that according to the NOAA Coastal Services Center (CSC), 45 hurricanes rated Category 1-5 have passed within 100 nautical mi of the LNP site. The applicant stated that the annual frequency of hurricanes is estimated to be 0.13 and 0.29 storms per year within 50- and 100-nautical mi of the LNP site, respectively.

The staff evaluated data from the NOAA CSC for hurricanes making landfall in or passing near Levy County, Florida between 1851 and 2008. The staff found that during this time period there were a total of 28 Category 1, 15 Category 2, 9 Category 3, and 2 Category 4 storms that passed within 100 nautical mi of Levy County. The staff recognizes that there are differences in the number of storms reported in the area between the staff and the applicant. However, the staff finds these differences to be small and does not consider them to have an impact on the safety analysis. Therefore, the staff accepts the applicant's descriptions of the number of hurricanes in the vicinity of Levy County, Florida.

The staff agrees with the applicant that the largest threat to the LNP site from hurricanes will be high winds, heavy precipitation, and potential flooding due to storm surges.

2.3.1.4.2.5 Normal Operating Heat Sink Design Parameters

Many plants use a cooling tower as an 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 the AP1000 DCD Tier 1 Table 5.0-1 and AP1000 DCD Tier 2 Table 2-1. Therefore, the applicant need not identify meteorological characteristics for evaluating the design of an UHS cooling tower. The applicant states in LNP COL FSAR Section 2.3.1.2.5 that the AP1000 reactor does not rely on site service water as a safety grade UHS.

A summary of statistically significant dry- and wet-bulb temperatures that were used by the applicant to determine the LNP site characteristic temperatures, as obtained from Jacksonville, Tallahassee, and Tampa, Florida, were provided in LNP COL FSAR Tables 2.3.1-207 and 2.3.1-210. These temperatures were based on the 30-year (1961-1990) Solar and Meteorological Surface Observation Network (SAMSON) database and the 23-year (1973-1996) EWD database from NOAA. The staff has performed an independent confirmatory analysis of the data provided in LNP COL FSAR Tables 2.3.1-207 and 2.3.1-210 and accepts them as correct and adequate.

The staff evaluated the applicant's design-basis temperatures primarily based on Tampa and Tallahassee, Florida hourly temperature data from 1948 to 2008 and 1943 to 2008, respectively. The Tampa, Florida observation station is located 78 mi to the south of the LNP site. This site is considered appropriate for comparison due to its close proximity to the Gulf of Mexico. The staff also compared the LNP site to the Tallahassee, Florida reporting station, which is located 138 mi northwest (NW) of the LNP site. This station was included because of its close proximity to the Gulf of Mexico. Using additional stations, as the applicant has done, such as Jacksonville, Florida, is conservative because it could only potentially result in more extreme temperatures.

Because the stations are located at approximately the same elevation as the LNP site, the staff expects that the temperature and humidity data recorded at Tampa and Tallahassee should be similar to the LNP site conditions. In order to confirm this hypothesis, the staff generated 2007 and 2008 dry-bulb statistics from the NCDC online database and compared them with similar statistics generated from the applicant's 2007 and 2008 onsite meteorological database. The results of this comparison appear below in Table 2.3.1-1:

Table 2.3.1-1. Dry-Bulb Statistics for Tampa, Tallahassee, and LNP

DRY-BULB STATISTIC	2007			2008		
	Tampa	Tallahassee	LNP	Tampa	Tallahassee	LNP
Maximum	36.0 °C	38.0 °C	34.6 °C	36.0 °C	36.0 °C	33.9 °C
1 percent Exceedance	33.0 °C	35.0 °C	32.8 °C	33.0 °C	34.0 °C	31.2 °C
Median	23.3 °C	22.0 °C	22.1 °C	24.0 °C	21.0 °C	21.4 °C
99 percent Exceedance	4.4 °C	-2.0 °C	2.0 °C	7.0 °C	-1.1 °C	-0.2 °C
Minimum	-2.0 °C	-7.2 °C	-3.9 °C	1.0 °C	-7.0 °C	-5.9 °C

The staff also compiled and compared the Tampa and Tallahassee dew point statistics with the onsite dew point data provided by the applicant (Table 2.3.1-2).

Table 2.3.1-2. Dew Point Statistics for Tampa, Tallahassee, and LNP

DEW POINT STATISTIC	2007			2008		
	Tampa	Tallahassee	LNP	Tampa	Tallahassee	LNP
Maximum	26.0 °C	27.0 °C	25.7 °C	26.0 °C	26.0 °C	25.3 °C
1 percent Exceedance	24.4 °C	24.0 °C	24.5 °C	24.0 °C	24.0 °C	24.2 °C
Median	17.2 °C	14.0 °C	17.1 °C	17.8 °C	16.0 °C	17.0 °C

This comparison shows that the Tampa and Tallahassee dry-bulb and dew point (humidity) data are generally representative of the LNP data.

Details regarding the description, design basis, and operation of the AP1000 PCS are provided in Tier 2 Section 6.2.2 of the AP1000 DCD. AP1000 DCD Section 6.2.2.1 states that the PCS is designed to withstand the effects of natural phenomena such as ambient temperature extremes. AP1000 DCD, Tier 2, Section 6.2.2.3, further states that the containment pressure analyses are based on an ambient temperature of 115 ° Fahrenheit (F) dry-bulb and 86.1 °F coincident wet-bulb. These are the maximum safety air temperature site parameter values listed in AP1000 DCD Tier 1 Table 5.0-1 and AP1000 DCD Tier 2 Table 2-1. As discussed in Section 2.3.1.4.2.7 of this SER, the applicant's site characteristic temperatures presented in LNP COL FSAR Table 2.0-201 are bounded by the AP1000 DCD site parameters.

2.3.1.4.2.6 Inversions and High Air Pollution Potential

The following discussion on inversions and high air pollution potential is intended to provide a general understanding of the phenomena in the site region but does not result in the generation of site characteristics for use as design or operating basis. NUREG-0800 states that the site's air quality should be described in detail, including identification of the site's AQCR and its attainment designation with respect to state and national ambient air quality standards.

The applicant stated that the LNP site is located in the North Central state climate division of the NCDC. The staff has confirmed that the EPA has designated that Levy County, Florida is in attainment for all criteria pollutants.

The applicant used mixing height data from Tampa, Florida to characterize the potential for inversions at the LNP site. Although Tampa, Florida is 78 mi to the south of the site, it is the closest available station with this type of data. LNP COL FSAR Table 2.3.1-208 listed the expected seasonal frequencies of inversions below 152 meters (m) (500 feet (ft.)) and LNP COL FSAR Table 2.3.1-209 listed the mean monthly mixing depths. The inversion frequency in Tampa, Florida averaged 28 percent in the summer season and 37 percent in the winter season. The lowest mean monthly mixing height occurs in January (730 m) and the greatest mean mixing depth occurs in May (1410 m). Using references[4,5] consistent with NUREG-0800, Section 2.3.1, the staff has verified that the information provided by the applicant is correct and adequate.

2.3.1.4.2.7 Ambient Air Temperatures

Along with the normal operating heat sink design temperatures presented in LNP COL FSAR Section 2.3.1.2.5 and reviewed in Section 2.3.1.4.2.5 of this SER, the applicant provided additional dry- and wet-bulb temperatures in LNP COL FSAR Section 2.3.1.2.7, which are summarized in LNP COL FSAR Table 2.3.1-10. The applicant based these additional ambient air temperature statistics on the SAMSON database, as previously discussed in SER Section 2.3.1.4.2.5, and NOAA EWD. Both of these sources are consistent with NUREG-0800, Section 2.3.1, and are acceptable to the staff. The staff relied primarily on Tampa, Florida hourly data during the period of 1938 through 2008 to review the applicant's temperatures. The results of this independent review are consistent with those presented by the applicant. Thus, the staff accepts the applicant's additional ambient temperatures as correct and adequate.

Comparison with AP1000 Site Parameters for Ambient Air Temperature

AP1000 DCD site parameters for ambient air temperature are defined as follows:

- <u>Maximum Safety Dry Bulb Temperature and Coincident Wet-Bulb Temperature</u>: These site parameter values represent a maximum dry-bulb temperature that exists for 2 hours

---

[4] Holzworth, George C., "Mixing Heights, Wind Speeds, and Potential for Urban Air Pollution Throughout the Contiguous United States," AP-101, Office of Air Programs, EPA, January 1972.

[5] 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.

or more, combined with the maximum wet-bulb temperature that exists in that population of dry-bulb temperatures.

- Minimum Safety Dry Bulb Temperature: This site parameter value represents a minimum dry-bulb temperature that exists within a set of hourly data for duration of 2 hours or more.

- Maximum Safety Noncoincident Wet-Bulb Temperature: This site parameter value represents a maximum wet-bulb temperature that exists within a set of hourly data for duration of 2 hours or more.

- Maximum Normal Dry-Bulb Temperature and Coincident Wet-Bulb Temperature: The maximum normal value is the 1-percent seasonal exceedance temperature. The maximum temperature is for the months of June through September in the northern hemisphere. The 1-percent seasonal exceedance is approximately equivalent to the annual 0.4-percent exceedance.

- Minimum Normal Dry-Bulb Temperature: The minimum normal value is the 99-percent seasonal exceedance temperature. The minimum temperature is for the months of December, January, and February in the northern hemisphere. The 99-percent seasonal exceedance is approximately equivalent to the annual 99.6-percent exceedance.

- Maximum Normal Noncoincident Wet-Bulb Temperature: The maximum normal value is the 1-percent seasonal exceedance temperature. The maximum temperature is for the months of June through September in the northern hemisphere. The 1-percent seasonal exceedance is approximately equivalent to the annual 0.4-percent exceedance.

The applicant's safety temperature site characteristic values are based on conservative 100-year estimates. The ambient air temperatures used for comparison against the AP1000 site parameters are presented in LNP COL FSAR Table 2.0-201.

Using a combination of NCDC hourly data from Jacksonville (1931-2008), Tallahassee (1943-2008), and Tampa (1938-2008), Florida, and climate data from the American Society of Heating, Refrigerating and Air-Conditioning Engineers (ASHRAE), the staff was able to verify that the applicant's site characteristic temperatures presented in LNP COL FSAR Table 2.0-201 are adequate and bounded by the AP1000 DCD site parameters.

In RAI 2.3.1-19, the staff requested that the applicant update the LNP COL FSAR to change the normal ambient design site characteristic temperatures to reflect the 0.4-percent annual exceedance temperatures, which are approximately equivalent to the 1-percent seasonal exceedance temperatures. In response to RAI 2.3.1-19, the applicant has committed to updating LNP COL FSAR Section 2.3.1.2.7.3, Table 2.0-201 and Table 2.3.1-210 to include the normal ambient site characteristic temperatures that correspond with the definition of the AP1000 DCD site parameter temperatures. The revised normal ambient design site characteristic temperatures remain bounded by the AP1000 DCD site parameters, thus the staff finds the applicant's response to RAI 2.3.1-19 to be acceptable. This commitment to update the FSAR is being tracked as **Confirmatory Item 2.3.1-3**.

Resolution of Confirmatory Item 2.3.1-3

Confirmatory Item 2.3.1-3 is an applicant commitment to update Section 2.3.1 of its FSAR. The staff verified that LNP COL FSAR Section 2.3.1 was appropriately updated. As a result, Confirmatory Item 2.3.1-3 is now closed.

2.3.1.4.3 Effects of Global Climate Change on Regional Climatology

The applicant presented a discussion on the potential effects of global climate change on the regional climatology of the site. The applicant stated that even the most reliable climate models are not capable of accurately predicting design-basis extremes in weather patterns.

NUREG-0800, Section 2.3.1, states that historical data used to characterize a site should extend over a significant time interval to capture cyclical extremes. During the course of the technical review, the staff made an effort to obtain the longest period of data available to determine the adequacy of the applicant's proposed site characteristics. For example, snow load was based on a 100-year return period, ambient design temperatures were based on a minimum of 65 years of hourly data and an estimated 100-year return period value. Tornado statistics were based on a 57.25 year period and tornado wind speeds were based on a 10^{-7} per year return interval as stated in DG-1143. Extreme winds were based on a 100-year return period, including 157 years of historical hurricane data (1851-2008).

The U.S. Global Change Research Program (USGCRP) released a report to the President and Members of Congress in June 2009 entitled "Global Climate Change Impacts in the United States." This report, produced by an advisory committee chartered under the Federal Advisory Committee Act, summarizes the science of climate change and the impacts of climate change on the United States.

The USGCRP report found that the average annual temperature of the Southeast (which includes the Florida coastline where the LNP site is located) did not change significantly over the past century as a whole, but the annual average temperature has risen about 1.6 °F since 1970 with the greatest seasonal increase in temperature occurring during the winter months. Climate models predict continued warming in all seasons across the Southeast and an increase in the rate of warming through the end of the 21st century. Average temperatures in the Southeast are projected to rise by 2 – 5 °F by the end of the 2050s, depending on assumptions regarding global greenhouse gas emissions.

The USGCRP report also states that there is a 10- to 15-percent decrease in observed annual average precipitation from 1958 to 2008 in the region where the LNP site is located. Future changes in total precipitation are more difficult to project than changes in temperature. Model projections of future precipitation generally indicated that southern areas of the United States will become drier. Except for indications that the amount of rainfall from individual hurricanes will increase, climatic models provide divergent results for future precipitation for most of the Southeast.

The USGCRP reports that the power and frequency of Atlantic hurricanes has increased substantially in recent decades, but the number of North American mainland land-falling hurricanes does not appear to have increased over the past century. The USGCRP reports that likely future changes for the United States and surrounding coastal waters include more intense hurricanes with related increases in wind and rain, but not necessarily an increase in the number of these storms that make landfall.

The USGCRP further states that there is no clear trend in the frequency or strength of tornadoes since the 1950s for the United States as a whole. The applicant stated that the number of recorded tornado events has generally increased since detailed records were routinely kept beginning around 1950. However, some of this increase is attributable to a growing population, greater public awareness and interest, and technological advances in detection. The USGCRP reaches the same conclusion.

The USGCRP reports that the distribution by intensity for the strongest 10 percent of hail and wind reports is little changed, providing no evidence of an observed increase in the severity of such events. Climate models project future increases in the frequency of environmental conditions favorable to severe thunderstorms. But the inability to adequately model the small-scale conditions involved in thunderstorm development remains a limiting factor in projecting the future character of severe thunderstorms and other small-scale weather phenomena.

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 related to this section.

2.3.1.6 *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 regional climatology, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL 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 has 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 (COL Action Item 2.3.1-1).

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. Because the applicant has correctly implemented these methodologies, as described above, the staff has determined 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*

Section 2.3.2, “Local Meteorology,” of the LNP COL FSAR addresses the local (site) meteorological parameters, the assessment of the potential influence of the proposed plant and its facilities on local meteorological conditions and the impact of these modifications on plant design and operation, and a topographical description of the site and its environs.

2.3.2.2 *Summary of Application*

Section 2.3 of the LNP COL FSAR, Revision 9, incorporates by reference Section 2.3 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Section 2.3, the applicant provided the following:

AP1000 COL Information Item

- LNP COL 2.3-2

The applicant provided additional information in LNP COL 2.3-2 to address COL Information Item 2.3-2 (COL Action Item 2.3.2-1). LNP COL 2.3-2 addresses the provision of local meteorology.

2.3.2.3 *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 local meteorology are given in Section 2.3.2 of NUREG-0800.

The applicable regulatory requirements for identifying local meteorology are:

- 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 characteristics of the site.

The related acceptance criteria from Section 2.3.2 of NUREG-0800 are as follows:

- Local summaries of meteorological data based on onsite measurements 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 (LWR Edition)," Section 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, Section 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 resulting from plant construction and operation. The acceptability of the information is determined through comparison with standard assessments.

- The description of local site airflow should include wind roses and annual joint frequency distributions of wind speed and wind direction by atmospheric stability for all measurement levels using the criteria provided in RG 1.23, Revision 1.

2.3.2.4 *Technical Evaluation*

The NRC staff reviewed Section 2.3.2 of the LNP 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 local meteorology. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information contained in the LNP COL FSAR.

AP1000 COL Information Item

- LNP COL 2.3-2

The applicant provided information in LNP COL 2.3-2 to resolve COL Information Item 2.3-2, which addresses the provision of local meteorology.

The NRC staff reviewed LNP COL 2.3-2, related to the provision of local meteorology included under Section 2.3 of the LNP COL FSAR. COL Information Item 2.3-2 in Section 2.3.6.2 of the AP1000 DCD states:

> Combined License applicants referencing the AP1000 certified design will address site-specific local meteorology information.

2.3.2.4.1 Normal and Extreme Values of Meteorological Parameters

2.3.2.4.1.1 Wind Summaries

The applicant produced monthly and annual wind summaries from the onsite meteorological data from February 1, 2007 through January 31, 2009. LNP COL FSAR Tables 2.3.2-201 through 2.3.2-241 presented the average joint frequency distribution of wind speed and direction by Pasquill Stability Category (i.e., stability class) for both the lower-level (10-m) and upper-level (60-m) measurement heights. The 2-year joint frequency distribution, based on the lower-level measurement height, was used as input to the atmospheric dispersion models discussed in LNP COL FSAR Sections 2.3.4 and 2.3.5. Using the hourly meteorological data provided by the applicant, the staff independently produced the 2-year joint frequency distributions at both the lower-level and upper-level measurement heights and has confirmed the applicant's wind summaries as correct and acceptable.

Graphical illustrations of the wind summaries (i.e., wind roses) from the 1-year period February 1, 2007, through January 31, 2008, were also produced by the applicant in LNP COL FSAR Figures 2.3.2-201 through 2.3.2-213. These figures show the average monthly wind speed and direction for 16 radial compass directions over all stability classes during the 1-year period of record. Although the wind roses only include data for 1 year, the staff has confirmed that the wind speed and wind direction frequencies for the two year period from February 1, 2007, through January 31, 2009, are very similar. Using the hourly meteorological data provided by the applicant, the staff independently produced the same wind roses and has confirmed the applicant's figures as correct and acceptable.

The applicant compared the onsite wind summaries against wind speed and direction from the Tallahassee, Gainesville, and Tampa, Florida stations. The 1-year onsite wind rose provided in LNP COL FSAR Figure 2.3.2-201 shows a higher frequency of east-west winds. This pattern is also depicted in the wind roses from Gainesville and Tampa, Florida. The applicant suggests that this is most likely due to the diurnal influence of the sea breeze effects. The Tallahassee, Florida wind roses show that north-south wind patterns are most frequent. This is also most likely due to the diurnal sea breeze effects generated from the east-west directed coastline along the panhandle of Florida.

LNP COL FSAR Table 2.3.2-208 shows that the total number hours identified as calm winds at the 10-meter wind level is 3223, which is 18.8 percent of the total observations reported during the period of February 1, 2007 through January 31, 2009. In RAI 2.3.2-1, the staff asked the applicant to explain this high frequency of calm winds. In response to RAI 2.3.2-1, the applicant explained that the conditions reported as calm were for wind speeds less than the manufacturer's stated sensitivity threshold for the instrument (0.4 meters per second (m/s)). The number of calm winds at the 60-meter level drops to 174 hours, or 1.04 percent, which is considerably less than at the 10-meter level. The applicant states that the calm wind speeds at the lower level can be attributed to the height of the surrounding forest canopy, and its corresponding influence on wind speeds at the 10-meter level. The staff accepts this explanation as reasonable. Therefore, RAI 2.3.2-1 is resolved. Through the use of the 2009 ASHRAE database, the staff determined that the percentage of 10-meter level calm winds at the LNP site was comparable to the percentage of calms recorded at the five surrounding NWS recording stations.

2.3.2.4.1.2 Ambient Temperature

The applicant provided, in LNP COL FSAR Table 2.3.2-241, an ambient temperature summary based on the onsite meteorological data collected from February 1, 2007, through January 31, 2008, and five surrounding weather reporting stations (Tampa, Gainesville, Orlando, Tallahassee, and Jacksonville, Florida). Although LNP COL FSAR Table 2.3.2-241 only includes data for 1 year, the staff has confirmed that the temperature data for the two year period from February 1, 2007, through January 31, 2009, are consistent.

Using the applicant's onsite meteorological monitoring program data from the 2-year period from February 1, 2007 through January 31 2009, and independently obtained hourly data from the five surrounding NWS observation stations, the staff has confirmed the values presented in LNP COL FSAR Table 2.3.2-241 as correct and acceptable.

2.3.2.4.1.3 Dew-Point Temperature

The applicant provided, in LNP COL FSAR Table 2.3.2-242, a dew-point temperature summary based on the onsite meteorological data collected from February 1, 2007, through January 31, 2008, and five surrounding weather reporting stations (Tampa, Gainesville, Orlando, Tallahassee, and Jacksonville, Florida). Although LNP COL FSAR Table 2.3.2-242 only includes data for 1 year, the staff has confirmed that the dew-point temperature data for the two year period from February 1, 2007, through January 31, 2009, are consistent.

Using the applicant's onsite meteorological monitoring program data from the 2-year period from February 1, 2007 through January 31 2009, and independently obtained hourly data from the five surrounding NWS observation stations, the staff has confirmed the values presented in LNP COL FSAR Table 2.3.2-242 as correct and acceptable.

2.3.2.4.1.4 Atmospheric Moisture

The applicant presented relative humidity, precipitation, and fog data summaries from the five NWS observation stations surrounding the LNP site as well as the 1-year period of record from February 1, 2007, through January 31, 2008, from the LNP onsite meteorological data.

2.3.2.4.1.4.1 Relative Humidity

Maximum relative humidity values usually occur during the early morning hours and minimum relative humidity values typically are observed in the mid-afternoon. The applicant summarized the monthly diurnal relative humidity based on data from five surrounding reporting stations (Tampa, Gainesville, Orlando, Tallahassee, and Jacksonville) in LNP COL FSAR Table 2.3.2-243.

The staff reviewed the data listed in the NCDC LCDs for each of the five surrounding NWS observations stations to verify the relative humidity statistics presented by the applicant and discussed in the LNP COL FSAR. The staff concludes that the values presented by the applicant are correct.

2.3.2.4.1.4.2 Precipitation

LNP COL FSAR Table 2.3.2-244 compared long-term monthly and annual precipitation measurements from the five reporting stations (Tampa, Gainesville, Orlando, Tallahassee, and Jacksonville, Florida) to the 1-year period of record of February 1, 2007, through January 31, 2008, from the LNP onsite meteorological data. The staff has independently verified that the average monthly precipitation data recorded onsite are generally consistent with the historical data recorded at the reporting stations near the LNP site. Table 2.3.2-244 shows that Tallahassee, Florida has a higher annual average amount of precipitation than the other reporting stations. However, the total annual precipitation for 2007 was similar for all five of the stations (between 38.49 and 46.09 inches). Although LNP COL FSAR Table 2.3.2-244 only contains onsite data for 1 year, the staff has confirmed that the onsite precipitation data for the two year period from February 1, 2007, through January 31 2009, are consistent.

The staff reviewed the data listed in the NCDC LCDs for each of the five surrounding NWS observations stations to verify the precipitation statistics presented by the applicant and discussed in the LNP COL FSAR. The staff concludes that the values presented by the applicant are correct.

2.3.2.4.1.5 Fog

The applicant summarized the occurrence of fog based on data from five surrounding weather reporting stations (Tampa, Gainesville, Orlando, Tallahassee, and Jacksonville, Florida). On average, there are 43, 23, 39, 43, and 43 days per year that fog is recorded at Tampa, Gainesville, Orlando, Tallahassee, and Jacksonville, respectively. LNP COL FSAR Table 2.3.2-245 presented the average number of days of fog per month. The staff has independently confirmed the values presented in this table as correct and adequate.

The staff reviewed the data listed in the NCDC LCDs for each of the five surrounding NWS observations stations to verify the fog statistics presented by the applicant and discussed in the LNP COL FSAR. The staff concludes that the values presented by the applicant are correct.

2.3.2.4.1.6 Atmospheric Stability

The applicant classified atmospheric stability in accordance with the guidance provided in RG 1.23, Revision 1. Atmospheric stability is a critical parameter for estimating atmospheric dispersion characteristics in LNP COL FSAR Sections 2.3.4 and 2.3.5. Dispersion of effluents is greatest for extremely unstable atmospheric conditions (i.e., Pasquill stability Class A) and decreases progressively through extremely stable conditions (i.e., Pasquill stability Class G). The applicant based its stability classification on temperature change with height (i.e., delta-temperature or $\Delta T/\Delta Z$) between the 60-m and 10-m height, as measured by the LNP onsite meteorological measurements program from February 1, 2007, through January 31, 2009.

Frequency of occurrence for each stability class is one of the inputs to the dispersion models used in LNP COL FSAR Sections 2.3.4 and 2.3.5. The applicant included these data in the form of a joint frequency distribution (JFD) of wind speed and direction data as a function of stability class. A comparison of a JFD developed by the staff from the hourly data submitted by the applicant with the JFD developed by the applicant showed reasonable agreement.

The applicant used the 2-year period of record of onsite meteorological data to produce statistics on the temporal variations of atmospheric stability as shown in LNP COL FSAR Tables 2.3.2-201 through 2.3.2-240. The staff confirmed the pattern of stability classes presented by the applicant in LNP COL FSAR Section 2.3.2.1.7. This pattern showed that the most frequent stability classes were either neutral (D) or slightly stable (E). By creating a staff derived JFD of wind speed, wind direction, and atmospheric stability and comparing it against the applicant's JFD, the staff was able to independently confirm the values presented by the applicant as correct and adequate.

2.3.2.4.2 Potential Influence of the Plant and its Facilities on Local Meteorology

2.3.2.4.2.1 Topographical Description

The applicant stated that the LNP site and surrounding area is relatively flat, with no significant terrain features that will otherwise be expected to adversely or unusually impact natural dispersion downwind of the plant. In RAI 2.3.5-3, the staff asked the applicant to discuss the influence of the Gulf of Mexico and the resulting land and sea breezes on the atmospheric dispersion estimates around the plant. In its response to RAI 2.3.5-3, the applicant discussed the strong east-west wind direction that has been shown to occur in the site area. The applicant also discussed the lower wind speeds that have been documented at the 10-meter level of the meteorological tower. Due to the lower wind speeds and the strong east-west wind direction pattern, higher predictions of relative concentration (χ/Q) and relative deposition (D/Q) can be expected in the sectors with the highest wind direction frequency. The staff agrees with this assessment of influence from the Gulf of Mexico and considers RAI 2.3.5-3 to be closed. The results of the short and long term atmospheric dispersion analysis are discussed in LNP COL FSAR Sections 2.3.4 and 2.3.5. LNP COL FSAR Figure 2.3.2-222 shows topographic features within an 80-km (50-mi) radius of the LNP site. Through an NRC staff site visit (ML100780287) and United States Geological Survey (USGS) maps, the staff has independently verified the topographical assessment provided by the applicant and accepts the description as correct and adequate.

2.3.2.4.2.2 Fogging and Icing Effects Attributable to Cooling Tower Operation

Ground fogging could occur if ground elevations in the plant vicinity were comparable to expected heights of the cooling tower plumes. The applicant stated that the expected cooling towers for Units 1 and 2 are mechanical draft towers. The applicant states that ground level fogging could occur in the immediate vicinity of the mechanical draft cooling towers. However, those events would only be expected at onsite locations and under relatively cold and moist atmospheric conditions and when building wake and downwash effects have an adverse influence on the dispersion of the cooling tower plumes. Based on previous observations and cooling tower plume modeling results (details in following section of this SER), the staff agrees and accepts the applicant's discussion.

The applicant stated that there are no large safety-related plant structures or other nearby structures that are expected to be affected by icing from cooling tower plumes due to the meteorological conditions that could reasonably be expected to occur. Because of the few days (approximately 3 days per year) with ambient temperatures below freezing at the Orlando and Tampa reporting stations, the staff agrees that the threat of ice formation is sufficiently low. The staff agrees and accepts the applicant's discussion of icing effects.

2.3.2.4.2.3 Assessment of Heat Dissipation Effects on the Atmosphere

LNP Units 1 and 2 will use two mechanical draft cooling towers to dissipate heat to the atmosphere. Potential meteorological effects due to operation of the cooling towers may include enhanced ground-level fogging and icing, cloud shadowing and precipitation enhancement, and increased ground-level humidity.

The applicant used the EPA's CALPUFF computer model to evaluate cooling tower plume behavior and to estimate the frequency of occurrence and length of visible cooling tower plumes.

The staff used the Seasonal/Annual Cooling Tower Impact (SACTI) computer code for estimating the impacts from fogging, icing, and drift deposition from the operation of the mechanical draft cooling towers. The staff found that there is a minimal threat of fogging and icing in the vicinity immediately surrounding the cooling towers. The staff agrees with the applicant's statement that because the closest public road (US Highway 19) is located 1.4 km (0.9 mi) from the nearest cooling tower, additional fogging and icing is not predicted or expected to occur in the vicinity of any public roadway.

The applicant also stated that a small amount of dissolved and suspended solids may result in solid particle deposition on the surface, primarily in close proximity to the plant. The staff has determined that nearly two months of salt accumulation would result in 0.07 milligrams per cubic centimeter (mg/cm^2), which is near the upper end of the "Light Contamination Level" range defined by the Institute of Electrical and Electronic Engineers (IEEE) standard[6]. The staff believes that total accumulation reaching amounts that require mitigation is highly unlikely due to local precipitation removing any salt deposits before it reaches a level of concern.

The staff independently confirmed the information presented in this FSAR section. The staff agrees and accepts the applicant's conclusion.

2.3.2.4.3 Local Meteorological Conditions for Design and Operating Basis

Meteorological conditions for design and operating basis are discussed in LNP COL FSAR Section 2.3.1.2 and reviewed by the staff in Section 2.3.1.4.2 of this SER.

2.3.2.5 *Post Combined License Activities*

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

2.3.2.6 *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 regional climatology and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

COL Information Item 2.3-2 states that a COL applicant shall address the site-specific local meteorological information. As set forth above, the applicant has presented and substantiated information describing the local meteorological conditions and topographic characteristics important to evaluating the adequacy of the design and siting of this plant. The staff has reviewed the information provided and for the reasons given above, concludes that the identification and consideration of the meteorological and topographical characteristics of the

---

[6] IEEE Guide for Application of Power Apparatus Bushings, IEEE Standard C.57.19.100-1995, Aug 1995.

site and the surrounding area are acceptable and meet the requirements of 10 CFR 100.20(c) and 10 CFR 100.21(d). The staff finds that the applicant has provided a sufficient description to adequately address COL Information Item 2.3-2 (COL Action Item 2.3.2-1).

The staff also 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 and topographic characteristics. Because the applicant has correctly implemented these methodologies, as described above, the staff has determined 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 Measurement Programs

2.3.3.1 *Introduction*

The LNP onsite meteorological measurement program addresses the need for onsite meteorological monitoring and the resulting data. The NRC staff 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.

This section verifies that the applicant successfully implemented an appropriate onsite meteorological measurements program and that data from this program provides an acceptable basis for estimating atmospheric dispersion for design-basis accidents (DBAs) and routine releases from an AP1000 design.

2.3.3.2 *Summary of Application*

Section 2.3 of the LNP COL FSAR, Revision 9 incorporates by reference Section 2.3 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Section 2.3, the applicant provided the following:

AP1000 COL Information Item

- LNP COL 2.3-3

The applicant provided additional information in LNP COL 2.3-3 to address COL Information Item 2.3-3 (COL Action Item 2.3.3-1). LNP COL 2.3-3 addresses the onsite meteorological measurements program.

In addition, this LNP COL FSAR section addresses Interface Item 2.9 related to the onsite meteorological measurement program.

2.3.3.3 *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 onsite meteorological measurements program are given in Section 2.3.3 of NUREG-0800.

The applicable regulatory requirements for identifying onsite meteorological measurements program are as follows:

- 10 CFR 100.20(c)(2), with respect 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), with respect 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 offsite; and (2) radiological dose consequences of postulated accidents meet prescribed dose limits at the exclusion area boundary (EAB) and the outer boundary of the low population zone (LPZ).

- 10 CFR Part 50, “Domestic licensing of production and utilization facilities,” Appendix A, “General Design Criteria for Nuclear Power Plants,” General Design Criterion (GDC) 19, “Control Room,” with respect to the meteorological considerations used to evaluate the personnel exposures inside the control room during radiological and airborne hazardous material accident conditions.

- 10 CFR 50.47(b)(4), 10 CFR 50.47(b)(8), and 10 CFR 50.47(b)(9), as well as Section IV.E.2 of 10 CFR Part 50, Appendix E, “Emergency Planning and Preparedness for Production and Utilization Facilities,” with respect to the onsite meteorological information available for determining the magnitude and continuously assessing the impact of the releases of radiological materials to the environment during a radiological emergency.

- 10 CFR Part 50, Appendix I, “Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criteria,” with respect to meteorological data used in determining the compliance with 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 Part 20, "Standards for Protection Against Radiation," Subpart D, "Radiation Dose Limits for Individual Members of the Public," with respect to the meteorological data used to demonstrate compliance with dose limits for individual members of the public.

The following RG is applicable to this section:

- RG 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants," Revision 1

The related acceptance criteria from Section 2.3.3 of NUREG-0800 are as follows:

- The preoperational and operational monitoring programs should be described, including: (1) a site map (drawn to scale) that shows tower location and true north with respect to man-made structures, topographic features, and other features that may influence site meteorological measurements; (2) distances to nearby obstructions of flow in each downwind 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, archiving, and analysis 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. These data should be used by the applicant to calculate: (1) the short-term atmospheric dispersion estimates for accident releases discussed in SER Section 2.3.4; and (2) the long-term atmospheric dispersion estimates for routine releases discussed in SER Section 2.3.5.

The applicant should identify and justify any deviations from the guidance provided in RG 1.23, Revision 1. Deviations from guidance are discussed in further detail in Chapter 1 of this SER.

2.3.3.4 *Technical Evaluation*

The NRC staff reviewed Section 2.3.3 of the LNP COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL applications 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 onsite meteorological measurements program. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the LNP COL FSAR:

<u>*AP1000 COL Information Item*</u>

- LNP COL 2.3-3

The NRC staff reviewed LNP COL 2.3-3 related to the onsite meteorological measurements program included under Section 2.3 of the LNP COL FSAR. The specific text of this COL information item in Section 2.3.6.3 of the AP1000 DCD states:

> Combined License applicants referencing the AP1000 certified design will address the site-specific onsite meteorological measurements program.

The staff's evaluation is based on the descriptions provided by the applicant in LNP COL FSAR Section 2.3.3 and a pre-application readiness assessment held November 3-7, 2008. The purpose of the readiness assessment was to: (1) become familiar with the prospective applicant's site and site selection process, plans, schedules, and initiatives; (2) observe and review the preoperational onsite meteorological monitoring program; and (3) review the prospective applicant's plans for its operational onsite meteorological monitoring program.

The NRC staff relied upon the review procedures presented in NUREG-0800, Section 2.3.3, to independently assess the technical sufficiency of the information presented by the applicant.

2.3.3.4.1 Preoperational Meteorological Measurement Program

The onsite meteorological measurements program at the LNP site began in February 2007. LNP COL FSAR Figure 2.3.3-201 shows the location of the meteorological tower with respect to LNP Units 1 and 2 along with the topography of the site. RG 1.23, Revision 1, Section 3 describes an acceptable method for siting of the onsite meteorological observation tower. The meteorological tower is a 60.4-m guyed, open-latticed meteorological tower located at an elevation of approximately 13.7 m. The largest structures in the vicinity that have the potential to influence the meteorological measurements are the mechanical draft cooling towers. RG 1.23, Revision 1, indicates that obstructions to flow (such as buildings) should be located at least 10 obstruction heights from the meteorological tower to prevent adverse building wake effects. The height of the closest mechanical draft cooling tower is 17.1 m. The LNP meteorological tower, located approximately 600 m from the nearest cooling tower, is of a sufficient distance. The tower design is consistent with the guidance provided in RG 1.23, Revision 1; therefore, it is acceptable to the staff.

Due to the relatively short distance between the mechanical draft cooling towers and the onsite meteorological tower, the staff was concerned that moist plumes from the cooling towers could potentially affect the measurements taken at the meteorology tower. In RAI 2.3.3-7, the staff requested that the applicant provide a discussion in the LNP COL FSAR on the potential effects of the cooling tower plumes on the onsite meteorological system measurements during plant operation. In response to RAI 2.3.3-7, the applicant provided a discussion on the frequency in which the meteorological measurements system may be impacted by plumes from the mechanical draft cooling towers. The applicant stated that visible plumes greater than 500 m in length are expected to occur less than 3 percent of the time. Winds from the northeast (the direction from cooling towers to the meteorological tower) have been observed to occur 9 percent of the time during the 2-year period from February 1, 2007, through January 31, 2009. Therefore, a visible plume from the cooling towers extending to the meteorological monitoring tower would occur less than 0.3 percent of the time. Through cooling tower plume modeling and analysis of the hourly onsite meteorological dataset, the staff has confirmed the information provided in the RAI response and the discussion and accepts it as correct and adequate. Therefore, the staff considers the clarifications requested in RAI 2.3.3-7 to be resolved. The applicant has agreed to update the FSAR to include this information. Therefore, LNP COL FSAR Section 2.3.3.1 is being tracked as **Confirmatory Item 2.3.3-1**.

<u>Resolution of Confirmatory Item 2.3.3-1</u>

Confirmatory Item 2.3.3-1 is an applicant commitment to update section 2.3.3.1 of its FSAR. The staff verified that LNP COL FSAR Section 2.3.3.1 was appropriately updated. As a result, Confirmatory Item 2.3.3-1 is now closed.

Also, the applicant describes the area surrounding the tower as relatively flat. Moreover, the area is flat within several miles of the site with no appreciable or noticeable variation in the terrain that would have a significant influence on the observed meteorological parameters. Through the use of USGS maps, the staff as confirmed the information presented in the LNP COL FSAR in regards to the topography surrounding the meteorological tower. In the immediate vicinity of the tower base and within the security fence, gravel has been used as a means of controlling weeds. The presence of this gravel is not extensive and is not expected to have an influence on the measured parameters. The staff confirmed the description of the area immediately surrounding the meteorological tower during an NRC staff site visit (ML100780287).

LNP COL FSAR Table 2.3.3-201 provides the heights at which the measurements are made in the onsite meteorological measurements program. Wind speed and direction, ambient temperature, and delta-temperature are recorded at both the 60 m and 10 m levels. Dew point temperature is recorded at the 10 m level. Solar radiation, precipitation and barometric pressure are all recorded at the base of the tower at the 2.0 m level. The height of the measurements complies with the recommended instrument heights described in Section 2 of RG 1.23, Revision 1; therefore, they are acceptable to the staff.

2.3.3.4.1.1 Wind Systems

Wind direction, wind speed, and wind direction variance (i.e., sigma theta) are monitored at both the lower- (10-m) and upper-level (60-m) of the tower. The wind sensors are mounted on a 3.7-m retractable boom oriented perpendicular to the prevailing wind flow. Section 3 of RG 1.23, Revision 1, states that wind sensors should be mounted at a distance equal to at least twice the horizontal dimension of the tower. This is to reduce the possible interference of the tower structure on the wind instruments. As shown in LNP COL FSAR Table 2.3.3-202, these measurements are based on the guidance provided in Sections 2 and 3 of RG 1.23, Revision 1; therefore, the wind systems are acceptable to the staff.

2.3.3.4.1.2 Temperature Systems

Ambient temperature and delta-temperature are monitored at both the lower- and upper-level of the tower. Two channels of differential temperature are monitored simultaneously between the lower- and upper-levels. The temperature probes are mounted in aspirated shields attached to a 2.5-m retractable boom. Dew point temperature is measured at the 10-m level of the tower. Section 3 of RG 1.23, Revision 1, states that ambient temperature and moisture measurements should be made to avoid air modification by heat and moisture sources. As shown in LNP COL FSAR Table 2.3.3-202, the temperature systems are based the guidance provided in Sections 2 and 3 of RG 1.23, Revision 1; therefore the temperature systems are acceptable to the staff.

2.3.3.4.1.3 Precipitation and Solar Radiation Systems

Precipitation and solar radiation are measured at 2.0 m above ground-level by sensors located near the base of the tower. As shown in LNP COL FSAR Table 2.3.3-202, the precipitation sensor is based on RG 1.23, Revision 1, and the solar radiation sensor is based on American National Standards Institute/American Nuclear Society (ANSI/ANS) 2.5-1984[7]. Since no accuracies are specified in RG 1.23, Revision 1 for solar radiation instrumentation, the applicant has chosen to follow the recommendations in ANSI/ANS 2.5-1984, a document endorsed by the NRC. Therefore, the precipitation and solar radiation systems are acceptable to the staff.

2.3.3.4.1.4 Maintenance and Calibration

The applicant stated that the meteorological equipment is checked and calibrated on a routine basis in accordance with RG 1.23, Revision 1. In order to achieve the required level of system reliability, as specified in RG 1.23, Revision 1, the applicant employs the following maintenance techniques: (1) calibrating the datalogger input channels semiannually; (2) calibrating or replacing the wind sensors with National Institute for Standards and Technology (NIST)-traceable calibrated sensors semiannually; (3) calibrating or replacing barometric pressure, dew-point temperature, and solar radiation channel sensors with NIST-traceable calibrated sensors; (4) calibrating and checking the consistency between the two ambient/differential temperature channels; (5) checking the guyed wires and the tower anchors annually.

In RAI 2.3.3-8, the staff requested that the applicant update the LNP COL FSAR to clarify how often the onsite meteorological temperature sensors (thermistors) are calibrated and replaced. In response to RAI 2.3.3-8, the applicant stated that LNP COL FSAR Section 2.3.3.1.4 will be updated to clarify that the thermistors are calibrated every 6 months to ensure proper sensor

---

[7] ANS, 1984. Standard for Determining Meteorological Information at Nuclear Power Sites. ANSI/ANS-2.5-1984. American Nuclear Society, La Grange Park, IL.

operation. This commitment to update the FSAR is being tracked as **Confirmatory Item 2.3.3-2**.

Resolution of Confirmatory Item 2.3.3-2

Confirmatory Item 2.3.3-2 is an applicant commitment to update Section 2.3.3 of its FSAR. The staff verified that LNP COL FSAR Section 2.3.3 was appropriately updated. As a result, Confirmatory Item 2.3.3-2 is now closed.

The staff finds that the instrument maintenance and calibration techniques are consistent with the guidance provided in RG 1.23, Revision 1; therefore, they are acceptable to the staff.

2.3.3.4.1.5 Data Reduction

The applicant described in Section 2.3.3.1.5 of the LNP COL FSAR that data from the datalogger, located near the meteorological tower, are retrieved via remote connection through a cellular telephone link. Using a host computer, an offsite meteorological consultant retrieves the meteorological data from the datalogger on a daily basis (except weekends and holidays). The data is compared against data from an Automatic Weather Observing System operated by the municipality of Ocala, as well as data from the nearby Crystal River Energy Complex. Erroneous data are discarded prior to being saved in the historical database. The edited and reviewed 15-minute averaged data are then stored on electronic media.

RG 1.23, Revision 1, states that: (1) the digital sampling rate for data should be at least once every 5 seconds; (2) digital data should be compiled as 15-minute average values for real-time display in the appropriate emergency response facilities; and (3) digital data should be compiled and archived as hourly values for use in historical climatic and dispersion analyses. Data should also be compiled into annual JFDs of wind speed and wind direction by atmospheric stability class.

The following information, summarized from Section 2.3.3.1.5 of the LNP COL FSAR, is the routine output as part of the onsite measurements program:

- Temperature, pressure, precipitation, solar radiation, and dew-point temperature summaries as daily and monthly averages

- Hourly precipitation totals

- Hourly averages of pressure, temperature, delta-temperature, dew-point temperature, upper- and lower-level wind direction and wind speed, upper- and lower-level wind direction variance, Pasquill stability classes, and accumulated solar radiation

- 15-minute averages of all parameters, except precipitation, which is a 15-minute total value

- Joint frequency distributions of upper- and lower-level wind speed, wind direction, and Pasquill stability class

The data reduction and compilation techniques listed above comply with the recommendations provided in RG 1.23, Revision 1 and are therefore acceptable to the staff.

2.3.3.4.1.6 Accuracy of Measurements

LNP COL FSAR Table 2.3.3-202 summarizes the accuracy of the measurements taken as part of the LNP onsite meteorological measurements program. The accuracy of the 2-year period of record for the data provided was consistent with the requirements of RG 1.23, Revision 1, with the exception of the dew-point temperature measurements, which met the requirements of NRC endorsed ANSI/ANS 2.5-1984. Therefore, the accuracy of the measurements is acceptable to the staff.

2.3.3.4.2 Operational Meteorological Measurement Program

The applicant stated that the operational meteorological monitoring program will be a continuation of the pre-operational program. The pre-operational and operational monitoring programs are described jointly in the LNP COL FSAR. Since the pre-operational monitoring program meets the guidance provided in RG 1.23, Revision 1, the staff finds the continuation of this program to be acceptable.

2.3.3.4.3 Meteorological Data

As discussed in SER Section 2.3.2.4.1.1, the applicant provided JFDs of wind speed, wind direction, and atmospheric stability for both the 10-meter and 60-meter levels based on hourly measurements taken from February 1, 2007, through January 31, 2009.

The staff performed a quality review of the 2007-2009 hourly meteorological database using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data," issued July 1982. The staff used computer spreadsheets to perform further review. As expected, the staff's examination of the data revealed generally stable and neutral atmospheric conditions at night and unstable conditions during the day. Wind speed, wind direction, and stability class frequency distributions for each measurement channel were reasonable. As discussed in SER Section 2.3.2.4.1.1, the staff verified and accepts the lower- and upper-level JFDs and wind roses provided by the applicant.

In order to evaluate the accuracy of the 2007-2009 dataset, the staff compared the hourly temperature measurements to the observation sites at Jacksonville, Tallahassee, and Tampa. These comparisons showed agreeing patterns with the surrounding weather reporting sites. Based on these results, the staff believes that the data collected is acceptable and representative of site conditions.

Based on an independent quality review of the onsite meteorological data and a comparison with off-site weather reporting station data, the staff accepts the 2-years of onsite data provided by the applicant. The staff has determined that the data is representative of the site and is an acceptable basis for estimating atmospheric dispersion for accidental and routine releases in LNP COL FSAR Sections 2.3.4 and 2.3.5.

2.3.3.5 *Post Combined License Activities*

Part 10 of the LNP COL application describes proposed COL conditions, including inspection, test, analysis, and acceptance criteria (ITAAC). Table 3.8-1 in Part 10 of the COL application includes the emergency planning (EP) ITAAC. The following EP ITAAC involve demonstrating that the operational onsite meteorological monitoring program appropriately supports the LNP emergency plan:

- EP Program Element 7.1: The licensee has established a technical support center (TSC), which receives, stores, processes, and displays plant and environmental information, and enables the initiation of emergency measures and the conduct of emergency assessment (Acceptance Criteria 7.1.5).

- EP Program Element 7.2: The licensee has established an emergency operating facility in which meteorological data is acquired, displayed, and evaluated pertinent to offsite protective measures (Acceptance Criteria 7.2.2).

- EP Program Element 7.6: The means exists to provide meteorological information, consistent with Appendix 2 of 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. LNP meteorological equipment will be able to assess and monitor actual or potential offsite consequences of a radiological condition related to atmospheric measurements (Acceptance Criteria 7.6).

- EP Program Element 8.3: The means exists 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.

- EP Program Element 8.4: The means exists to acquire and evaluate meteorological information.

EP, including EP ITAAC are addressed in SER Section 13.3, “Emergency Planning.”

2.3.3.6 *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 onsite meteorological measurements program, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff’s technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

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 has reviewed the information provided in LNP COL 2.3-3. The staff concludes that the applicant has established consideration of the onsite meteorological measurements program and the resulting database 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 Appendix I to 10 CFR Part 50. 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 Appendix E to 10 CFR Part 50. Part 5, "Emergency Plan" of the LNP 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 (COL Action Item 2.3.3-1).

2.3.4 Short-Term Diffusion Estimates (Related to RG 1.206, Section C.III.1, Chapter 2, C.I.2.3.4, "Short-Term Atmospheric Dispersion Estimates for Accident Releases"

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."

2.3.4.2 *Summary of Application*

Section 2.3.4 of the LNP COL FSAR, Revision 9, incorporates by reference Section 2.3.4 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Section 2.3, the applicant provided the following:

AP1000 COL Information Item

- LNP COL 2.3-4

The applicant provided additional information in LNP COL 2.3-4 to address COL Information Item 2.3-4 (COL Action Item 2.3.4-1). LNP 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 LNP 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.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the short-term diffusion estimates are given in Section 2.3.4 of NUREG-0800.

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, with respect 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) with respect 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), with respect to the atmospheric dispersion characteristics used in the evaluation of the EAB and LPZ radiological dose consequences for postulated accidents.

The following RGs are applicable to this section:

- RG 1.78, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," Revision 1

- RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1

- RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants"

The related acceptance criteria from Section 2.3.4 of NUREG-0800 are as follows:

- A description of the atmospheric dispersion models used to calculate χ/Q values for accidental releases of radioactive and hazardous materials to the atmosphere.
- 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
- 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.
- Hourly cumulative frequency distributions of χ/Q values from the effluent release point(s) to the EAB and LPZ should be constructed to describe the probabilities of these χ/Q values being exceeded.
- Atmospheric dispersion factors used for the assessment of consequences related to atmospheric radioactive releases to the control room for design basis, other accidents and for onsite and offsite 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 intake, and unfiltered inleakage pathways.

2.3.4.4 *Technical Evaluation*

The NRC staff reviewed Section 2.3.4 of the LNP COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the COL 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 short-term diffusion estimates. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information contained in the LNP COL FSAR:

AP1000 COL Information Item

- LNP COL 2.3-4

The NRC staff reviewed LNP COL 2.3-4 related to the short-term diffusion estimates included under Section 2.3.4 of the LNP COL FSAR. The COL Information Item 2.3-4 in Section 2.3.6.4 of the AP1000 DCD 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.

The NRC 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 computer code PAVAN (NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design-Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations") to estimate χ/Q values at the EAB and at the outer boundary of the LPZ for potential accidental releases of radioactive material. The PAVAN model implements the methodology outlined in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1.

The PAVAN code estimates χ/Q values for various time-averaged 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 distances for which χ/Q values are calculated.

For each of the 16 downwind direction sectors (e.g., N, NNE, NE, ENE), 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 values and the annual average (8760-hour) χ/Q values 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 is identified and becomes the short-term site characteristic χ/Q value for that time period.

In RAI 2.3.4-4, the staff requested that the applicant remove LNP COL FSAR Table 2.3.4-205 and the discussion associated with the 50 percent EAB and LPZ χ/Q values because they are discussed in the Environmental Report and are not compared against any AP1000 DCD site parameter. The applicant agreed to remove the table and the discussion from the FSAR. RAI 2.3.4-4 is, therefore, resolved. This agreement to update the FSAR is being tracked as **Confirmatory Item 2.3.4-1**.

Resolution of Confirmatory Item 2.3.4-1

Confirmatory Item 2.3.4-1 is an applicant commitment to update Section 2.3.4 of its FSAR. The staff verified that LNP COL FSAR Section 2.3.4 was appropriately updated. As a result, Confirmatory Item 2.3.4-1 is now closed.

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 methodology 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 Data Input

2.3.4.4.2.1 Offsite Dispersion Estimates

The meteorological input to PAVAN used by the applicant consisted of a JFD of wind speed, wind direction, and atmospheric stability based on hourly onsite data from a 2-year period from February 1, 2007 through January 31, 2009. The wind data were obtained from the 10-m 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 levels on the onsite meteorological tower.

The staff has completed a detailed review related to the acceptability and representativeness of the hourly meteorological data as discussed in SER Sections 2.3.2 and 2.3.3. 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 input to ARCON96 used by the applicant consisted of wind speed, wind direction, and atmospheric stability data based on hourly onsite data from a 2-year period from February 1, 2007, through January 31, 2009. The wind data were obtained from the 10-m and 60-m levels 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 levels on the onsite meteorological tower.

The staff has completed a detailed review related to the acceptability and representativeness of the hourly meteorological data as discussed in SER Section 2.3.3. 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 for its PAVAN model runs. The staff evaluated the applicability of the PAVAN diffusion parameters and concluded that no unique topographic features (such as rough terrain, restricted flow conditions, or coastal or desert areas) preclude the use of the PAVAN model for the LNP site. In RAI 2.3.5-3, the staff asked the applicant to discuss the influence of the Gulf of Mexico and the resulting land and sea breezes on the atmospheric dispersion estimates. The applicant responded by explaining that the daily interchanges of onshore and offshore flow directions appear to be contributing to low average wind speed. In general, the location of the LNP site would be expected to result in higher predictions of χ/Q values due to the lower wind speeds or to an increase in the frequency of wind directions in specific sectors. Based on an independent analysis of the short-term dispersion estimates, the staff accepts the applicant's description.

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 and incorporated by the applicant 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 wide range of meteorological conditions. Because 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. Therefore, the staff finds the applicant's use of the ARCON96 diffusion parameter assumptions acceptable.

2.3.4.4.4 Relative Concentration for Accident Consequences Analysis

2.3.4.4.4.1 Conservative Short-Term Atmospheric Dispersion Estimates for EAB and LPZ

The applicant modeled one ground-level release point and used the AP1000 DCD dimensions (AP1000 DCD Figure 3.8.2-1) for the minimum building cross section and containment heights for building wake effects. Including the building wake effects for a ground-level release has little influence on the predicted χ/Q values. A ground-level release assumption that assumes the appropriate building dimensions is acceptable to the staff. This is acceptable because the PAVAN model includes both plume meander and building wake effects, which are mutually exclusive. As discussed in LNP COL FSAR Section 2.1 the EAB and LPZ are defined as two overlapping circles centered on the reactor building of each unit.

While performing a confirmatory analysis of the χ/Q values for ground level releases, the staff questioned the applicant's use of a significant number of calm or light wind conditions in their PAVAN model run. In RAI 2.3.4-5, the staff requested that the applicant follow the guidance provided in Section C.1.1.1 of RG 1.145, which states that if the meteorological instrumentation conforms to RG 1.23 (i.e., if the wind sensors have a starting threshold less than 0.45 m/s) then calms should be assigned a wind speed equal to the vane or anemometer starting speed, whichever is higher. In response to this RAI, the applicant updated its methodology to include a JFD with lower wind speed classes to represent calms in accordance with meteorological instrumentation limits, as recommended in RG 1.145, Revision 1. The applicant also updated the χ/Q results to reflect the maximum possible sector-dependent 2-hour χ/Q values instead of extrapolated 0.5 percent sector-dependent χ/Q values from PAVAN. The highest 2-hour χ/Q values typically occur under stability Class G (extremely stable) conditions and at low wind speeds; the applicant generated its maximum possible sector-dependent 2-hour χ/Q values assuming G stability and a wind speed less than the wind sensor starting threshold of 0.45 m/s. The staff independently confirmed the applicant's results, and accepts the content of the RAI response as correct and adequate; therefore, RAI 2.3.4-5 is closed. The staff also finds that the LNP COL FSAR revised site characteristics in Table 2.0-201 remain bounded by the AP1000 DCD site parameters. The commitment to update the FSAR Section 2.3.4 text as well as FSAR Tables 2.3.4-201 through 2.3.4-204, and 2.0-201 to reflect the updated JFD and the maximum possible sector χ/Qs is being tracked as **Confirmatory Item 2.3.4-2**.

Resolution of Confirmatory Item 2.3.4-2

Confirmatory Item 2.3.4-2 is an applicant commitment to update Section 2.3.4 of its FSAR as well as FSAR Tables 2.3.4-201 through 2.3.4-204, and 2.0-201 to reflect the updated JFD and the maximum possible sector χ/Qs. The staff verified that LNP COL FSAR Section 2.3.4 and FSAR Tables 2.3.4-201, 2.3.4-202, 2.3.4-203, 2.3.4-204, and 2.0-201 were appropriately updated. As a result, Confirmatory Item 2.3.4-2 is now closed. LNP COL FSAR Table 2.3.4-201 compared the site-specific EAB and LPZ χ/Q values to the corresponding site parameters provided in AP1000 DCD Table 2-1. This comparison showed that the AP1000 DCD EAB and LPZ χ/Q values bounded the revised site-specific values provided by the applicant in its response to RAI 2.3.4-5.[8]

Using the information provided by the applicant, including the 10-m level joint frequency distributions of wind speed, wind direction, and atmospheric stability presented in LNP COL FSAR Tables 2.3.2-201 through 2.3.2-208, the staff confirmed the applicant's χ/Q values by running the PAVAN computer code and obtaining consistent results. The applicant's joint frequency distributions used twelve wind speed categories. These wind speed categories were based on RG 1.23, Revision 1, but included an additional category to correspond with the manufacturer's stated instrument threshold wind speed. The staff accepts the short-term χ/Q values presented by the applicant.

[8] 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.

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 ARCON96:

- Onsite Hourly Meteorological Data: February 1, 2007 – January 31, 2009
- AP1000 DCD Table 15A-7: Control Room Source / Receptor Data
- AP1000 DCD Figure 15A-1: Site Plan with Release and Intake Locations
- LNP COL FSAR Table 2.3.4-207: Release / Receptor Azimuthal Angles
- LNP COL FSAR Figure 2.1.1-203: Plant Layout on the LNP Site

Two receptor (i.e., air intake) points, the control room heating, ventilation, and air conditioning (HVAC) intake and control room door, were modeled for the following eight release points:

- Containment Shell
- Fuel Building Blowout Panel
- Fuel Building Rail Bay Door
- Steam Vent
- Power-Operated Relief Valve (PORV) / Safety Valves
- Condenser Air Removal Stack
- Plant Vent
- PCS Air Diffuser

LNP COL FSAR Table 2.3.4-206 compared the site-specific control room χ/Q values to the corresponding site parameters provided in the DCD. This comparison showed that the AP1000 control room χ/Q values conservatively bounded the site-specific values. This comparison is reproduced in LNP COL FSAR Table 2.0-201.

The staff confirmed the applicant's atmospheric dispersion estimates by running the ARCON96 computer model and obtaining similar results (i.e., values on average within ± 5.2 percent). 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.

2.3.4.4.5 Onsite and Offsite Hazardous Materials

A review of the identification of onsite and offsite hazardous materials that could threaten control room habitability is performed in SER Sections 2.2.1, 2.2.2, and 2.2.3. The accident scenarios, including release characteristics and atmospheric dispersion model descriptions are also found 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 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 short-term diffusion estimates, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP 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 (Related to RG 1.206, Section C.III.2, Chapter 2, C.I.2.3.5, "Long Term Atmospheric Dispersion Estimates for Routine Releases"

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.

2.3.5.2 *Summary of Application*

Section 2.3 of the LNP COL FSAR, Revision 9 incorporates by reference Section 2.3 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Section 2.3, the applicant provided the following:

AP1000 COL Information Item

- LNP COL 2.3-5

The applicant provided additional information in LNP COL 2.3-5 to address COL Information Item 2.3-5 (COL Action Item 2.3.5-1). LNP COL 2.3-5 addresses long-term χ/Q and D/Q estimates for calculating concentrations in air and the amount of material deposited on the ground as a result of routine releases of radiological effluents to the atmosphere during normal plant operation.

In addition, this LNP COL FSAR section addresses Interface Item 2.4 related to the limiting meteorological parameters (χ/Q values) for routine releases.

2.3.5.3 *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 long-term diffusion estimates are given in Section 2.3.5 of NUREG-0800.

The applicable regulatory requirements for the applicant's description of atmospheric dispersion and dry deposition estimates for routine releases of radiological effluents to the atmosphere are as follows:

- 10 CFR Part 20, Subpart D, with respect to demonstrating compliances 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 Sections II.B, II.C, and II.D of Appendix I of 10 CFR Part 50, with respect 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) with respect to establishing atmospheric dispersion site characteristics such that radiological effluent release limits associated with normal operation can be met for any individual located offsite.

The following RGs are applicable to this section:

- RG 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants," Revision 1

- 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 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1

- RG 1.112, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors," Revision 1

The related acceptance criteria from Section 2.3.5 of NUREG-0800 are as follows:

- A detailed description of the atmospheric dispersion and deposition models used by the applicant to calculate annual average concentrations in air and amount of material deposited as a result of routine releases or 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 22½ degree direction sector within a 5-mi [8-km] 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 Section 2.3.5.2 of RG 1.206:
(1) maximum annual average χ/Q values and D/Q values at or beyond the site boundary and at specified locations of potential receptors of interest utilizing 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 50 mi (80 km) from the plant using appropriate meteorological data.

2.3.5.4 *Technical Evaluation*

The NRC staff reviewed Section 2.3.5 of the LNP 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 long-term diffusion estimates. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the LNP COL FSAR:

AP1000 COL Information Item

- LNP COL 2.3-5

The NRC staff reviewed LNP COL 2.3-5 related to the long-term diffusion estimates included in Section 2.3.5 of the LNP COL FSAR. COL Information Item 2.3-5 (COL Action Item 2.3.5-1) in Section 2.3.6.5 of the AP1000 DCD 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.

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 χ/Q and D/Q values resulting from routine releases. The XOQDOQ model implements the constant mean wind direction methodology 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 for long time periods (i.e., annual averages), the plumes horizontal distribution is assumed to be evenly distributed within the downwind direction sector (e.g., "sector averaging"). A straight-line trajectory is assumed between the release point and all receptors.

2.3.5.4.2 Release Characteristics and Receptors

The applicant modeled one ground-level release point, assuming a minimum building cross-sectional area of 2,730 m^2 and a building height of 43.9 m (based on AP1000 DCD Figure 3.8.2-1), which is smaller than the height of the entire containment building at 71.4 m. This difference of height is acceptable to the staff because the applicant's building height directly leads to assuming a smaller building cross-section. This is a conservative assumption because a smaller building cross-section will lead to less air turbulence and higher χ/Q values.

The applicant assumed a ground-level release to model routine releases. A ground-level release is a conservative assumption at a relatively flat terrain site such as LNP, resulting in higher χ/Q and D/Q values when compared to a mixed-mode (e.g., part-time ground, 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., milk cow, milk goat, garden, meat animal, and resident) were presented in LNP COL FSAR Table 2.3.5-201. For sectors not containing a certain receptor type, the applicant assumed a distance of 5 mi. The applicant calculated the distances to each of these receptors from a location defined as the mid-point of the two proposed units. However, the staff has determined that using a shorter distance (outer edge of the power block area) results in χ/Q and D/Q values that are still bounded by the AP1000 DCD. The use of the shortest distance results in higher (more conservative) χ/Q values for ground level releases. Therefore, the assumptions presented by the applicant are acceptable to the staff.

2.3.5.4.3 Meteorological Data Input

The meteorological input to XOQDOQ used by the applicant consisted of a JFD of wind speed, wind direction, and atmospheric stability based on hourly onsite data from a 2-year period from February 1, 2007 through January 31, 2009. The wind data were obtained from the 10-m 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 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 SER Section 2.3.3, the staff considers the February 1, 2007, through January 31, 2009, onsite meteorological database suitable for input to the XOQDOQ model.

In response to RAI 2.3.5-3, the applicant stated that the proximity of the Gulf of Mexico to the site would be expected to have an influence on the wind direction and wind speed measurements. This influence would be expected to result in higher predictions of relative concentration (χ/Q) and relative deposition (D/Q), due to either low wind speeds or to an increase in the frequency of wind directions in specific sectors. This would be a result of the sea-breeze and land-breeze circulations expected near coastal sites. This pattern can be identified in LNP COL FSAR Tables 2.3.5-201 through 2.3.5-204. These tables show that the highest χ/Q and D/Q values are found in the WSW and W quadrants. This pattern corresponds with the downwind sectors of the most frequent wind directions identified in LNP COL FSAR Figure 2.3.2-201. The NRC was able to confirm the applicant's discussion through analysis of the hourly onsite data collected during a 2-year period from February 1, 2007 through January 31, 2009. Therefore, the staff finds the applicant's response acceptable and considers RAI 2.3.5-3 closed.

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, 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 LNP site. Therefore, the staff finds that the applicant's use of diffusion parameter assumptions, as outlined in RG 1.111, Revision 1 was acceptable.

In response to RAI 2.3.5-6, the applicant provided justification for the use of the XOQDOQ straight-line model trajectory model. The applicant stated that the LNP site is located in an area that is surrounded by flat terrain for a distance of more than 50 mi and that no significant special variations in dispersion or direction are expected to occur as a result of variations in terrain. The staff has reviewed this RAI response and agrees with the qualitative and quantitative statements made by the applicant. Therefore, the staff finds the applicant's response acceptable and considers RAI 2.3.5-6 closed.

2.3.5.4.5 Resulting Relative Concentration and Relative Deposition Factors

LNP COL FSAR Tables 2.3.5-201 through 2.3.5-204 lists the long-term atmospheric dispersion and deposition estimates for the EAB, LPZ, and special receptors of interest that the applicant derived from their XOQDOQ modeling results. LNP COL FSAR Tables 2.3.5-201 through 2.3.5-204 also describe the applicant's long-term atmospheric dispersion and deposition estimates for 16 radial sectors from the site boundary to a distance of 50-mi from the proposed facility.

The χ/Q values presented in LNP COL FSAR Tables 2.3.5-201 through 2.3.5-204 reflect several plume radioactive decay and deposition estimates for the EAB, LPZ, and special receptors of interest that the applicant derived from its XOQDOQ modeling results.

The χ/Q values presented in LNP COL FSAR Tables 2.3.5-201 through 2.3.5-204 reflect several plume radioactive decay and deposition scenarios. Section C.3 of RG 1.111, Revision 1 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. Section C.3.a of RG 1.111, Revision 1 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 are as follows:

- Undepleted/No Decay χ/Q values are χ/Q values 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 of radioactive decay

- Undepleted/2.26-Day Decay χ/Q values are χ/Q values 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-133.

- Depleted/8.00-Day Decay χ/Q values are χ/Q values 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 LNP COL FSAR Tables 2.3.2-201 through 2.3.2-208, the staff confirmed the applicant's χ/Q and D/Q values by running the XOQDOQ computer code and obtaining similar results (i.e., values on average within about 1-percent). The JFDs used by the applicant for the long-term diffusion estimates consisted of 11 wind speed categories. These wind speed categories were based on RG 1.23, Revision 1, but combined the first two non-calm wind speed categories into one category of 1.0-1.05 m/s. In light of the foregoing, the staff accepts the long-term χ/Q and D/Q values presented by the applicant.

COL Information Item 2.3-5 also states that, with regard to environmental assessment, estimates of annual average χ/Q values for 16 radial sectors to a distance of 50-mi from the plant should be provided. The applicant provided these values in LNP COL FSAR Tables 2.3.5-201 through 2.3.5-204. 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 related to this section.

2.3.5.6 *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 long-term diffusion estimates and there is no outstanding information expected to be addressed in the LNP COL FSAR relating to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL 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 50-mi (80-km) 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 Subpart D of 10 CFR Part 20 and Appendix I to 10 CFR Part 50. 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 one or more nuclear power plants can be safely operated on the applicant's proposed site and in accordance with the Commission's regulations, NRC staff evaluated the hydrologic site characteristics of the proposed site. These site characteristics included the maximum flood elevation of surface water and the maximum elevation of groundwater. The staff also described the characteristic ability of the site to attenuate a postulated accidental release of radiological material into surface water and groundwater before it reaches a receptor.

The staff prepared Sections 2.4.1 through 2.4.14 of this SER in accordance with the review procedures described in NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," using information presented in Section 2.4, "Hydrologic Engineering," of the Progress Energy Florida[9] (PEF) LNP Units 1 and 2 FSAR Revision 4, DCD Revision 19, applicant's responses to staff RAIs, and generally available reference materials (e.g., those cited in applicable sections of NUREG-0800).

The ultimate heat sink of the AP1000 reactor 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 a necessary part of the regulatory basis for this Section 2.4 review.

In Part 7 of the Combined License Application, the applicant described an administrative departure (STD DEP 1.1-1) that remaps Section 2.4 section numbers to the associated DCD section numbers. The staff determines that this departure has no safety significance.

---

[9] The applicant, Duke Energy Florida, LLC, was formerly identified as Progress Energy Florida, Inc. and Duke Energy Florida, Inc. In a letter dated April 15, 2013, Progress Energy Florida notified the NRC that its name was changing to Duke Energy Florida effective April 29, 2013. The name changes and a 2012 corporate merger between Duke Energy and Progress Energy are described in Chapter 1 of the SER. Because a portion of the review described in this chapter was completed prior to the name change, the NRC staff did not change references to "Progress Energy Florida" or "PEF" to "Duke Energy Florida" or "DEF" in this chapter.

2.4.1 Hydrologic Description

2.4.1.1 *Introduction*

FSAR Section 2.4.1 of the LNP COL application described 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 SER 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 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, item (6) above was not part of the staff's review.

2.4.1.2 *Summary of Application*

This section of the LNP COL FSAR 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. The applicant addressed these issues as follows:

<u>COL Information Item</u>

- LNP COL 2.4-1 Hydrological Description

In addition, this section addresses the following COL-specific information identified in Section 2.4.1.1 of Revision 19 of the DCD.

> 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*

The relevant requirements of the Commission regulations for the identification of floods and flood design considerations, and the associated acceptance criteria, are described in Section 2.4.1 of NUREG-0800 (NRC 2007a).

The applicable regulatory requirements for identifying site location and description of the site hydrosphere are as follows:

- 10 CFR Part 100, as it relates to identifying and evaluating hydrologic features of the site.

- 10 CFR 100.20(c), regarding requirements to consider physical site characteristics in site evaluations.

- 10 CFR 52.79(a)(1)(iii), 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.

The related acceptance criteria are as follows:

- RG 1.59, “Design Basis Floods for Nuclear Power Plants” (NRC 1977a), as supplemented by best current practices

- RG 1.102, “Flood Protection for Nuclear Power Plants” (NRC 1976b).

2.4.1.4 *Technical Evaluation*

The NRC staff reviewed Section 2.4.1 of the LNP 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.[10] The NRC staff’s review confirmed that the information in the application and incorporated by reference addresses the required information relating to the site hydrological description. The results of the NRC staff’s evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

[10] 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.

2.4.1.4.1 Site and Facilities

Information Submitted by the Applicant

The LNP site, 1,257 ha (3,105 ac) in size, is located southwest of Gainesville and west of Ocala in southern Levy County in Florida (Figure 2.4.1-1), approximately 12.8 km (8 mi) inland from the Gulf of Mexico, 4.8 km (3 mi) north of Lake Rousseau, and 15.5 km (9.6 mi) north of PEF's Crystal River Energy Complex (CREC). The two proposed units will be called LNP Unit 1 and LNP Unit 2.

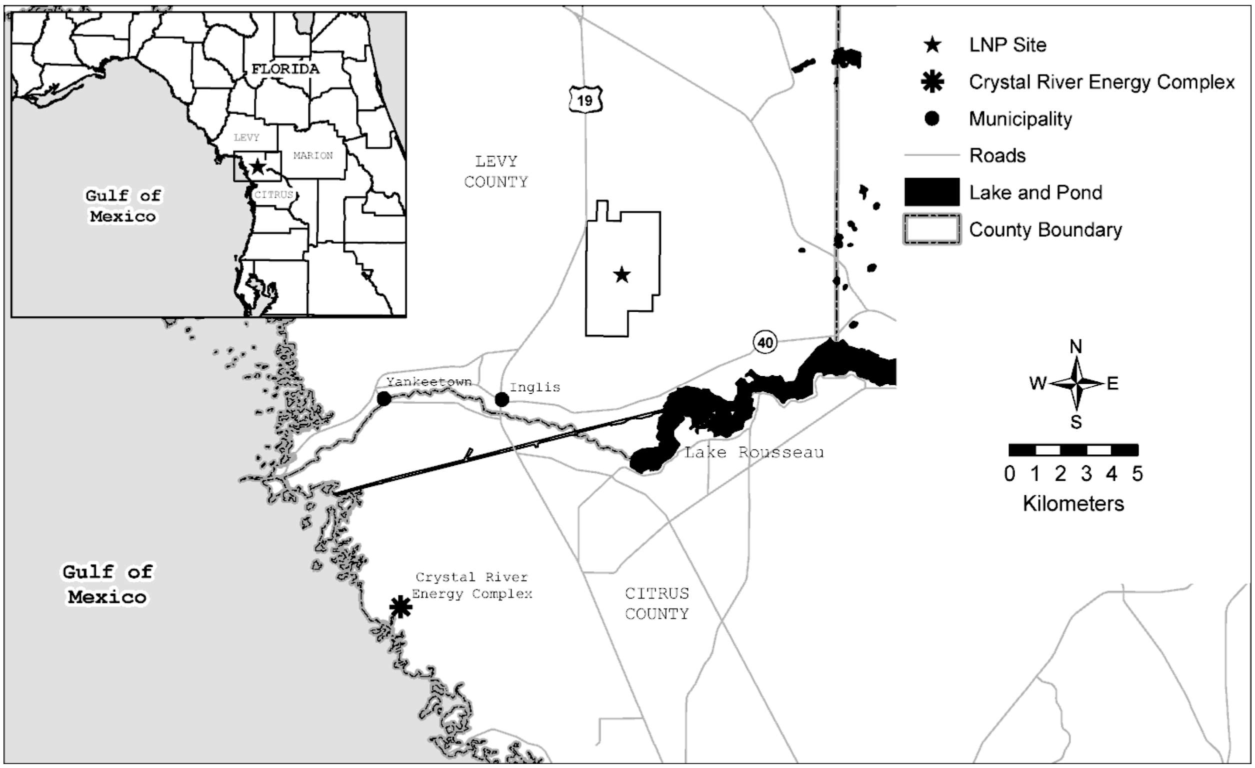


Figure 2.4.1-1. The LNP Site and Surrounding Area

Elevations at the LNP site range from 9.1 to 18.3 m (30 to 60 ft) National Geodetic Vertical Datum of 1929 (NGVD29). The applicant stated that the nominal plant grade would be 15.5 m (51 ft) NAVD88 with actual plant grade lower than 15.5 m (51 ft) NAVD88 (North American Vertical Datum of 1988) to accommodate drainage for local flooding. At the site audit, the applicant stated that elevation values referring to NGVD29 are approximately 0.3 m (1 ft) higher than the corresponding NAVD88 value on an average for the LNP site.

The Gulf of Mexico, the Cross Florida Barge Canal (CFBC), the Withlacoochee River, and Lake Rousseau are the major hydrologic features located near the LNP site. A 13.4-km (8.3 mi) stretch of the CFBC runs from below the Inglis Dam that impounds Lake Rousseau on the

Withlacoochee River to the Gulf of Mexico. Inglis Lock, Inglis Bypass Channel and Spillway, and the Inglis Dam are three water-control structures in the LNP site area and are operated by the South West Florida Water Management District (SWFWMD).

As stated in the FSAR, the proposed units will use a closed-loop normal cooling system with mechanical draft cooling towers. A new intake on the CFBC will provide cooling water for normal plant cooling. Two pipelines, one for each LNP unit, will discharge blowdown from the cooling towers to the existing CREC discharge canal. Onsite wells will provide water needed for general plant operations, including makeup to the service water system, potable water supply, and raw water to demineralized water, fire protection water, and media filter backwash.

NRC Staff's Technical Evaluation

The staff reviewed the information provided by the applicant to determine the adequacy of the information in support of hydrologic site characterization for the purpose of siting a nuclear reactor. The specific hydrology-related site characterization of the LNP site with respect to general description of the hydrosphere as described in NUREG-0800 (NRC 2007a) includes local intense precipitation, site drainage, probable maximum flood and associated water surface elevations, dam breaches and resulting flood elevations, storm surges and seiches with related flooding and low-water effects, tsunamis and associated flooding, ice formation, channel diversion, flooding protection requirements, safety-related water use, groundwater elevations, and accidental release of liquid radioactive effluents to ground and surface waters. The staff used the location of the LNP site, its hydrological and meteorological characteristics, and the interface of the plant with the elements of the hydrosphere to determine the site characteristics for safe siting and operation of the proposed LNP Units 1 and 2.

To ascertain the safe operation of a reactor at a site, the staff requires an accurate description of the site, the site region, and facilities at the site, including all safety-related facilities to determine whether the most conservative of plausible conceptual models are identified. In RAI 2.4.1-1, the staff requested additional information regarding the applicant's process to determine the conceptual models of the interface of the plant with the hydrosphere, including the hydrologic causal mechanisms to ensure that the most conservative of plausible conceptual models have been identified. In a letter dated June 15, 2009 (ML091680037), the applicant stated that the LNP site was characterized using conceptual modes that describe flooding from local intense precipitation, flooding in rivers and streams, flooding from upstream dam failures, and flooding from surges and tsunamis. In addition, the applicant also used conceptual site models to characterize subsurface properties and the accidental release of radioactive liquids.

The applicant stated that published information from local, State, and Federal agencies was used to document the physiography, hydrology, geology, meteorology, topography, and demography near the LNP site. The applicant also collected geological, hydrogeological, meteorological, and water quality data near the LNP site. The aforementioned data and information were used to develop site conceptual models. The applicant stated that conceptual site models developed for individual flood mechanisms, subsurface characteristics, and surface and subsurface pathways are described in responses to the staff's RAI corresponding to the respective FSAR sections.

The staff reviewed the applicant's response to RAI 02.04.01-01 and determined that the applicant appropriately used information and data published by local, State, and Federal agencies in addition to site-specific data to conceptualize the hydrologic mechanisms and site characteristics that may affect safety of proposed LNP Units 1 and 2. The staff concluded, therefore, that the applicant has provided sufficient information for describing the interface of the plant with the hydrosphere and to characterize the hydrologic causal mechanisms at and near the LNP site.

To perform its safety assessment, the staff requires an accurate description of the site, the site region, and facilities at the site, including all safety-related facilities. The staff conducted a hydrology site audit November 4–6, 2008. The staff's audit included a tour of the LNP site, the meteorological tower, the CFBC, the proposed makeup water intake location, the Inglis Lock, and the Inglis Bypass Channel and Spillway. To determine the accuracy and acceptability of the models used to estimate the design-basis flood, the staff issued **RAI 02.04.01-02**, which states:

> To meet the requirements of GDC 2, 10 CFR 52.17 and 10 CFR Part 100, the applicant should include a complete description of all spatial and temporal datasets used in support of its conclusions regarding safety of the plant. Data and descriptions should be sufficiently detailed to allow the staff to review the applicant's conclusions regarding the safety of the plant and to determine ~~of~~ the design bases of safety-related SSCs. Please provide input and output files associated with the HEC-HMS and HEC-RAS model simulations performed for the FSAR.

The applicant responded to the staff's RAI 02.04.01-02 in a letter dated June 23, 2009 (ML091830343). The applicant provided U.S. Army Corps of Engineers (USACE) Hydrologic Engineering Center-Hydrologic Modeling System (HEC-HMS; USACE 2010a) input and sample output data sets along with model control specifications and meteorological data. The applicant also provided USACE Hydrologic Engineering Center-River Analysis System (HEC-RAS; USACE 2010b) input and sample output datasets along with geometry data.

The staff reviewed the data sets provided by the applicant and determined that these data sets were suitable for staff to independently carry out a review of the applicant's flooding analyses. Subsequent subsections of this report describe the staff's independent and confirmatory analyses to verify the applicant's safety conclusions. To determine the appropriate and consistent usage of datums and elevations, the staff issued **RAI 02.04.01-03**, which states:

> To meet the requirements of GDC 2, 10 CFR 52.17 and 10 CFR Part 100, the applicant should include a complete description of all spatial and temporal datasets used by the applicant in support of its conclusions regarding safety of the plant. Data and descriptions should be sufficiently detailed to allow the staff to review the applicant's conclusions regarding the safety of the plant and to determine the design bases of safety-related SSCs. Please provide clarification regarding the use of the term MSL in the FSAR and clearly state the units of measurements and the contour interval on all the pertinent figures in the FSAR.

The applicant responded to the staff's RAI 02.04.01-03 in a letter dated June 15, 2009 (ML091680037). The applicant confirmed that its use of the term MSL in the FSAR can be converted to NGVD29 (Elevation ft NGVD29 = X ft MSL - 0.893 ft.). The applicant identified locations in the FSAR and changed text to replace the term MSL (or msl) with NGVD29. The applicant also stated the approximate elevation offset to convert elevations expressed in NGVD29 to NAVD88. The applicant also identified and fixed a typographical error. The applicant appropriately annotated some FSAR figures. The applicant made these changes in FSAR Revision 4.

The staff reviewed the applicant's response and determined that the applicant has corrected the inconsistencies in the FSAR. The staff independently used the National Oceanic and Atmospheric Administration (NOAA) National Geodetic Survey (NGS) VERTCON tool (NGS 2011) to verify that elevations near the LNP site referring to the NGVD29 datum are 0.31 m (1 ft) greater than those referring to the NAVD88 datum. Based on its independent review, the staff determined that the applicant's response to RAI 02.04.01-03 is acceptable.
The staff compared the information presented by the applicant in FSAR Section 2.4.1 with publicly available maps and data regarding the LNP site and its surrounding region. The proposed LNP site is located in Florida's Levy County approximately 71 km (44 mi) south-southwest from the City of Gainesville, Florida; 8 km (5 mi) east-northeast of Yankeetown, Florida; 4.8 km (3 mi) north of Inglis Lock on Lake Rousseau; and 16 km (10 mi) northeast of the CREC (Figure 2.4.1-1). The Gulf of Mexico is located approximately 13.7 km (8.5 mi) west-southwest of the LNP site.

2.4.1.4.2 Hydrosphere

Information Submitted by the Applicant

The LNP site lies mainly in the Waccasassa River Basin, with a small portion falling in the Withlacoochee River Basin (Figure 2.4.1-2). There are no named streams on the LNP site and the drainage is mainly overland toward the Lower Withlacoochee River and the Gulf of Mexico located southwest of the LNP site. Freshwater bodies in the vicinity include the Withlacoochee River and Lake Rousseau. Wetlands dominate the LNP site. Salt marshes are located between Highway 19 located west of the site and the Gulf of Mexico.

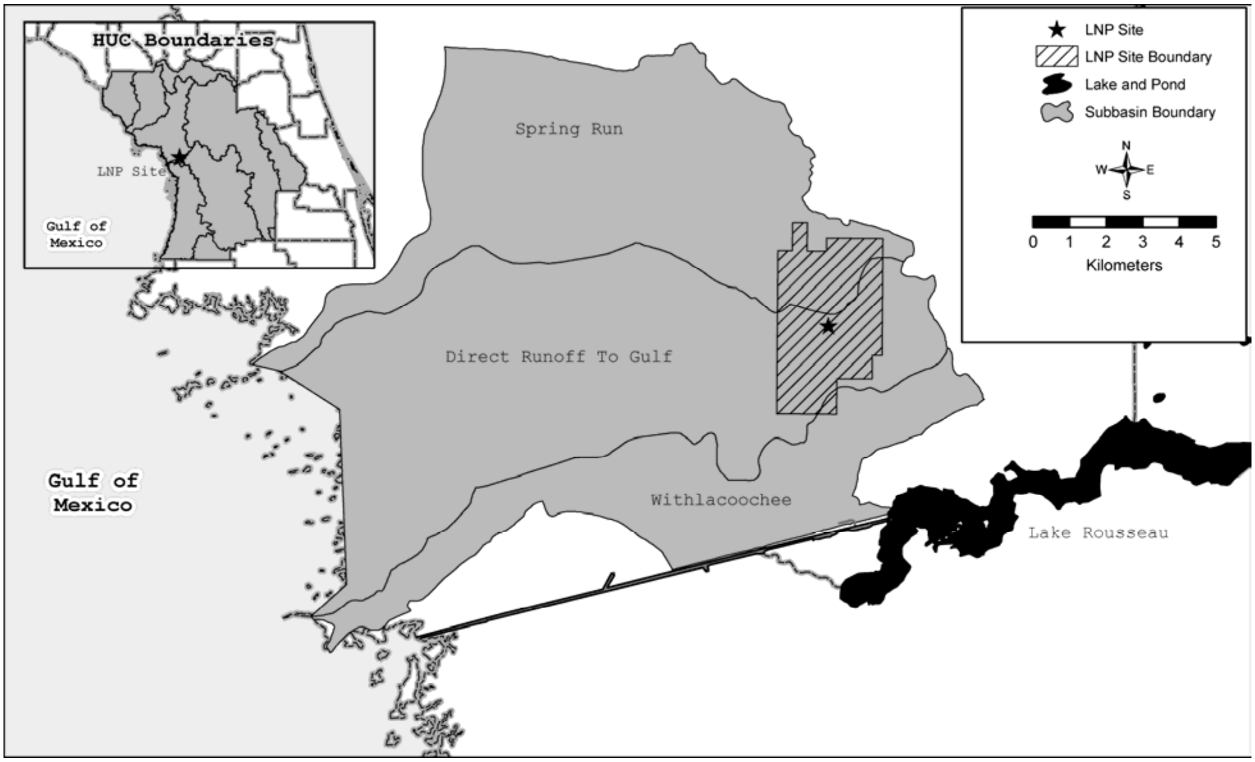


Figure 2.4.1-2. The Subbasins Within Which the LNP Site is Located

The Withlacoochee River Basin which has an area of 14,087 km^2 (5,439 mi^2), is partially located in the northern portion of the SWFWMD. The Withlacoochee River originates in Green Swamp and flows northwest approximately 253 km (157 mi) before discharging into the Gulf of Mexico near Yankeetown (Figure 2.4.1-3). The average gradient of the river is approximately 0.17 m/km (0.9 ft/mi). Little Withlacoochee River, Big Grant Canal, Jumper Creek, Shady Brook, Outlet River of Lake Panasoffkee, Leslie Heifner Canal, Orange State Canal, Tsala Apopka Outfall Canal, and Rainbow River are the major tributaries of the river. The Withlacoochee River and the Rainbow River contribute most of the water to Lake Rousseau.

The Upper Withlacoochee River extends from its headwaters in Green Swamp to its confluence with the Little Withlacoochee River. The Middle Withlacoochee River extends from its confluence with the Little Withlacoochee River downstream to U.S. Highway 41 approximately 1.0 km (0.6 mi) east of Lake Rousseau. The Lower Withlacoochee River extends from U.S. Highway 41 to its discharge in the Gulf of Mexico and includes Lake Rousseau, a portion of the CFBC, and the three water-control structures mentioned above. Rainbow River, fed by a first order natural spring, is 9.2 km (5.7 mi) in length and discharges approximately 21 m^3/s (727 cfs) daily into the Withlacoochee River.

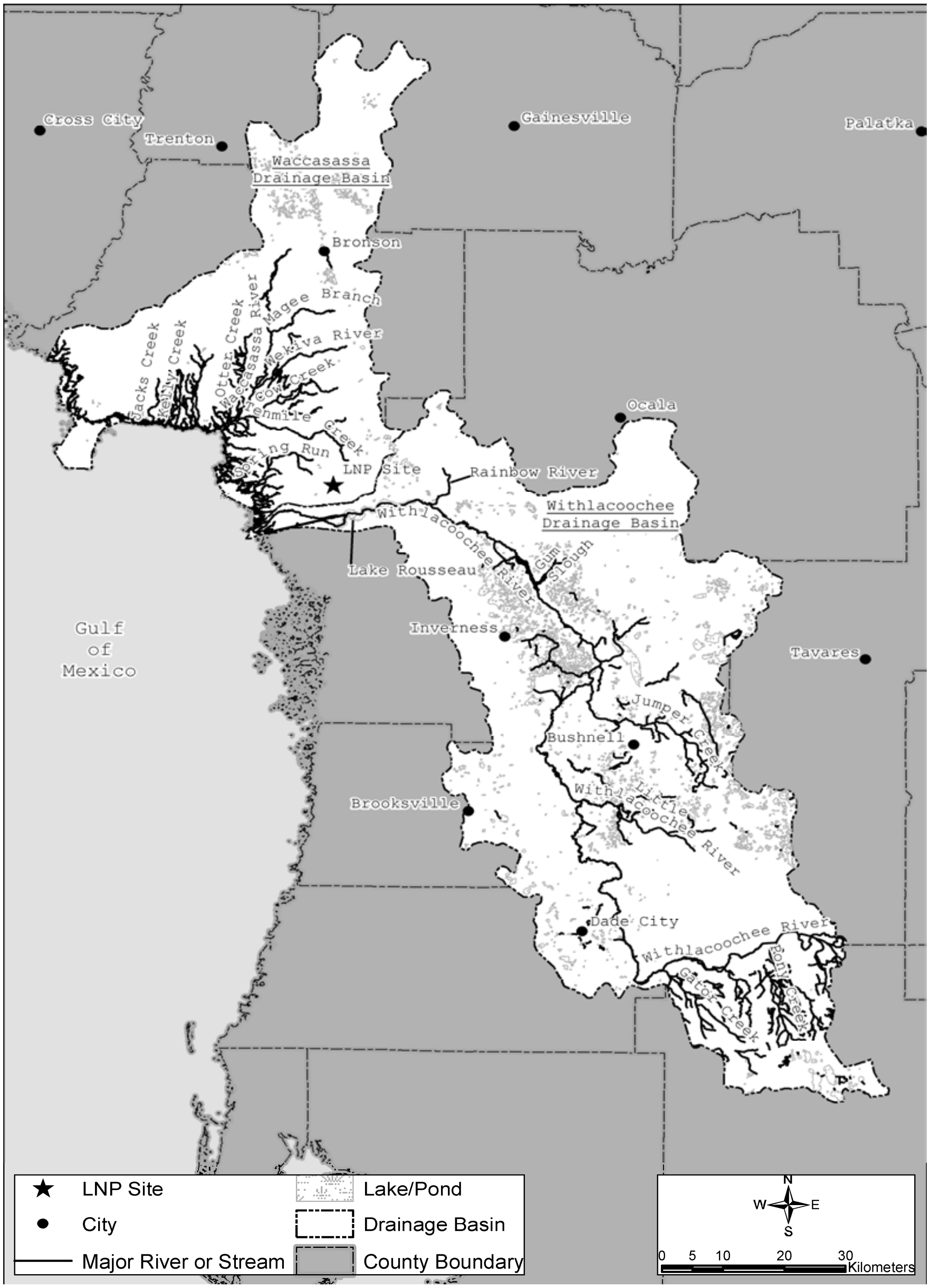
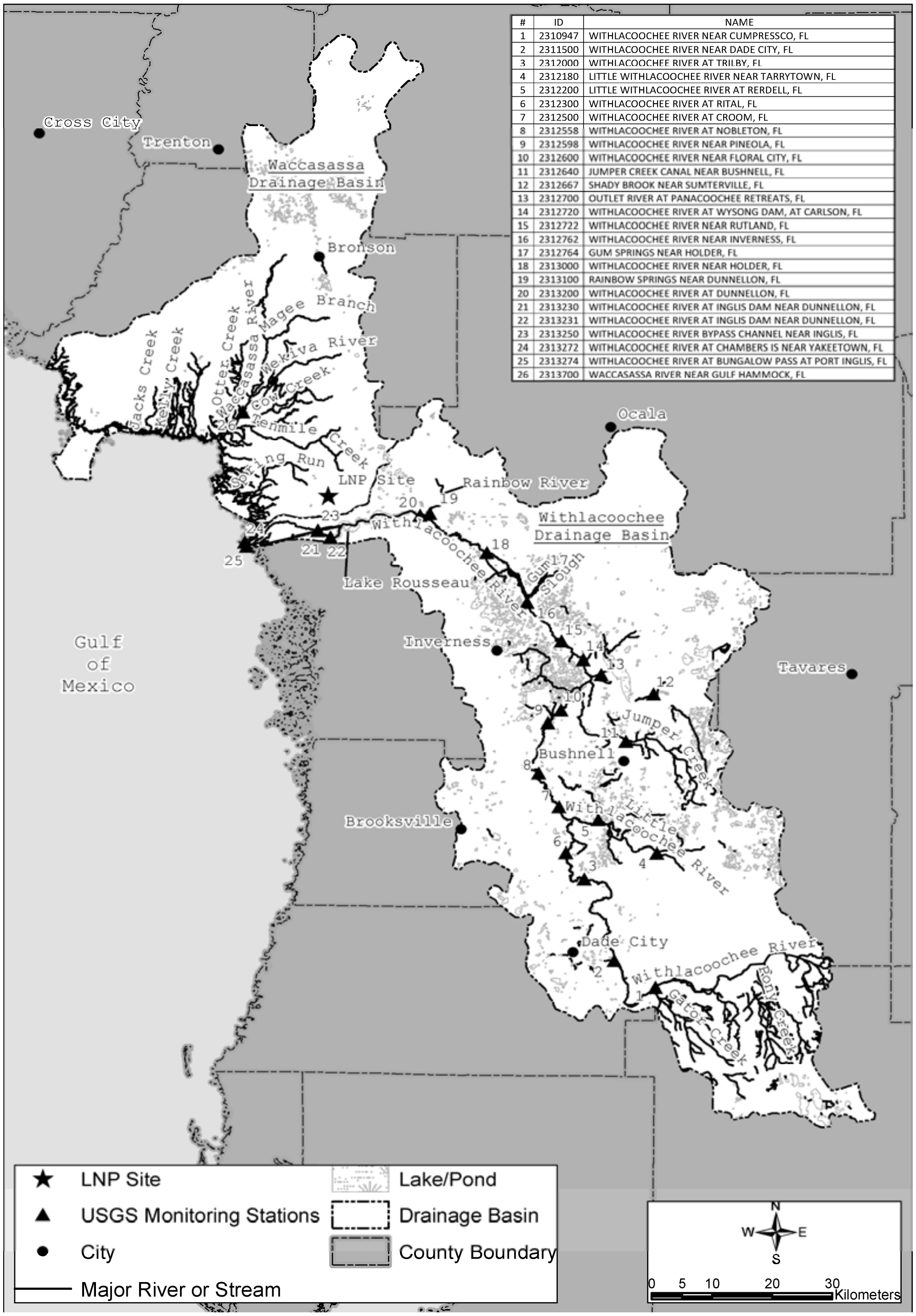


Figure 2.4.1-3. The Withlacoochee and Waccasassa River Basins



#	ID	NAME
1	2310947	WITHLACOOCHEE RIVER NEAR CUMPRESSCO, FL
2	2311500	WITHLACOOCHEE RIVER NEAR DADE CITY, FL
3	2312000	WITHLACOOCHEE RIVER AT TRILBY, FL
4	2312180	LITTLE WITHLACOOCHEE RIVER NEAR TARRYTOWN, FL
5	2312200	LITTLE WITHLACOOCHEE RIVER AT RERDELL, FL
6	2312300	WITHLACOOCHEE RIVER AT RITAL, FL
7	2312500	WITHLACOOCHEE RIVER AT CROOM, FL
8	2312558	WITHLACOOCHEE RIVER AT NOBLETON, FL
9	2312598	WITHLACOOCHEE RIVER NEAR PINEOLA, FL
10	2312600	WITHLACOOCHEE RIVER NEAR FLORAL CITY, FL
11	2312640	JUMPER CREEK CANAL NEAR BUSHNELL, FL
12	2312667	SHADY BROOK NEAR SUMTERVILLE, FL
13	2312700	OUTLET RIVER AT PANACOOCHEE RETREATS, FL
14	2312720	WITHLACOOCHEE RIVER AT WYSONG DAM, AT CARLSON, FL
15	2312722	WITHLACOOCHEE RIVER NEAR RUTLAND, FL
16	2312762	WITHLACOOCHEE RIVER NEAR INVERNESS, FL
17	2312764	GUM SPRINGS NEAR HOLDER, FL
18	2313000	WITHLACOOCHEE RIVER NEAR HOLDER, FL
19	2313100	RAINBOW SPRINGS NEAR DUNNELLON, FL
20	2313200	WITHLACOOCHEE RIVER AT DUNNELLON, FL
21	2313230	WITHLACOOCHEE RIVER AT INGLIS DAM NEAR DUNNELLON, FL
22	2313231	WITHLACOOCHEE RIVER AT INGLIS DAM NEAR DUNNELLON, FL
23	2313250	WITHLACOOCHEE RIVER BYPASS CHANNEL NEAR INGLIS, FL
24	2313272	WITHLACOOCHEE RIVER AT CHAMBERS IS NEAR YAKEETOWN, FL
25	2313274	WITHLACOOCHEE RIVER AT BUNGALOW PASS AT PORT INGLIS, FL
26	2313700	WACCASASSA RIVER NEAR GULF HAMMOCK, FL

Figure 2.4.1-4. USGS Streamflow Gauges in the Withlacoochee and the Waccasassa River Basins

Figure 2.4.1-4 shows six USGS stream gauges near the LNP site, five on the Lower Withlacoochee River and one on the Rainbow River. At some gauges, only gauge height data are available while at other gauges both gauge height and discharge measurements are available. The applicant provided a summary of the data available at these gauges in FSAR Table 2.4.1-201.

The CFBC was conceived as a northern inland waterway between the Gulf of Mexico and northeast Florida in the 1960s. The design depth and width of the canal were 3.7 and 45.7 m (12 and 150 ft), respectively. Due to its adverse environmental and economic impact, construction of the CFBC was stopped in 1971. The CFBC bisected the original course of the Lower Withlacoochee River and severed the connection between Lake Rousseau and the original course. Water is now released from Lake Rousseau through the Inglis Bypass Channel and Spillway into the original course of the Lower Withlacoochee River. Flow through the Inglis Dam only occurs during large floods.

Lake Rousseau is a 1,685-ha (4,163-ac), 9.2-km (5.7-mi) long impoundment on the Withlacoochee River located approximately 17.7 km (11 mi) upstream of the mouth of the river near the city of Inglis. The lake was constructed in 1909 by Florida Power Corporation for power generation. The water level in the lake is controlled by the Inglis Bypass Channel and Spillway, the Inglis Dam, and the Inglis Lock. The operating level is maintained between 7.3 and 8.5 m (24 and 28 ft) NGVD29 with an optimum level at 8.4 m (27.5 ft) NGVD29. Normal discharge of 43.6 m^3/s (1,540 cfs), which is also the maximum discharge capacity of the spillway with a crest elevation of 8.5 m (28 ft) NGVD29, is passed through the Inglis Bypass Channel and Spillway. Flow exceeding this discharge is passed through the Inglis Dam to the CFBC through a short, original course of the Withlacoochee River downstream of the dam.

Inglis Lock is 182.9 m (600 ft) long and 25.6 m (84 ft) wide and was designed as a navigational lock for vessels traveling between Lake Rousseau and the Gulf of Mexico. The lock has not been used since 1999 because its upstream gate is in need of repair. There are currently no plans to repair the gate.

Inglis Dam has a reinforced concrete, two-bay, gated spillway with ogee weirs with a crest elevation of 8.5 m (28 ft) NGVD29. The maximum allowable lake level is 8.5 m (28 ft) NGVD29. Other water-control structures such as the Lake Tsala Apopka Dam, Slush Pond Dam, and Gant Lake Dam exist upstream of Lake Rousseau but do not directly affect the water level in the lake. The Tsala Apopka chain of lakes and the water-control structures are located in central portion of the Withlacoochee River Basin. The system comprises three pools: Hernando, Inverness, and Floral City. The control structures regulate flow between the river and the pools. The Floral City pool is the highest, with a high-water level of 12.7 m (41.8 ft) NGVD29 and a 10-year flood guidance level of 13.2 m (43.4 ft) NGVD29. The 10-year flood guidance levels of the Hernando and Inverness pools are 12.3 and 12.7 m (40.5 and 41.8 ft) NGVD29, respectively. The three pools range in storage capacity from 36,634,409 m^3 to 74,008,908 m^3 (29,700 to 60,000 ac-ft). The operations of the Tsala Apopka system are described by the SWFWMD (2007). The applicant stated that the USACE National Inventory of Dams lists seven dams on Saddle Creek that create settling areas. The seven Saddle Creek settling areas range in storage from 62,908 m^3 (51 ac-ft) for settling area number 6 to 19,452,008 m^3 (6 to 15,770 ac-ft) for settling area number 2. Slush Pond Dam has a storage of 62,908 m^3 (51 ac-ft) and Gant Lake Dam has a storage of 651,278 m^3 (528 ac-ft).

The relatively undeveloped Waccasassa River Basin, which has an approximate area of 2,334 km^2 (901 mi^2), is located in the southern part of the Suwannee River Water Management District. Named drainages in the basin include the Waccasassa River, Jakes Creek, Kelly Creek, Otter Creek, Magee Branch, Wekiva Creek, Cow Creek, Ten Mile Creek, and Spring Run. The basin generally drains southwest towards the Gulf of Mexico and does not have any known water-control structures.

There is no known public water supply from Lake Rousseau or from the Withlacoochee River; the primary source of public water supply is from groundwater near the LNP site.

NRC Staff's Technical Evaluation

The staff reviewed the applicant's responses to the RAIs and determined that the description of the hydrosphere and the interfaces of the proposed units with the hydrosphere are adequately accounted for in site characterization. The staff used publicly available data from USGS, Natural Resources Conservation Service (NRCS), NOAA and its own observations from the site tour to perform its review.

The staff used the Watershed Boundary Dataset available from the Natural Resources Conservation Service (NRCS 2010) to independently confirm the location of the LNP site and the hydrologic setting in its vicinity. Most of the LNP site is located in the Waccasassa River Basin in Florida. Most of the LNP site is located in subbasins named Spring Run and Thousandmile Creek-Halverson Creek Frontal (Figure 2.4.1-5). A small portion of the LNP site is located in the West Lake Rousseau-Cross Florida Barge Canal drainage, which is a subbasin of the Withlacoochee River Basin. Although Spring Run and Thousandmile Creek-Halverson Creek Frontal are subbasins of the Waccasassa River Basin, the streams within these two subbasins drain directly to the Gulf of Mexico (Figure 2.4.1-5). The West Lake Rousseau-Cross Florida Barge Canal drainage, a subbasin of the Withlacoochee River Basin, is hydrologically separate from the Waccasassa River Basin.

Based on its independent review of hydrologic data at and in the vicinity of the LNP site, the staff determined that the applicant has accurately described the hydrologic interfaces for the proposed units at the LNP site.

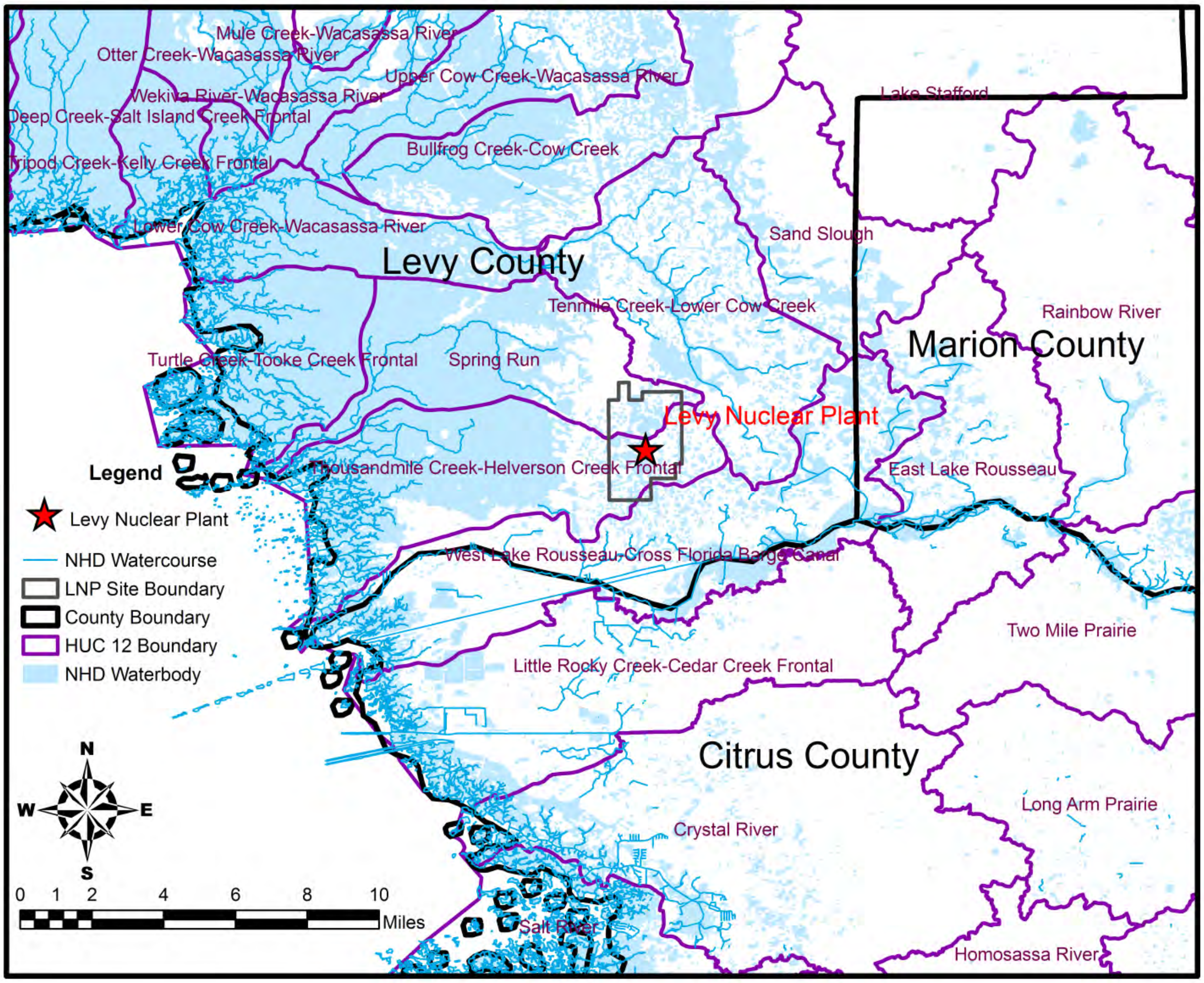


Figure 2.4.1-5. Subwatersheds Near the LNP Site. Waterbodies and watercourses data were obtained from the National Hydrography Dataset.

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 COL FSAR related to this section.

As set forth above, the applicant has presented and substantiated information to establish the site description. The staff has reviewed the information provided and, for the reasons given above, concludes that the applicant has provided sufficient details about the site description for the staff to determine, as documented in Section 2.4.1 of this SER, that the applicant has met the relevant requirements of 10 CFR 52.79(a)(1)(iii) and 10 CFR Part 100 with respect to determining the acceptability of the site. This addresses COL information item 2.4-1. In conclusion, the applicant has provided sufficient information for satisfying 10 CFR Part 52 and 10 CFR Part 100.

2.4.2 Floods

2.4.2.1 *Introduction*

FSAR Section 2.4.2 of the LNP COL application discusses historical flooding at the proposed site or in the region of the site. The information summarizes and identifies individual flood-producing mechanisms, and combinations of flood-producing phenomena, to establish the design-basis flood for SSCs important to safety. The discussion also covers the potential effects of local intense precipitation on SSCs important to safety.

Section 2.4.2 of this SER provides a review of the following specific areas and flood-causing mechanisms: (1) local flooding on the site and drainage design; (2) stream flooding; (3) surges; (4) seiches; (5) tsunami; (6) dam failures; (7) flooding caused by landslides; (8) effects of ice formation on waterbodies; (9) combined event criteria; (10) other site-related evaluation criteria; and (11) additional information requirements prescribed in the “Contents of Application” sections of applicable subparts to 10 CFR Part 52. Flood causing mechanisms listed above are also discussed in detail in subsequent subsections of this SER.

2.4.2.2 *Summary of Application*

This section of the COL FSAR addresses information about site-specific flooding. The applicant addressed the information as follows:

<u>*COL Information Item*</u>

- LNP COL 2.4-2

In addition, this section addresses the following COL-specific information identified in Section 2.4.1.2 of Revision 19 of the DCD.

> Combined License applicants referencing the AP1000 design will address the following site-specific information on historical flooding and potential flooding factors, including the effects of local intense precipitation.
>
> - Probable Maximum Flood on Streams and Rivers – Site-specific information that will be used to determine design basis flooding at the site. This information will include the probable maximum flood 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 of flood level.

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 the Commission regulations for the identification of floods and flood design considerations, and the associated acceptance criteria, are described in Section 2.4.2 of NUREG-0800 (NRC 2007a).

The applicable regulatory requirements for identifying floods are as follows:

- 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 52.79(a)(1)(iii), 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.

The related acceptance criteria are as follows:

- RG 1.59, "Design Basis Floods for Nuclear Power Plants" (NRC 1977a) as supplemented by best current practices

- RG 1.102, "Flood Protection for Nuclear Power Plants" (NRC 1976b).

2.4.2.4 *Technical Evaluation*

The NRC staff reviewed Section 2.4.2 of the LNP 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 site-specific flooding description. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

2.4.2.4.1 Flood History

Information Submitted by the Applicant

The applicant stated that historical measurements of gauge heights and/or discharges are available at five USGS stations near the LNP site. These stations and their records, reported by the applicant in the FSAR, are summarized in Table 2.4.2-1.

Table 2.4.2-1. Historical Flood Measurements Near the LNP Site

Name (USGS ID)	Stage Measurement (Maximum stage on date)*	Discharge Measurement	Comment
Withlacoochee River at Dunnellon, Florida (02313200)	1963–2007 (9.26 m (30.37 ft) NDVD29 on 9/27/2004)		Discharge data not available
Withlacoochee River at Inglis Dam near Dunnellon, Florida (02313230)	1985–2007 (8.54 m (28.03 ft) NGVD29 on 3/27/2005)	1969–2007	
Withlacoochee River below Inglis Dam near Dunnellon, Florida (02313231)	1969–2007 (2.82 m (9.25 ft) NGVD29 on 3/20/1998)		Discharge data not available
Withlacoochee River Bypass Channel near Dunnellon, Florida (02313250)	1971–2007 (8.57 m (28.11 ft) NGVD29 on 1/2/1994)	1970–2007	
Withlacoochee River at Chambers near Yankeetown, Florida (02313272)	2005–2007 (1.36 m (4.47 ft) NAVD88 during high tides on 6/13/2006 and 0.14 m (0.46 ft) NAVD88 during low tides on 3/21/2006)		Discharge data not available

* As noted previously, the staff independently verified that elevations near the LNP site referring to the NGVD29 datum are 0.31 m (1 ft) greater than those referring to the NAVD88 datum.

The applicant stated that the National Weather Service (NWS) Advanced Hydrologic Prediction Service (AHPS) has identified a flood stage of 8.8 m (29 ft), a moderate flood stage of 9.1 m (30 ft) NGVD29, and a major flood stage of 9.4 m (31 ft) all with respect to gauge datum for the USGS station 02313200, Withlacoochee River at Dunnellon, Florida. The applicant stated that during 1963–2007, the major flood stage has not been exceeded at this gauge, the moderate flood stage was exceeded for 22 days during September 27 – October 18, 2004, and the flood stage has been exceeded for 15 of the 44 years of record. Based on historical data, the applicant concluded that flooding at the LNP site is unlikely but lower elevation areas near Lake Rousseau, the Withlacoochee River, and the CFBC may become flooded during periods of high water.

NRC Staff's Technical Evaluation

The information presented in this section describes the NRC staff's review of information and analyses by the applicant and presented in LNP FSAR Section 2.4.2. The NRC staff's independent analysis, where needed for the review, is also included. An accurate description of historical flooding, flooding mechanisms, and combination of these mechanisms and a thorough analysis of the effects of local intense precipitation on the proposed site is needed for the staff to complete its safety review. To understand the process used to determine the design basis flood, the staff issued **RAI 02.04.02-01**, which states:

> To meet the requirements of GDC 2, 10 CFR 52.17, and 10 CFR Part 100, the applicant should include information concerning design basis flooding at the plant site, including consideration of appropriate combinations of individual flooding mechanisms in addition to the most severe effects from individual mechanisms themselves. Please describe the process followed to determine the conceptual models for floods from local intense precipitation, probable maximum flood in the drainage area upstream of the site, surges, seiche, tsunami, seismically induced dam failures, landslides, and ice effects to ensure that the design basis flood is based on the most conservative of plausible conceptual models.

The applicant responded to the staff's RAI 02.04.02-01 in a letter dated July 13, 2009 (ML091950612). The applicant stated that conceptual site models were developed to estimate flooding from local intense precipitation, flooding in streams and rivers, flooding from upstream dam failures, flooding from surges and seiches, flooding from tsunami, flooding from landslides, and flooding from ice effects. The applicant used a runoff coefficient of 1.0, or an equivalent assumption of no precipitation loss to maximize the runoff from the local intense precipitation on the plant area. The applicant assumed that all stormwater conveyance features, including ditches, sewers, and culverts, would be non-functional during the local intense precipitation event. The applicant conceptualized that runoff from the plant area during the local intense precipitation event would be delivered offsite as flow over broad-crested weirs at downstream control points such as peripheral roads. Using this conceptualization, the applicant estimated the backwater profile to determine the maximum water surface elevations at the SSCs important to safety. The applicant described the conceptual models for other flooding mechanisms in the respective FSAR sections.

The staff reviewed the applicant's response to RAI 02.04.02-01 and determined that the applicant postulated a conservative conceptual model of flooding during local intense precipitation because it used no precipitation losses and used downstream controls to estimate backwater effects. The staff determined, therefore, that the applicant has provided sufficient information for the staff's independent review.

An accurate description of the history of flooding in the site area and adjacent region is required for the staff to perform its safety assessment. To analyze the history of flooding at the site, the staff used the information provided by the applicant and supplemented it with publicly available sources of information and field observations from the safety audit.

To review the historical floods near the LNP site, the staff independently obtained peak streamflow data from USGS real-time and historical stream gauges. The location of these gauges is shown in Figure 2.4.2-1.

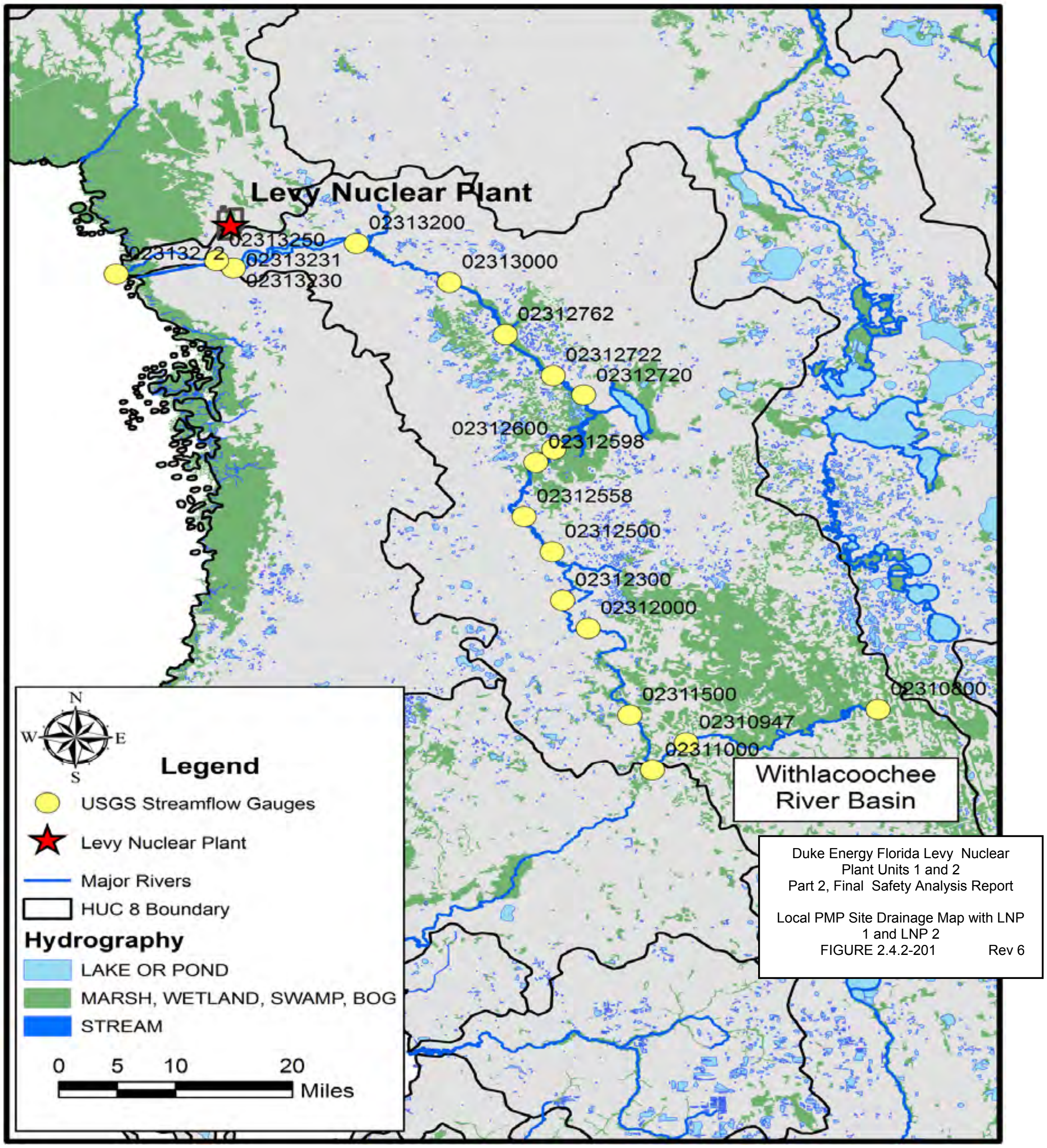


Figure 2.4.2-1. The Withlacoochee River Basin and USGS Streamflow Gauges

These gauges are located in the Withlacoochee River Basin. There are no gauges in the Spring Run and Thousandmile Creek-Halverson Creek Frontal subbasins of the Waccasassa River Basin. The staff reviewed the location of these gauges and determined that the gauges that represent flooding conditions most appropriately near the LNP site are: (1) USGS Gauge Number 02313200, Withlacoochee River at Dunnellon, Florida, (2) USGS Gauge Number 02313230, Withlacoochee River at Inglis Dam near Dunnellon, Florida, (3) USGS Gauge Number 02313231, Withlacoochee River below Inglis Dam near Dunnellon, Florida, and (4) USGS Gauge Number 02313250, Withlacoochee River Bypass Channel near Inglis, Florida. The staff summarized the records and data available at these USGS gauges and is presented in Table 2.4.2-2.

Table 2.4.2-2. Staff-Obtained Historical Flood Records for USGS Streamflow Gauges near the LNP Site

Name (USGS ID)	Stage Measurement (Maximum stage on date)	Peak Discharge Measurement (Maximum discharge on date)	Comment
Withlacoochee River at Dunnellon, Florida (02313200)	Since February 6, 1963 (9.26 m [30.37 ft] NGVD29 on September 27, 2004)		Data available for gauge height only
Withlacoochee River at Inglis Dam near Dunnellon, Florida (02313230)	Since October 1, 1985 (8.62 m [28.28 ft] NGVD29 on June 19, 1982)	Since 1970 Water-Year (171 m^3/s [6,030 cfs] on October 19, 2004)	Maximum stage and maximum discharge occurred on different dates
Withlacoochee River below Inglis Dam near Dunnellon, Florida (02313231)	Since October 1, 1969 (2.82 m [9.25 ft] NGVD29 on March 20, 1998)		Data available for gauge height only
Withlacoochee River Bypass Channel near Dunnellon, Florida (02313250)	Since September 9, 1971 (8.63 m [28.31 ft] NGVD29 on May 19, 1977)	Since 1970 Water-Year (52 m^3/s [1,840 cfs] on October 1, 1987)	Maximum stage and maximum discharge occurred on different dates

The staff concluded, based on available historical flood data at USGS streamflow gauges, that the finished grade elevation of the LNP site would be located approximately 6.1 m (20 ft) above the highest observed floodwater surface elevation in the Withlacoochee River near the site.

The staff also obtained historical gauge height data from NWS AHPS for Withlacoochee River at Dunnellon and Holder. The NWS AHPS website (2011) reported that the historical crests of the Withlacoochee River at Dunnellon show three instances when the flood stage exceeded the major flood stage of 9.4 m (31 ft) above gauge datum: 10.1 m (33 ft) on April 1, 1960, 9.6 m (31.6 ft) on October 12, 1961, and 9.59 m (31.45 ft) on July 17, 1934. The staff found that the NWS AHPS reported Withlacoochee River at Holder exceeding major flood stage of 3.35 m (11 ft) above gauge datum on five occasions: 4.05 m (13.28 ft) on April 5, 1960, 3.67 m (12.05 ft) on October 10, 1960, 3.54 m (11.63 ft) on July 8, 1934, 3.43 m (11.25 ft) on October 13, 2004, and 3.40 m (11.17 ft) on September 26, 1933. The NWS AHPS website does not report data for the other USGS gauges shown in Table 2.4.2-2. Because the Withlacoochee River at Dunnellon is the nearer location where NWS AHPS data is available, the staff used this location in its independent assessment. Based on the data reported by the NWS AHPS, the staff determined that the Withlacoochee River does occasionally exceed major flood stage. However, the highest reported stage for the river at Dunnellon is approximately 5.5 m (18 ft) below the proposed grade elevation of the LNP site. Based on its independent assessment, the staff determined that the LNP site has not been flooded by the Withlacoochee River during the period stream discharge and stage data have been recorded.

2.4.2.4.2 Flood Design Considerations

Information Submitted by the Applicant

The applicant stated that safety-related SSCs at the LNP site are protected against floods and flood waves caused by probable maximum events. Seismic Category I SSCs within the plant are designed for flooding due to natural phenomena and the basemat and exterior walls of these structures are designed for upward and lateral pressures from probable maximum flood (PMF) and high groundwater levels. The applicant has also stated that because the plant will be sited at a higher finished grade, no dynamic water forces will occur and that the finished grade will be adequately sloped to prevent dynamic forces associated with the probable maximum precipitation (PMP).

The applicant estimated the design basis flood elevation at the LNP site to be 15.17 m (49.78 ft) NAVD88 and it results from a probable maximum storm surge combined with wind-induced setup.

NRC Staff's Technical Evaluation

An accurate description of flooding mechanisms and combinations of these is required for the NRC staff to perform its safety assessment.

The NRC staff reviewed the applicant's responses to the RAIs to determine whether the process followed by the applicant to determine the design-basis flood is adequate. The NRC staff also used observations from its safety audit site tour and other independent data sources in its safety review. To analyze the effects of hydrodynamic forces on SSCs, the staff issued **RAI 02.04.02-02**, which states:

> To meet the requirements of GDC 2, 10 CFR 52.17 and 10 CFR Part 100, the applicant should include a determination of the capacity of site drainage facilities. Section 2.4.2.2 of the FSAR states "No dynamic water forces associated with high water levels will occur because of a higher finished plant grade. The dynamic forces associated with the probable maximum precipitation (PMP) are not factors in the analysis or design because the finished grade will be

> adequately sloped." Please clarify how sloping of the grade excludes consideration of dynamic forces in the analysis and design of safety-related SSCs during the local PMF event or provide an analysis that shows safety-related SSCs would be safe under the static and dynamic effects of the local PMF.

The applicant responded to the staff's RAI 02.04.02-02 in a letter dated July 13, 2009 (ML091950612). The applicant stated that the site grading would be performed such that the floor elevations of SSCs would be above the highest grade elevation. The applicant stated that the plant grade would be sloped away from the SSCs such that runoff would flow away from them. The applicant performed an analysis to estimate the water surface elevation during the local intense precipitation event and reported that the maximum water surface elevation including backwater effects would be less than the nominal plant grade floor elevation of 15.5 m (51 ft) NAVD88.

The staff reviewed the applicant's response and calculations performed to account for the backwater effects during the local intense precipitation event. As stated above, the applicant used a runoff coefficient of 1.0 for estimating the runoff from the local intense precipitation event. A runoff coefficient of 1.0 indicates that no infiltration or evapotranspiration losses were allowed and therefore, all of the precipitation contributed to runoff generation. This assumption resulted in maximization of runoff during the local intense precipitation event.

To perform the flooding analysis, the applicant divided the main plant area into seven drainage zones. The applicant estimated the time of concentration conservatively for each zone using Kirpich's Formula (Chow 1964). The applicant used the time of concentration to estimate the rainfall intensity, which is a parameter in the Rational Formula for peak discharge. The applicant represented the flow dynamics within the zones using a set of cross sections in the USACE HEC-RAS software. HEC-RAS was set up to simulate a steady-state backwater profile with the flow depth at the downstream boundary estimated using the broad-crested weir equation with the discharge set to the peak discharge estimated from the Rational Formula for the zone. The discharges at each of the cross sections were estimated by prorating the peak discharge for the zone by the ratio of contributing area upstream of the respective cross section to the total surface area of the zone. The staff determined that the applicant's approach is appropriate for estimation of water surface elevations near the safety-related SSCs because it considers the effects of the backwater flow profile upstream of the broad-crested weir that acts to control the depth of flow. Flow depths estimated from a steady-state hydraulic routing calculations envelop those from an unsteady hydraulic routing calculation if the peak discharges used in both simulations are the same. Therefore, the staff determined that the steady-state backwater profile would result in a conservative estimate of the greatest flow depth on the plant area during a transient local intense precipitation event.

The applicant used Manning's roughness coefficient values of 0.035 for peripheral areas and 0.025 for powerblock areas. The staff reviewed the Manning's roughness coefficients used by the applicant to determine whether they are appropriately conservative. The surface of the powerblock area would consist of concrete, asphalt pavement, or compacted gravel and grass. Chow (1959) recommends Manning's roughness coefficient ranges of 0.023 to 0.036 for gravel surfaces with dry rubble sides, a range of 0.013 to 0.016 for asphalt surface, and a range of 0.016 to 0.025 for straight and uniform earthen areas. The staff concluded that the applicant has used Manning's roughness coefficient values that correspond to the higher end of the recommended ranges. Higher Manning's roughness coefficient values result in higher water surface elevations. Therefore, the staff concluded that the applicant has conservatively estimated the floodwater surface elevation near the safety-related SSCs during the local intense precipitation event.

2.4.2.4.3 Effects of Local Intense Precipitation

Information Submitted by the Applicant

The applicant has also stated that water would not pond on safety-related SSCs of the LNP Units 1 and 2 because the roofs do not have drains or parapets and are sloped so rainfall is directed to gutters located along the edge of the roofs. The site drainage system is designed to drain runoff from a 50-year precipitation event to catch basins, underground pipes, or to open ditches. The drainage system is assumed to be non-functional during a local PMP event and the runoff from this event would be drained by overland flow on the ground surface away from safety-related SSCs to onsite retention ponds and eventually to the Lower Withlacoochee River and to the Gulf of Mexico.

Grading and drainage for the LNP site is shown in Figure 2.4.2-2. The LNP site is subdivided into seven drainage zones, A through G.

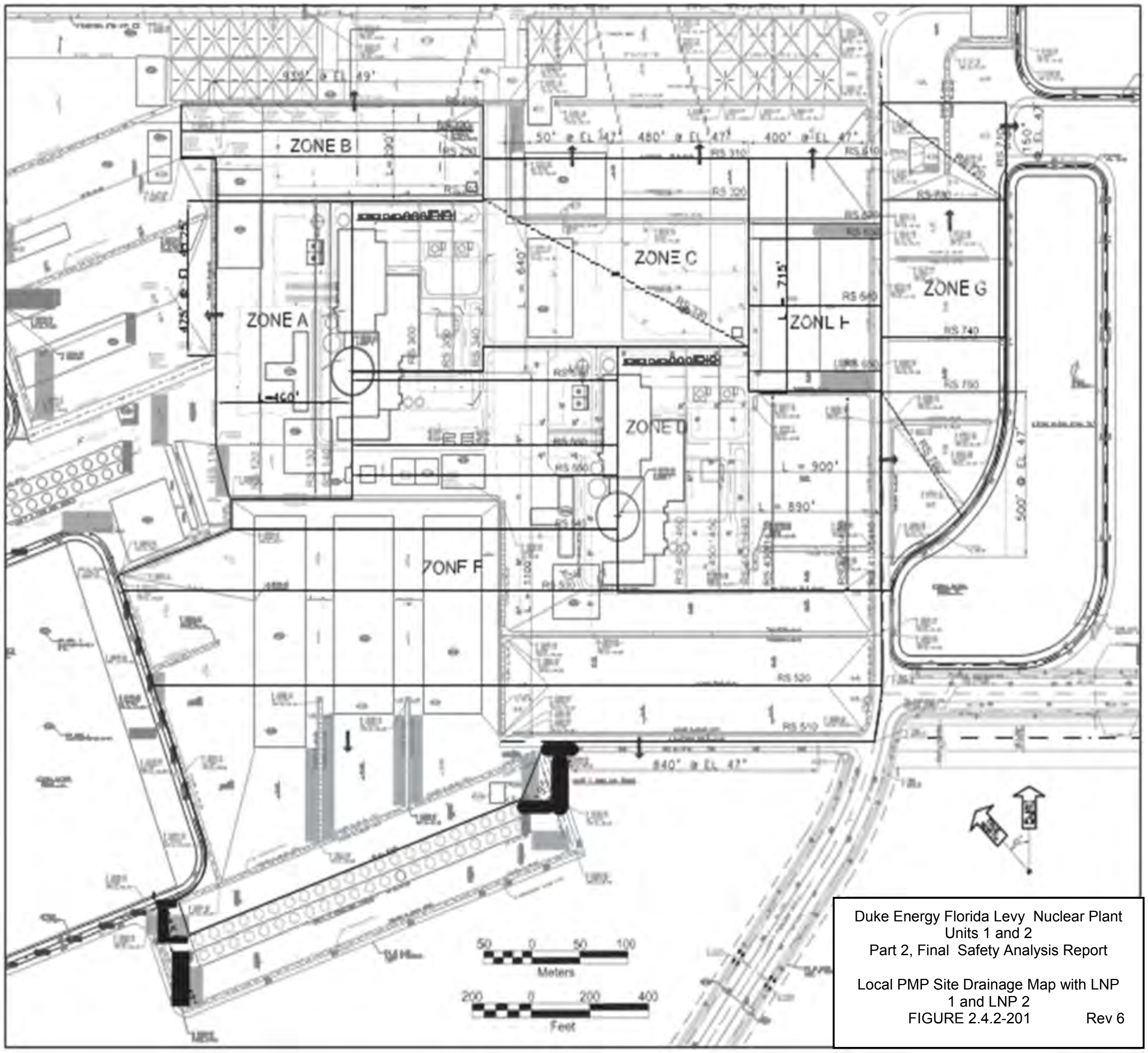


Figure 2.4.2-2. Local PMP Site Drainage Map with LNP 1 and LNP 2

The applicant determined the local PMP values for the LNP site using the procedure described in Hydrometeorological Report (HMR) No. 52 (Hansen et al. 1982). Local PMP values were taken as the 2.6-km^2 (1-mi^2) PMP values for durations ranging from 5 minutes to 24 hours. Table 2.4.2-3 shows the local PMP values estimated by the applicant.

Table 2.4.2-3. The Applicant-Estimated Probable Maximum Precipitation for the 2.6-km^2 (1-mi^2) Area

Duration		
Minutes	Hours	Precipitation (cm [in.])
5	0.08	15.95 (6.28)
15	0.25	24.92 (9.81)
30	0.5	36.37 (14.32)
60	1	49.80 (19.61)
360	6	94.51 (37.21)
720	12	114.91 (45.24)
1440	24	133.15 (52.42)

Runoff during the local PMP event was estimated using the rational method with the runoff coefficient set to 1.0. There are no safety-related facilities in drainage Zone G. The water levels for each of the other six drainage zones were estimated assuming that the peak runoff discharging out of the zone would behave as a discharge over a broad-crested weir. The water surface elevations estimated by the applicant for each of the other six zones are listed in Table 2.4.2-4.

Table 2.4.2-4. Maximum Water Surface Elevations on the LNP Site Estimated by the Applicant

Drainage Zone	Maximum Water Surface Elevation (m [ft] NAVD88)	Maximum Flow Velocity (m/s [ft/s])
A	15.3 (50.3)	0.4 (1.3)
B	15.3 (50.1)	0.6 (2.1)
C	15.5 (50.7)	1.1 (3.7)
D	15.4 (50.5)	0.6 (1.9)
E	15.4 (50.4)	0.8 (2.7)
F	15.4 (50.5)	1.2 (3.8)
D+G	15.4 (50.5)	1.0 (3.2)

In the FSAR, the applicant stated that roads in Zones A through F that may fall in the path of the overland flow during the local PMP event would be lowered to preclude safety-related facilities from being affected.

Based on the historical rainfall measured at the Ocala, Florida NWS Cooperative Station No. 086414, the applicant reported an annual mean precipitation of 126.19 cm (49.68 in.), a monthly mean precipitation range of 6.27 to 18.29 cm (2.47 to 7.20 in.), a highest monthly precipitation of 41.58 cm (16.37 in.) all recorded in April 1982, and a maximum daily precipitation of 29.77 cm (11.72 in.) recorded on April 8, 1982. The applicant stated that the LNP site is not expected to support long-term accumulation of ice and snow, and therefore, did not consider these as potential flooding mechanisms.

NRC Staff's Technical Evaluation

An accurate description of the method used to estimate local intense precipitation and the values obtained by the applicant is needed for the NRC staff to perform its safety assessment.

The NRC staff reviewed the applicant's responses to RAIs 2.4.2-1, 2.4.2-2, 2.4.2-3, and 2.4.2-4, which are discussed further in this section of the SER, to determine whether the effects of local intense precipitation considered by the applicant are adequate. The NRC staff also used observations from its safety audit site tour and other independent data sources in its safety review.

The staff independently estimated the local intense precipitation as the 1-hour, 2.6-square-km (1-square-mile) PMP from HMR 52 (Hansen et al. 1982). The staff-estimated local intense precipitation values are listed in Table 2.4.2-5.

Table 2.4.2-5. The Staff-Estimated Local Intense Precipitation at the LNP Site

Duration	Multiplier to 1-hour Precipitation Depth	Depth of Precipitation (cm [in.])
5 min	0.32 (HMR 52, Figure 36)	15.7 (6.2)
15 min	0.50 (HMR 52, Figure 37)	24.6 (9.7)
30 min	0.73 (HMR 52, Figure 38)	36.1 (14.2)
1 hour	1.0	49.3 (19.4)

The staff compared the applicant's estimate of the local intense precipitation with its own independent estimate. The applicant's estimates for the local intense precipitation are 1 percent higher than the staff's. The staff concluded that the applicant has appropriately and conservatively estimated the local intense precipitation at the LNP site. To obtain clarification regarding the site grade elevation and to determine the safety of SSCs, the staff issued **RAI 02.04.02-03**, which states:

> To meet the requirements of GDC 2, 10 CFR 52.17 and 10 CFR Part 100, the applicant should include a complete description of all spatial and temporal datasets used in support of its conclusions regarding safety of the plant. Data and descriptions should be sufficiently detailed to allow the staff to review the applicant's conclusions regarding the safety of the plant and to determine ~~of~~ the design bases of safety related SSCs. Please clarify if the stated site grade elevation of 15.5 m (51 ft) NGVD29 is subject to change.

The applicant responded to the staff's RAI 02.04.02-03 in a letter dated July 13, 2009 (ML091950612). The applicant stated that the nominal plant grade floor elevation of SSCs at the LNP site would be 15.5 m (51 ft) NAVD88 and is not subject to change. The staff used the nominal plant grade floor elevation of 15.5 m (51 ft) NAVD88 as the finished floor elevation of safety-related SSCs at the LNP site for all safety determinations in the hydrologic engineering sections of this report.

To determine the appropriateness of the methods used to estimate flood discharges and elevations during the local intense precipitation event, the staff issued **RAI 02.04.02-04**, which states:

> To meet the requirements of GDC 2, 10 CFR 52.17, and 10 CFR Part 100, please clarify (1) the description of the methodology used to estimate the times of concentration for each drainage zone, (2) the locations and characteristics of the broad-crested weirs, and (3) the estimated backwater profile from the broad-crested weirs to the safety-related SSCs.

The applicant responded to the staff's RAI 02.04.02-04 in a letter dated July 13, 2009 (ML091950612). The applicant stated that the Kirpich Formula was used to estimate the time of concentration for each drainage zone. The Kirpich Formula uses the length of the drainage area measured along the flow and the average slope of the drainage area and is frequently used in design of urban drainage systems (Chow 1964). The staff concluded therefore, that the applicant's approach is appropriate.

The applicant described the location and characteristics of the broad-crested weirs used in the estimation of the floodwater surface elevation during the local intense precipitation event. The applicant stated that the broad-crested weirs are typically located at roads, tops of embankments, crests of site grades, or where the slope of the grade changes significantly. The applicant used the broad-crested weir equation (USACE 1987) to estimate the discharge over the weirs. The broad-crested weir equation uses a coefficient of discharge (USACE 1987). The staff reviewed the method described by USACE (1987) and the applicant's calculation package and determined that the applicant appropriately selected the discharge coefficient for the LNP site where the ratio of water depth over the broad-crested weir to the weir breadth is expected to be smaller than 0.5.

The applicant described its procedure for estimation of the backwater profiles for each of the seven runoff zones. Table 2.4.2-6 lists the characteristics of the runoff zones and the estimated flood properties during the local intense precipitation event.

Table 2.4.2-6. Characteristics of the Runoff Zones and Estimated Flood Properties

Runoff Zone	Area (ha [ac])	Peak Discharge (m^3/s [cfs])	Maximum Floodwater Surface Elevation (m [ft] NAVD88)	Maximum Flow Velocity (m/s [ft/s])
A	3.8 (9.4)	13.2 (465)	15.3 (50.3)	0.4 (1.3)
B	2.6 (6.5)	14.1 (499)	15.3 (50.1)	0.6 (2.1)
C	6.9 (17.0)	27.1 (957)	15.5 (50.7)	1.1 (3.7)
D	5.6 (13.9)	14.9 (525)	15.4 (50.5)	0.6 (1.9)
E	22.0 (54.3)	60.0 (2,120)	15.4 (50.4)	0.8 (2.7)
F	3.0 (7.3)	10.2 (361)	15.4 (50.5)	1.2 (3.8)
D+G	10.7 (26.4)	32.3 (1140)	15.4 (50.5)	1.0 (3.2)

Based on the review of the applicant's responses to the staff's RAIs, review of the applicant's calculation packages, and the staff's independent estimation of the local intense precipitation at the LNP site, the staff concluded that the applicant has adequately and conservatively estimated the effects of the local intense precipitation at the LNP site because (1) the local intense precipitation was conservatively estimated, (2) no precipitation losses were allowed, (3) an appropriate simulation model (HEC-RAS) was used, and (4) values used for Manning's roughness coefficients were conservative. The staff agrees with the applicant that the floodwater surface elevations in the powerblock area near the safety-related SSCs would not exceed the nominal plant grade floor elevation of 15.5 m (51 ft) NAVD88.

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 COL FSAR related to this section.

As set forth above, the applicant has presented and substantiated information to establish the site description. The staff has reviewed the information provided and, for the reasons given above, concludes that the applicant has provided sufficient details about the site description for the staff to determine, as documented in Section 2.4.2 of this SER, that the applicant has met the relevant requirements of 10 CFR 52.79(a)(1)(iii) and 10 CFR Part 100 with respect to determining the acceptability of the site. This addresses COL Information Item 2.4-2.

2.4.3 Probable Maximum Flood (PMF) on Streams and Rivers

2.4.3.1 *Introduction*

FSAR Section 2.4.3 describes the hydrological 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 of this SER provides a review of the following specific areas: (1) design basis for flooding in streams and rivers, (2) design basis for site drainage, (3) consideration of other site-related evaluation criteria, and (4) any additional information requirements prescribed in the "Contents of Application" sections of the applicable subparts to 10 CFR Part 52.

2.4.3.2 *Summary of Application*

This section of the COL FSAR addresses the site-specific information about PMFs on streams and rivers. The applicant addressed the information as follows:

AP1000 COL Information Item

- LNP COL 2.4-2

This section addresses the following COL-specific information identified in Section 2.4.1.2 of Revision 19 of the AP1000 DCD.

> The COL applicants referencing the AP1000 design will address the following site-specific information on historical flooding and potential flooding factors, including the effects of local intense precipitation:
>
> - Probable Maximum Flood on Streams and Rivers – Site-specific information that will be used to determine 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.
>
> 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 the Commission regulations for the identification of floods and flood design considerations, and the associated acceptance criteria, are described in Section 2.4.3 of NUREG-0800 (NRC 2007a).

The applicable regulatory requirements for identifying probable maximum flooding on streams and rivers are as follows:

- 10 CFR Part 100, 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), as it 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

- 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.

The related acceptance criteria are as follows:

- RG 1.27, "Ultimate Heat Sink for Nuclear Power Plants" (NRC 1976a).
- RG 1.29, "Seismic Design Classification" (NRC 2007b).
- RG 1.59, "Design Basis Floods for Nuclear Power Plants" (NRC 1977a) as supplemented by best current practices.
- RG 1.102, "Flood Protection for Nuclear Power Plants" (NRC 1976b).
- RG 1.206 "Combined License Applications for Nuclear Power Plants (LWR Edition)" (NRC 2007c).

2.4.3.4 *Technical Evaluation*

The NRC staff reviewed Section 2.4.3 of the LNP 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 site-specific PMF on streams and rivers. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

An accurate description of the assessment of the PMF level is needed for the staff to perform its safety assessment. To understand the process followed in the analysis of in-stream flooding, the staff issued **RAI 02.04.03-01**, which states:

> To meet the requirements of GDC 2, 10 CFR 52.17, and 10 CFR Part 100, estimates of the following characteristics are needed, and should be based on conservative assumptions of hydrometeorologic characteristics in the drainage area: (a) the area of the watershed used to estimate flooding in streams and rivers, (b) the total depth of PMP and the PMP hyetograph, (c) the maximum PMF water surface elevation in streams and rivers with coincident wind-waves, and (d) hydraulic characteristics that describe dynamic effects of PMF on SSCs important to safety. Please describe the process followed to determine the conceptual models for floods in streams and rivers and in site drainage system to ensure that the design basis flood is based on the most conservative of plausible conceptual models.

The applicant responded to the staff's RAI 02.04.03-01 in a letter dated June 23, 2009 (ML091760626). The applicant stated that the LNP safety-related SSCs would be located entirely in the Waccasassa River Basin and would also be located away from nearby waterbodies. The applicant also stated that because there are no named streams on the LNP site and because there are no known water-control structures in the Waccasassa River Basin, safety-related SSCs of the LNP units would not be affected by flooding in the Waccasassa River Basin. The runoff from the LNP site drains to the southwest towards the Lower Withlacoochee River and the Gulf of Mexico. The Withlacoochee River and Lake Rousseau are located approximately 4.8 km (3 mi) south of the LNP site and are located in the Withlacoochee River Basin, which is hydrologically separated from the Waccasassa River Basin.

The applicant stated that to determine the design basis flood, it used guidance provided by NRC RGs 1.206 and 1.59 and American National Standards Institute (ANSI)/American Nuclear Society (ANS)-2.8-1992. The applicant considered the Withlacoochee River Basin upstream of the Inglis Dam as the drainage area for determination of the PMF. The Withlacoochee River Basin above Inglis Dam was divided into 18 subbasins. The applicant estimated the PMP over the basin using the procedures described in HMRs 51 and 52 and ANSI/ANS-2.8-1992. The applicant used a PMP storm lasting 9 days; an antecedent storm, with 40 percent of the estimated PMP depths, was used during the first 3 days; the middle 3 days were dry (no precipitation); and the full PMP storm occurred during last 3 days.

The applicant described its approach for determining the PMF in the Withlacoochee River Basin to determine whether the LNP site may be affected by it. The drainage area of the Withlacoochee River Basin is approximately 5,232 km^2 (2,020 mi^2). The applicant estimated the PMP over the Withlacoochee River Basin for determination of the PMF. The PMF water surface elevation in Lake Rousseau was determined to be 9.1 m (29.7 ft) NAVD88 and the plant grade floor elevation of LNP SSCs would be at 15.5 m (51 ft) NAVD88. The applicant concluded that there is a substantial margin, 6.5 m (21.3 ft), between the plant grade floor elevation of LNP SSCs and the maximum PMF water surface elevation in Lake Rousseau.

The applicant used unit hydrographs to determine the runoff from the PMP storm for each subbasin of the Withlacoochee River Basin above Inglis Dam. The applicant used no initial loss. The applicant used a constant loss rate during the PMP storm. The runoff hydrograph from each subbasin was routed using the Muskingum routing method in the stream reaches to determine the inflow hydrograph to Lake Rousseau. The inflow to Lake Rousseau was routed through the lake using its stage-storage-discharge relationship and characteristics of the outlet works.

The staff reviewed the applicant's response to RAI 02.04.03-01 and determined that the applicant has provided sufficient information regarding the conceptual models used in the FSAR analyses. The staff agrees with the applicant that there are no streams or rivers of sufficient size in the Spring Run and Thousandmile Creek-Halverson Creek Frontal subbasins of the Waccasassa River Basin to pose a flooding hazard to SSCs at the LNP site. The overland flow in these Frontal subbasins resulting from the local intense precipitation would flow generally southwest. Because the existing grade elevation at the proposed location of the LNP units' powerblock area would be raised, the staff concluded that the floodwater surface elevation produced by the local intense precipitation at the LNP site, presented by the applicant in FSAR

Section 2.4.2 is appropriate. The staff also agrees with the applicant that the most conservative scenario for flooding in streams and rivers that may pose a hazard at the LNP site would occur from a PMF in the adjoining Withlacoochee River Basin. Therefore the staff concluded that the applicant has correctly and conservatively identified the alternative conceptual models for flooding in river and streams near the LNP site.

2.4.3.4.1 Probable Maximum Precipitation

Information Submitted by the Applicant

The applicant estimated the generalized cumulative PMP depths for different areas and durations from HMR 51 (Schreiner and Riedel 1978). The drainage area of the Withlacoochee River Basin upstream of the Inglis Dam was estimated to be 5,232 km^2 (2,020 mi^2). From the cumulative PMP depths for various area sizes, the applicant estimated the 6-hour incremental PMP depths.

The preferred orientation of the PMP isohyetal pattern from HMR 52 (Hansen et al. 1982) is 205º. The applicant estimated that the PMP isohyetal pattern that produced the maximum volume of precipitation within the Withlacoochee River Basin was 150º (Figure 2.4.3-1 [adapted from FSAR Rev 0 Figure 2.4.3-205]). Because the difference in orientation between the preferred and the maximum-volume orientation directions exceeds 40º, the applicant adjusted the incremental PMP depths, which resulted in a small decrease in the unadjusted incremental values.

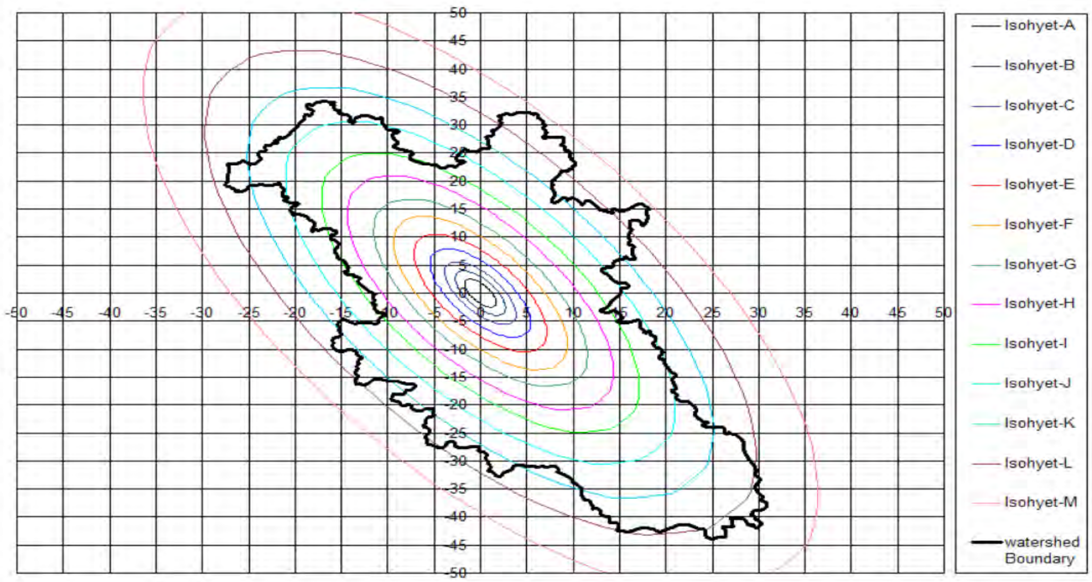
The applicant estimated the values of the isohyets corresponding to the maximum precipitation volume within the Withlacoochee River Basin for the three 6-hour durations with the highest incremental precipitation using the procedure described in HMR 52 (Hansen et al. 1982). The PMP spatial pattern size that maximized the precipitation in the basin was determined to be 3,885 km^2, (1,500 mi^2). Based on this PMP isohyetal pattern, the applicant estimated the basin-average incremental precipitation depths for each of the twelve 6-hour durations. Table 2.4.3-1 lists the 72-hour basin-average PMP for the Withlacoochee River Basin.

The applicant developed the 216-hour, or 9-day design storm for the Withlacoochee River Basin using a 72-hour antecedent storm at 40 percent of the PMP depths shown in Table 2.4.3-1, followed by a 72-hour period of no rain, and the last 72-hour period with precipitation values rearranged from those shown in the last column of Table 2.4.3-1 (100 percent PMP).

NRC Staff's Technical Evaluation

The staff reviewed the applicant's analysis for the estimation of PMP in the Withlacoochee River Basin above Inglis Dam. The staff independently estimated the PMP following the procedures described in HMRs 51 (Schreiner and Riedel 1978) and 52 (Hansen et al. 1982) to verify the applicant's PMP estimates. The staff-estimated PMP depths agree with the applicant's estimates. The staff concluded, therefore, that the applicant has correctly and conservatively estimated the PMP in the Withlacoochee River Basin above Inglis Dam.

LNP COL 2.4-2



Progress Energy Florida
Levy Nuclear Plant
Units 1 and 2
Part 2, Final Safety Analysis Report

Orientation of the Storm Pattern for Maximum Precipitation Volume

FIGURE 2.4.3-205 Rev 0

Figure 2.4.3-1. Spatial Pattern of PMP Storm over the Withlacoochee River Basin

Table 2.4.3-1. The 72-hour Basin-Average PMP for the Withlacoochee River Basin Estimated by the Applicant

Six-hour Duration	Time Since Beginning of the PMP Storm (hr)	Cumulative PMP Depth (cm [in.])	Incremental PMP Depth (cm [in.])
1	6	36.12 (14.22)	36.12 (14.22)
2	12	52.86 (20.81)	16.74 (6.59)
3	18	62.61 (24.65)	9.75 (3.84)
4	24	69.22 (27.25)	6.60 (2.60)
5	30	74.09 (29.17)	4.88 (1.92)
6	36	77.93 (30.68)	3.81 (1.50)
7	42	81.00 (31.89)	3.10 (1.22)
8	48	83.59 (32.91)	2.59 (1.02)
9	54	85.80 (33.78)	2.21 (0.87)
10	60	87.70 (34.53)	1.91 (0.75)
11	66	89.36 (35.18)	1.68 (0.66)
12	72	90.86 (35.77)	1.47 (0.58)

2.4.3.4.2 Precipitation Losses

Information Submitted by the Applicant

The applicant estimated the initial and constant loss rates, which are used by the HEC-HMS computer model and are based on the recommendations of the Federal Energy Regulatory Commission (FERC). The applicant assumed that the entire Withlacoochee River Basin would have saturated soils at the start of the PMP storm, that there would be no initial loss, and that the constant loss during the PMP storm would occur at the minimum rate. The applicant used soils data for the Withlacoochee River Basin available from the SWFWMD to estimate the soil hydrologic groups for each of the subbasins. U.S. Department of Agriculture (USDA) NRCS recommendations (NRCS 1986) for minimum infiltration rates were used for each soil hydrologic group to estimate area-weighted average for each subbasin.

NRC Staff's Technical Evaluation

The staff reviewed the loss rates used by the applicant in its PMF estimation. The staff determined, using a review of the applicant's calculations, that no initial loss was applied to the

PMP storm. The assumption of no initial loss is conservative because it maximizes runoff. However, the applicant used a constant loss rate for the duration of the PMP storm under consideration. The constant loss rate varies, depending on soil type in different parts of the Withlacoochee River Basin. The loss rates ranged from 0.13 to 0.74 cm/h (0.05 to 0.29 in/h). During a PMP storm, especially when an antecedent storm, 40 percent of the PMP occurs prior to the full PMP storm, the soils in the basin would be close to saturation and therefore would only support minimal continuing loss rates. The staff reviewed the applicant's method of estimating the constant loss rate based on spatial distribution of soils in the subbasins. The staff agrees that the applicant's approach is reasonable and conservative because it accounts for subbasin-specific conditions and uses minimum infiltration rates for the different hydrologic soil groups, respectively.

2.4.3.4.3 Runoff and Stream Course Models

Information Submitted by the Applicant

The applicant subdivided the Withlacoochee River Basin into 18 subbasins. Lake Rousseau was assumed to be the 19th subbasin.

Runoff from the subbasins was estimated using a unit hydrograph approach based on Snyder's synthetic unit graphs. Some of the parameters for the Snyder's unit hydrograph were obtained from subbasin geometry; these include the flow path length from outlet to the hydraulically farthest point L and the length of flow path from outlet to centroid of the subbasin L_c. Other parameters were obtained from literature and these include the lag coefficient C_t and the peaking coefficient C_p.

The mean monthly discharge in the Withlacoochee River at USGS gauge 02313000 was used as the baseflow. Muskingum routing was used for streams. The applicant used a trial-and-error procedure to estimate the parameters of the Muskingum routing method. First, the applicant obtained an estimate of 10-, 25-, 50-, and 100-year return period flood discharges at USGS gauge 02313000 using a Log-Pearson Type III distribution subsequently adjusted for the difference in drainage areas at USGS gauge 02313000 and that for the whole Withlacoochee River Basin. The applicant estimated a precipitation-discharge relationship using 24-hour rainfall data for the same return periods. The applicant used the precipitation-discharge relationship to estimate the 500-year and the standard project rainfall amounts. The applicant applied the HEC-HMS model to reproduce the 10-year, 25-year, 50-year, 100-year, 500-year, and the standard project floods using previously estimated rainfall rates and by varying the Muskingum routing parameters.

The applicant used Lake Rousseau bathymetry data from a commercial source and the USGS digital terrain data to develop stage-storage curve for the lake. The applicant obtained the stage-discharge relationships for the Inglis Dam and the Inglis Lock from the State of Florida Environmental Protection Agency. The low-lying area around Inglis Dam was considered to act as an ogee spillway.

<u>NRC Staff's Technical Evaluation</u>

The staff reviewed the methodology adopted by the applicant in the development of the stream course model. The Withlacoochee River Basin is generally flat and has a few storage areas within the basin. The applicant ignored the storage and detention capacity of these storage areas in the hydrologic model used to estimate the PMF. Ignoring the storage and detention capacity would lead to higher peak discharges and quicker runoff response within the basin because precipitation excess would not be retained or detained by these storage areas. The staff determined that the applicant has adequately presented delineations of the subbasins and the stream network within the Withlacoochee River Basin above the Inglis Dam. To obtain a clear understanding of the applicant's process to determine the design-basis flood using combinations of events, the staff issued **RAI 02.04.03-02**, which states:

> To meet the requirements of GDC 2, 10 CFR 52.17, and 10 CFR Part 100, the applicant should include information concerning design basis flooding at the plant site, including consideration of appropriate combinations of individual flooding mechanisms in addition to the most severe effects from individual mechanisms themselves. Please clarify the combined events criterion used to identify the design basis flood at the LNP site and to explicitly state the value of the design basis flood in the FSAR including a description of any adjustment made for long-term sea level rise.

The applicant responded to staff's RAI 02.04.03-02 in a letter dated June 23, 2009 (ML091760626). The applicant stated that various flood scenarios involving Lake Rousseau, the Withlacoochee River, the CFBC, and the Gulf of Mexico were considered. The applicant stated that various individual flooding mechanisms as well as combinations of these, as described in ANSI/ANS-2.8-1992 were considered. The individual flooding events considered included precipitation- and snowmelt-induced floods, failures of dams and other water-control structures, landslides, storm surges, seiches, wind-wave action, ice jams, channel changes and blockages, tsunami, volcanic eruptions, and glaciers. Of these scenarios, the applicant stated that flooding from snowmelt, landslides, ice jams, volcanic eruptions, and glaciers were not considered because these events are unlikely at and near the LNP site.

The applicant stated that the combined events considered for estimation of design basis flood consisted of wind influence, seasonal compatibility, storm optimization, and reservoirs. The applicant stated that wind influence was not explicitly considered during the PMF analysis because the LNP site is located approximately 3 mi from Lake Rousseau. The applicant also did not consider seasonality in the PMF analysis but used an estimate of worst-case flood conditions. The applicant stated that the Withlacoochee River meanders through a broad, flat plain and the river basin contains several swamplands, marshes, ponds, and shallow lakes. The applicant stated that it did not consider any reservoirs or waterbodies upstream of Lake Rousseau because floodwaters in the basin would spread into marshlands and lowlands adjacent to the river channel.

The applicant stated that the design basis flood elevation for the LNP safety-related SSCs results from the storm surge caused by a probable maximum hurricane (PMH) in combination with 10 percent exceedance tides and wind-effects.

The applicant stated that it estimated the long-term sea level rise near the LNP site using data from the tidal gauge located at Cedar Key, Florida. The applicant stated that the upper 95 percent confidence bound of sea level rise at the Cedar Key, Florida, is 1.99 mm/yr (0.08 in/yr), which would result in a 60-year rise of approximately 0.1 m (0.4 ft).

The staff reviewed the applicant's response to RAI 02.04.03-02 and concluded that the applicant has provided sufficient information regarding the design basis floodwater surface elevation at the LNP site. However, in order to determine whether the applicant followed a clear, consistent, and conservative approach in characterizing the hydrometeorological and hydrological parameters, the staff issued **RAI 02.04.03-03**, which states:

> To meet the requirements of GDC 2, 10 CFR 52.17, and 10 CFR Part 100, estimates of the following characteristics are needed, and should be based on conservative assumptions of hydrometeorologic characteristics in the drainage area: (a) the area of the watershed used to estimate flooding in streams and rivers, (b) the total depth of PMP and the PMP hyetograph, (c) the maximum PMF water surface elevation in streams and rivers with coincident wind-waves, and (d) hydraulic characteristics that describe dynamic effects of PMF on SSCs important to safety. Please justify (1) the use of unit hydrograph method for estimating the runoff from precipitation falling on the surface of Lake Rousseau and (2) the appropriateness of Snyder's unit hydrograph under PMP conditions given the assumption of linearity in the unit hydrograph approach of runoff generation.

The applicant responded to the staff's RAI 02.04.03-03 in a letter dated June 23, 2009 (ML091760626). The applicant provided a justification for the use of a unit hydrograph for estimation of runoff from the surface of Lake Rousseau during the PMP event. The applicant presented the assumption behind the unit hydrograph theory. The applicant stated that the use of unit hydrograph theory is best suited for estimation of runoff from the surface of a lake because the assumption of the theory would be minimal. The applicant also suggested that because several unit hydrograph methods, such as the Single-Linear Reservoir method and the Nash method were conceptualized using a reservoir, the unit hydrograph theory should be applicable for runoff estimation from their surfaces.

The staff disagrees with this approach. The unit hydrograph (UH) theory is used to describe the time distribution of surface runoff at the outlet produced by a constant and uniform rainfall excess event over a watershed. The time delay and attenuation in discharge compared to the rainfall excess event occurs because of the physical obstruction to overland flow over the surface of the watershed. Within the watershed, overland flow also accumulates into channels and streams. Both of these characteristics (overland flow and presence of channels and streams) are not present when considering runoff from the surface of a lake or reservoir and therefore a UH is not an appropriate tool to describe its response to a rainfall event.

The applicant provided a set of justifications to support using unit hydrographs for drainage basins of large areas. The applicant stated that several storage areas exist within the Withlacoochee River Basin such as intermittent streams, connected lakes and wetlands, and sinkholes. The applicant stated that in drainage basins with large floodplains with vegetation and other obstructions within the overbank areas, average velocities are likely to remain fairly constant or even decrease to some extent as flow rate increases. The applicant concluded that this behavior would reduce nonlinearity effects.

The staff reviewed the applicant's response to RAI 2.4.3-3 and concluded that the applicant has provided no other supporting evidence, such as data from observed rainfall and runoff events that support this hypothesis. Generally, as discharge increases, flow depth increases, and therefore velocity of flow increases. The staff concluded that the applicant has not presented sufficient information to support the case that nonlinear response in the Withlacoochee River Basin is insignificant.

The applicant acknowledged that published literature recommends derivation of unit hydrographs from large historical storms if the intent is to apply the unit hydrograph for estimation of hypothetical floods such as the PMF from hypothetical storms, such as the PMP.

The applicant also quoted text from Sivapalan et al. (2002) to justify linear runoff response in the Withlacoochee River Basin. The same reference (Sivapalan et al. 2002) also includes this observation, that the applicant did not include in its response: "On the other hand, Robinson et al. [1995], using numerical simulations, showed that nonlinearity at small scales is dominated by the hillslope response, that nonlinearity at large scales is dominated by channel network hydrodynamics, and that nonlinearity does not really disappear at any scale."

The staff disagrees with the applicant that the response of the Withlacoochee River Basin can be considered linear. Because the applicant was not able to provide a technically sound and conservative assessment of the PMF in the Withlacoochee River Basin, the staff issued **RAI 02.04.03-05**, which states:

> In reply to the staff's RAI 2.4.3-03, the applicant stated that application of a UH to predict runoff from the surface of a reservoir is acceptable. The staff disagrees with this approach. The UH theory is used to describe the time distribution of surface runoff at the outlet produced by a constant and uniform rainfall excess event over a watershed. The time delay and attenuation in discharge compared to the rainfall excess event occurs because of the physical obstruction to overland flow over the surface of the watershed. Within the watershed, overland flow also accumulates into channels and streams. Both of these characteristics (overland flow and presence of channels and streams) are not present when considering runoff from the surface of a lake or reservoir and therefore a UH is not an appropriate tool to describe its response to a rainfall event. The applicant should use a rainfall-runoff response function that is appropriate for the surface of Lake Rousseau.
>
> In reply to the staff's RAI 2.4.3-03, the applicant's response includes text quoted from Sivapalan et al. (2002). The same reference (Sivapalan et al. 2002) also includes this observation, that the applicant did not include in its response: "On

> the other hand, Robinson et al. [1995], using numerical simulations, showed that nonlinearity at small scales is dominated by the hillslope response, that nonlinearity at large scales is dominated by channel network hydrodynamics, and that nonlinearity does not really disappear at any scale." The staff disagrees with the applicant that the response of the Withlacoochee River Basin can be considered linear. The applicant should use UHs that are appropriately representative of overland flow and runoff generation conditions in the basin and conservative in predicting the discharge in the Withlacoochee River at the time a PMP event is likely to occur.

The applicant responded to the staff's RAI 02.04.03-05 in a letter dated June 18, 2010 (ML101740490). The applicant's reply to the staff's RAI presented justification for using a unit hydrograph for the surface area of Lake Rousseau. The applicant stated that using a unit hydrograph would result in a conservative estimate of the peak flood discharge because the lag times associated with upstream drainage areas is larger than a day. The staff agreed with the applicant that using a unit hydrograph for the surface area of Lake Rousseau would result in a more conservative discharge. The staff's review is required to ascertain that the analyses used to support safety conclusions in an FSAR are representative of the hydrologic characteristics of the study area in addition to being conservative and the staff believes that the applicant has not demonstrated this requirement conclusively for the study area. The staff also reviewed the applicant's sensitivity analysis used to determine whether the estimated unit hydrographs would accurately predict large flood events in the Withlacoochee River Basin. While the staff agreed with the applicant that its unit hydrographs estimate peak discharge of relatively large floods conservatively, the staff found that the applicant had not applied all literature recommendations for adjustment of unit hydrographs for application to extremely large floods approaching the PMF. To resolve the outstanding questions with regard to the PMF analysis and the appropriate choice of representative parameters, the staff issued **RAI 02.04.03-06**, which states:

> In RAI 2.4.3-05 (RAI ID 4628, Question 17566), the staff requested the applicant to provide a probable maximum flood (PMF) analysis for the Withlacoochee River watershed that used (1) an appropriate rainfall-runoff response function for Lake Rousseau and (2) unit hydrographs for the subbasins of the Withlacoochee River watershed that are appropriately representative of overland flow and runoff generation conditions in the basin and conservative in predicting the discharge in the Withlacoochee River at the time a probable maximum precipitation (PMP) event is likely to occur.
>
> The applicant's response, dated June 18, 2010, stated that the applicant's approach to a unit hydrograph for generation of runoff from the precipitation falling on the surface of Lake Rousseau would result in a conservative estimate of the probable maximum flood because the lag times associated with subbasins upstream of Lake Rousseau are larger than a day. Therefore, the applicant stated that use of the alternative approach of assuming no lag in generation of runoff from precipitation falling on the surface of Lake Rousseau would not be conservative because peak runoff from the upstream subbasins would not coincide with the peak runoff from Lake Rousseau. While NRC agrees that using a unit hydrograph for Lake Rousseau would be more conservative, the analysis

that supports safety conclusions in the FSAR must be representative of the hydrologic characteristics of the study area, in addition to being conservative. The applicant must provide an appropriate rainfall-runoff response function for Lake Rousseau and update the PMF analysis based on this response function.

The applicant's June 18, 2010, response also described a sensitivity analysis that was performed to determine the ability of the subbasin unit hydrographs to predict large floods including the standard project flood. The applicant stated that Snyder peak coefficient, the parameter C_p, was increased from its regional value of 0.6 to 0.8, a 33 percent increase that would result in a corresponding increase of 33 percent to peak discharge. The FSAR Rev 1 Table 2.4.3-221 shows that a C_p value of 0.8 was used for all subbasins. However, the text in FSAR Rev 1 Section 2.4.3.3.1 states that a value of 0.6 was used for C_p.

While the applicant has demonstrated that the unit hydrographs it employs estimate the peak discharge of relatively large floods conservatively, the literature guidance also recommends reduction in time to peak for the unit hydrographs that are used to predict large floods such as the PMF. NRC requests that the applicant:

(1) verify that the value of Snyder peaking coefficient, C_p, used in the PMF analysis is 0.8

(2) adjust time to peak discharge appropriately for each subbasin unit hydrograph

(3) update the PMF analysis

(4) provide input files for the PMF analysis, and

(5) provide related updates to FSAR Section 2.4.3, ensuring that the text is consistent with the analysis performed.

The applicant responded to the staff's RAI 02.04.03-06 in a letter dated November 16, 2010 (ML103300096). The applicant stated that it used a direct runoff function with zero travel time to estimate the contribution from Lake Rousseau's surface. The applicant also verified that a C_p value of 0.8 was used in the PMF analysis and that the C_p value of 0.6 was just the base case reported in the FSAR. The applicant stated that it modified the subbasin unit hydrographs, except that for the surface area of Lake Rousseau by further increasing the peak discharges predicted by unit hydrographs obtained from setting C_p to 0.8 by 25 percent. The applicant also reduced the lag time, or the time to peak discharge of the unit hydrographs, as recommended in literature. The applicant re-estimated the PMF in the Withlacoochee River Basin after making the above changes to the unit hydrographs. The applicant provided text changes to the FSAR that will be incorporated in a future revision. The staff is tracking this proposed FSAR text change as **Confirmatory Item 2.4.3-1**.

Resolution of Confirmatory Item 2.4.3-1

Confirmatory Item 2.4.3-1 is an applicant commitment to update Section 2.4.3 of its FSAR. The staff verified that LNP COL FSAR Section 2.4.3 was appropriately updated. As a result, Confirmatory Item 2.4.3-1 is now closed.

The staff reviewed the applicant's response to RAI 02.04.03-06 and determined that the applicant has chosen to use characterizations that are consistent with the hydrologic characteristics in the Withlacoochee River Basin above the Inglis Dam, specifically the use of a direct discharge function for the surface area of Lake Rousseau. The staff also determined that the applicant has conservatively applied guidance available in literature to adjust unit hydrographs for use in prediction of floods approaching the magnitude of a PMF, specifically increasing the value of Cp and reducing the lag time. The applicant's revised PMF discharges showed a larger and earlier peak. The staff concluded therefore, that the applicant has used appropriate and conservative methods in the estimation of the PMF in the Withlacoochee River Basin above the Inglis Dam.

2.4.3.4.4 Probable Maximum Flood Flow

Information Submitted by the Applicant

The applicant estimated the PMF in the Withlacoochee River Basin using the HEC-HMS computer program with input using the estimated PMP in the basin, the loss rates described in Section 2.4.3.4.2 of this SER, and the unit hydrographs for the 19 subbasins. The applicant assumed that Lake Rousseau was full at the start of the PMP event in the Withlacoochee River Basin. The estimated peak PMF inflow into Lake Rousseau was 1,720 m^3/s (60,755 cfs) and it occurred 4 weeks after the start of the PMP event.

NRC Staff's Technical Evaluation

The staff reviewed the information related to estimation of probable maximum flood flow that was provided by the applicant. To determine that the parameters used in the estimation of PMF flow are representative of the hydrometeorological conditions and demonstrate the required level of conservatism, the staff issued **RAI 02.04.03-04**, which states:

> To meet the requirements of GDC 2, 10 CFR 52.17, and 10 CFR Part 100, estimates of the following characteristics are needed, and should be based on conservative assumptions of hydrometeorologic characteristics in the drainage area: (a) the area of the watershed used to estimate flooding in streams and rivers, (b) the total depth of PMP and the PMP hyetograph, (c) the maximum PMF water surface elevation in streams and rivers with coincident wind-waves, and (d) hydraulic characteristics that describe dynamic effects of PMF on SSCs important to safety. Please clarify the estimation of base flow used in the determination of the PMF discharge.

The applicant responded to the staff's RAI 02.04.03-04 in a letter dated June 23, 2009 (ML091760626). The applicant stated that ANSI/ANS-2.8-1992 recommends that the mean monthly flow during the month of occurrence of the PMF should be used as the baseflow. The applicant stated that because seasonality was not considered in the PMP and subsequent PMF estimations, the mean annual flow was assumed to be the baseflow. The baseflow used was 28.5 m^3/s (1,008 cfs), which was estimated from monthly streamflow statistics published by the USGS for the streamflow gage 02313000, Withlacoochee River near Holder. The applicant also presented mean monthly flow values at this streamflow gauge. The mean monthly streamflow at the Holder gauge varies from 16.1 m^3/s (570 cfs) in June to 46.1 m^3/s (1627 cfs) in September. The applicant also performed an analysis by using mean monthly flow for the months of August through November (mean monthly flow for these months are 35.2, 46.1, 45.8, and 29.1 m^3/s (1,243, 1,627, 1,617, and 1,029 cfs), respectively) to investigate the sensitivity of the PMF water surface elevation. The PMF water surface elevation changed less than 0.03 m (a tenth of a foot). The applicant concluded that the PMF water surface elevation is insensitive to baseflow.

The staff reviewed the descriptions and analysis details provided by the applicant and determined that the applicant has provided sufficient information regarding baseflow in the Withlacoochee River.

2.4.3.4.5 Water Level Determinations

Information Submitted by the Applicant

The applicant estimated the water surface elevations in Lake Rousseau using the HEC-HMS computer program input with the estimated inflow into Lake Rousseau and the Lake Rousseau stage-storage and stage-discharge relationships. The applicant conservatively assumed that the spillway gates on the Inglis Dam would be inoperable during the PMF event. Under these conditions, the applicant estimated that the maximum water surface elevation in Lake Rousseau would be 9.1 m (29.7 ft) NAVD88.

NRC Staff's Technical Evaluation

The staff reviewed the methodology adopted by the applicant in estimation of water surface elevations in Lake Rousseau under the PMF scenario. The staff agrees that the applicant has applied appropriate methods by specifically using the HEC-HMS computer program to route the PMF discharge through Lake Rousseau. The staff also agrees that the applicant has used conservative conditions, specifically the assumption that spillway gates on the Inglis Dam would be inoperable during the PMF event. Therefore, the staff concluded that the applicant has conservatively estimated the maximum water surface elevation in Lake Rousseau during the PMF event. The applicant-estimated maximum water surface elevation in Lake Rousseau during the PMF event—9.1 m (29.7 ft) NAVD88—is significantly lower than the nominal plant grade of LNP Units 1 and 2.

2.4.3.4.6 Coincident Wind-Wave Activity

Information Submitted by the Applicant

The applicant stated that the maximum water surface elevation in Lake Rousseau during the PMF, which is estimated to be 9.1 m (29.7 ft) NAVD88, would be approximately 6.5 m (21.3 ft) below the nominal plant grade floor elevation of 15.5 m (51 ft) NAVD88. Based on this large difference, the applicant concluded that it is unlikely that a wind-wave activity coincident with the PMF would affect the safety-related facilities of the proposed LNP units.

NRC Staff's Technical Evaluation

The staff reviewed the methodology adopted by the applicant for the estimation of wind-induced waves and determined that the applicant did not consider wind-induced waves to be significant because the LNP site is located approximately 4.8 km (3 mi) from Lake Rousseau. After reviewing the applicant's responses to RAIs 02.04.03-05 and 02.04.03-06, the staff has determined that the applicant-estimated maximum water surface elevation in Lake Rousseau during a PMF event (9.1 m (29.7 ft) NAVD88) is acceptable. The maximum water surface elevation of 9.1 m (29.7 ft) NAVD88 in Lake Rousseau does not include wind-wave effects. Because the maximum stillwater elevation of 9.1 m (29.7 ft) NAVD88 in Lake Rousseau is more than 6.4 m (21 ft) below the nominal plant grade of LNP Units 1 and 2, the staff concluded that there is significant margin available between the stillwater elevation and the nominal plant grade. Wind-wave activity from a 2-year coincident wind is unlikely to exceed the available margin. Therefore, the staff concluded that a PMF in the Withlacoochee River Basin would not result in flooding at the LNP site.

The staff had not determined the maximum water surface elevation near the LNP site because the applicant's PMF analysis for the Withlacoochee River Basin was incomplete (see RAIs 02.04.03-05 and 02.04.03-06 above). Because of this issue, the determinations of the PMF water surface elevation and the design basis floodwater surface elevation at the LNP site were incomplete. Therefore, the staff considers RAIs 02.04.03-05 and 02.04.03-06 to be resolved.

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 COL FSAR related to this section.

As set forth above, the applicant has presented and substantiated information to establish the site description. The staff has reviewed the information provided and, for the reasons given above, concludes that the applicant has provided sufficient details about the site description for the staff to determine, as documented in Section 2.4.3 of this SER, that the applicant has met the relevant requirements of 10 CFR 52.79(a)(1)(iii) and 10 CFR Part 100 with respect to determining the acceptability of the site. This addresses COL Information Item 2.4-2.

2.4.4 Potential Dam Failures

2.4.4.1 *Introduction*

FSAR Section 2.4.4 of the LNP COL application addresses 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 SER presents a review of the specific areas related to dam failures. The specific areas of review are as follows: (1) flood waves resulting from severe dam breaching or failure, including those due to hydrologic failure as a result of overtopping for any reason, routed to the site and the resulting highest water surface elevation that may result in the flooding of SSCs important to safety; (2) successive failures of several dams in the path to the plant site caused by the failure of an upstream dam due to plausible reasons, such as a probable maximum flood, landslide-induced severe flood, earthquakes, or volcanic activity and the effect of the highest water surface elevation at the site under the cascading failure conditions; (3) dynamic effects of dam failure-induced flood waves on SSCs important to safety; (4) failure of a dam downstream of the plant site that may affect the availability of a safety-related water supply to the plant; (5) effects of sediment deposition or erosion during dam failure-induced flood waves that may result in blockage or loss of function of SSCs important to safety; (6) failure of onsite water-control or storage structures such as levees, dikes, and any engineered water storage facilities that are located above site grade and may induce flooding at the site; (7) the potential effects of seismic and non-seismic data on the postulated design bases and how they relate to dam failures in the vicinity of the site and the site region; and (8) 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 COL FSAR addresses the site-specific information about potential dam failures. The applicant addressed the information as follows:

<u>*AP1000 COL Information Item*</u>

- LNP COL 2.4-2

In addition, this section addresses the following COL-specific information identified in Section 2.4.1.2 of Revision 19 of the AP1000 DCD.

> Combined License applicants referencing the AP1000 design will address the following site-specific information on historical flooding and potential flooding factors, including the effects of local intense precipitation.

- Probable Maximum Flood on Streams and Rivers – Site-specific information that will be used to determine design basis flooding at the site. This information will include the probable maximum flood 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 of flood level.

No further action is required for sites within the bounds of the site parameter for flood level.

This section of the SER relates to dam failures.

2.4.4.3 *Regulatory Basis*

The relevant requirements of the Commission regulations for the identification of floods, flood design considerations and potential dam failures, and the associated acceptance criteria, are described in Section 2.4.4 of NUREG-0800 (NRC 2007a).
The applicable regulatory requirements for identifying the effects of dam failures are as follows:

- 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), as it 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.

- 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.

Appropriate sections of the following RGs are used by the staff for the identified acceptance criteria:

- RG 1.59, “Design Basis Floods for Nuclear Power Plants” (NRC 1977a), as supplemented by best current practices

- RG 1.102, "Flood Protection for Nuclear Power Plants" (NRC 1976b).

2.4.4.4 *Technical Evaluation*

The NRC staff reviewed Section 2.4.4 of the LNP 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 potential dam failure. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The staff needs an accurate description of the assessment of the potential dam failures to perform its safety assessment. In RAI 2.4.4-1, the staff requested additional information regarding the applicant's 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 on-site water control or storage structures to ensure that the most conservative of plausible conceptual models has been identified.

In a letter dated June 15, 2009 (ML091680038), the applicant's response stated that the safety-related SSCs of LNP Units 1 and 2 are located in the Waccasassa River Basin, which does not have any water-control structures. Therefore, the applicant concluded that the LNP site would be unaffected by severe breaching of upstream dams. Because the nearest water-control structures, Inglis Dam and Spillway and Inglis Lock, are present in the adjoining Withlacoochee River Basin, the applicant analyzed the potential failure of these with a coincident high tide in the Gulf of Mexico. The applicant estimated that the maximum water surface elevation in the Lower Withlacoochee River due to the failure of the Inglis Dam during a PMF event would be approximately 8.2 m (27 ft) lower than the nominal plant grade floor elevation. The applicant did not analyze other water-control structures in the Withlacoochee River Basin upstream of the Inglis Dam because the topographic relief in the river basin is low. The applicant postulated that the flood wave caused by an upstream dam failure would spread in marshlands adjacent to the river channel and therefore would not affect Lake Rousseau or the LNP site.

The staff reviewed the applicant's response and determined that the applicant has adequately identified the dam breach scenarios that may affect the LNP site. However, there are two issues that the staff would independently check in order to verify the applicant's conclusion that upstream dam failures in the Withlacoochee River Basin would not affect the LNP site. The two issues are related to the effects of peaking of unit hydrographs and upstream dam failures on the water surface elevation of Lake Rousseau during a PMF event. These issues are described below.

2.4.4.4.1 Dam-Failure Permutations

Information Submitted by the Applicant

The applicant did not identify any dam-failure permutations. The applicant only postulated and analyzed the failure of the Inglis Dam. The applicant used the Froehlich (1995) method to estimate the peak flow from a postulated failure of the Inglis Dam. To estimate the peak flow, the applicant postulated that Lake Rousseau's storage and height of water at the time of failure would be at their respective maximums, 41,938,381 m^3 (34,000 ac-ft) and 9.4 m (30.7 ft). The applicant-estimated peak discharge from the postulated failure of Inglis Dam is 1,722 m^3/s (60,811 cfs). The applicant noted that in comparison, its estimate of maximum outflow from Lake Rousseau during the PMF event in the Withlacoochee River Basin is 1,720 m^3/s (60,755 cfs).

The applicant used the USACE HEC-RAS model to simulate a steady flow of 1,722 m^3/s (60,811 cfs) through a channel reach downstream of the Inglis Dam. The applicant selected a downstream boundary condition at the shoreline on the Gulf of Mexico equal to the 10 percent exceedance high tide. The applicant obtained a maximum water surface elevation of 7.53 m (24.72 ft) NGVD29. The applicant concluded that a postulated failure of the Inglis Dam would not result in a maximum water surface elevation exceeding 7.3 to 7.6 m (24 to 25 ft) NGVD29 downstream of the dam.

NRC Staff's Technical Evaluation

The staff requires information about all existing and proposed water retaining and water-control structures in the vicinity of the LNP site to ascertain that their possible effects are accounted for in the estimation of the design-basis flood. Because the applicant did not identify dams and water-control structures upstream of Lake Rousseau, in addition to the inflow hydrograph issues described in RAIs 02.04.03-05 and 02.04.03-06, the staff were not able to complete the review of dam failures and their potential effects on the LNP site. In RAI 2.4.4-2, the staff requested additional information related to all existing and proposed water retaining and water control structures both upstream and downstream relative to the LNP site location, including a justification of why failure of these structures would not affect flood elevations near the LNP site.

The applicant responded to the staff's RAI 02.04.04-02 in a letter dated June 15, 2009 (ML091680038). The applicant stated that it reviewed the USACE's National Inventory of Dams database to determine characteristics of dams in the Withlacoochee River Basin. The applicant listed 15 dams in the Withlacoochee River Basin with a total storage capacity of 271 million m^3 (219,650 ac-ft). The heights of these dams range from 3.7 to 16.8 m (12 to 55 ft).

The applicant stated that the difference between the operating pool elevation of Lake Rousseau and the nominal plant floor grade elevation is 7.3 m (24 ft). Because topographical relief in the Withlacoochee River Basin is low, the applicant concluded that floodwaters from a dam-failure event would spread out into marshlands located adjacent to the river channel and therefore not reach the LNP site.

The staff reviewed the applicant's response to RAI 02.04.04-02 and determined that the LNP nuclear island, which has SSCs important to safety, is not located in the Withlacoochee River Basin. The applicant has analyzed a postulated failure of the Inglis Dam but did not consider upstream dam failures. The applicant's reasoning for not considering upstream dam failures is that due to the low topographical relief in the Withlacoochee River Basin, floodwaters from an upstream dam-failure event would spread out into marshlands. The staff determined that the applicant has not shown, using observed data or simulations, that floodwaters in the Withlacoochee River Basin would indeed spread out into marshlands and not affect the water surface elevation in Lake Rousseau.

The staff independently assessed the effect of upstream dam failures in the Withlacoochee River Basin. The applicant identified 15 dams in the Withlacoochee River Basin, 13 of which are located upstream of Lake Rousseau. The applicant stated in response to RAI 02.04.04-02 that there are seven settling areas located in the southern part of the Withlacoochee River Basin, three of which have storage capacities exceeding 12.3 million m^3 (10,000 ac-ft). The applicant also stated that all the settling areas are hydrologically disconnected from the Withlacoochee River. The staff performed a search of the National Inventory of Dams database and found that the Saddle Creek settling areas are listed as privately owned earthen dams. Although the staff was able to find some references to settling areas created near the southern end of the Withlacoochee River Basin (SWFWMD 2009a), it was unable to verify whether these settling areas are hydrologically disconnected from the Withlacoochee River. Therefore, the staff included all 13 dams located upstream of Lake Rousseau in its analysis.

The staff independently determined the effects of upstream dam breaches using two scenarios that may affect water surface elevation in Lake Rousseau and downstream of the lake. The staff's two scenarios are: (1) the estimation of water surface elevation in Lake Rousseau because of failures of all upstream dams during the PMF event while the Inglis Dam remains intact and (2) the estimation of water surface elevation downstream of Lake Rousseau with failure of Inglis Dam coincident with the first scenario. The first scenario would result in the maximum water surface elevation in Lake Rousseau because the Inglis Dam would not fail and the second scenario would maximize the water surface elevation downstream of the Inglis Dam because Inglis Dam's failure would augment the discharge through Lake Rousseau postulated in the first scenario.

The staff assumed that the dams on Saddle Creek settling areas would fail simultaneously as a group and their peak discharges would arrive simultaneously at the outlet of the subbasin in which they are located. The staff also assumed that the Lake Tsala Apopka group of dams, Rufe Wysong Dam, Gant Lake Dam, and the Slush Pond Dam would fail as a group and their peak discharges would arrive at the outlet of the subbasin in which the Lake Tsala Apopka group of dams is located. Because Rufe Wysong Dam, Gant Lake Dam, and the Slush Pond Dam are located upstream of the Lake Tsala Apopka group of dams, the staff's assumption does not consider the attenuation and time lag in their discharges that would occur as the discharge flows downstream. Therefore, the staff's assumption is conservative and would result in greater peak discharges in the Withlacoochee River Basin downstream of the Lake Tsala Apopka group of dams.

The staff used the Froehlich (1995) approach to estimate the peak discharges from all dams using the data provided by the applicant in response to RAI 02.04.04-02. The staff independently verified these peak discharges, which are listed in Table 2.4.4-1. The staff estimated that the combined peak discharge of the dams on Saddle Creek settling area would be 6,524 m^3/s (230,388 cfs) and that for the Lake Tsala Apopka group of dams, Rufe Wysong Dam, Gant Lake Dam, and the Slush Pond Dam would be 3,329 m^3/s (117,546 cfs).

Table 2.4.4-1. Staff-Estimated Peak Discharges from Postulated Failures of Dams Upstream of Lake Rousseau

Dam Name	Maximum Storage (m^3 [ac-ft])	Height (m [ft])	Peak Discharge[1] (m^3/s [cfs])
Brogden Bridge - Lake Tsala Apopka[2]	36,634,409 (29,700)	5.2 (17)	795,1 (28,077.9)
Golf Course Bridge - Lake Tsala Apopka[2]	50,983,503 (41,333)	4.0 (13)	628.4 (22,194.3)
Structure 353 Bridge - Lake Tsala Apopka[2]	74,008,908 (60,000)	5.3 (17.5)	1,014.2 (35,815.1)
Slush Pond[2]	62,908 (51)	15.2 (50)	463.1 (16,353.1)
Gant Lake Dam[2]	651,278 (528)	3.7 (12)	157.2 (5,552.7)
Rufe Wysong Dam[2]	1,603,526 (1,300)	4.6 (15)	270.5 (9,552.4)
Saddle Creek Settling Area No. 1[3]	13,340,206 (10,815)	7.9 (26)	999.5 (35,297.9)
Saddle Creek Settling Area No. 2[3]	19,452,008 (15,770)	7.3 (24)	1,011.6 (35,724.5)
Saddle Creek Settling Area No. 3[3]	4,576,217 (3,710)	5.8 (19)	494.1 (17,448.7)

To create a discharge hydrograph for the combined discharge of the two groups of dams, the staff assumed that all of the storage in the dams within a group would be released during their failure. The staff assumed that the hydrographs would have a triangular shape with a peak discharge equal to the combined peak discharge of the group.

The staff used the Withlacoochee River Basin HEC-HMS model provided by the applicant and modified it to include the two conservatively estimated discharge hydrographs resulting from the respective failures of the two groups of dams in the model at the appropriate locations. The staff simulated the PMF scenario, which now includes conservatively estimated upstream dam-failure hydrographs. The staff's HEC-HMS simulation resulted in a peak outflow discharge of 1,751 m^3/s (61,851 cfs) and a maximum water surface elevation of 9.1 m (29.7 ft) NGVD29 in Lake Rousseau. Therefore, the staff concluded that for the staff's first scenario listed above, the LNP site would be safe from flooding because the plant grade elevation is more than 6.1 m

(20 ft) above the maximum water surface elevation in Lake Rousseau caused by upstream dam failures coincident with the PMF event.

For the staff's second scenario, the staff concluded that the maximum water surface elevation in Lake Rousseau during upstream dam failures coincident with a PMF event in the Withlacoochee River Basin would not exceed 9.1 m (30 ft) NGVD29. Therefore, the applicant's estimate of peak discharge during a postulated failure of the Inglis Dam is conservative because the applicant used a water height of 9.4 m (30.7 ft). The peak discharge of 1,751 m^3/s (61,851 cfs) from Lake Rousseau as estimated by the staff is greater than that estimated by the applicant (1,716 m^3/s [60,597 cfs]) by about 2 percent. The staff's independent assessment described below also showed that increasing the applicant-estimated peak discharge from Lake Rousseau by 50 percent did not result in an appreciable rise in the maximum water surface elevation downstream of Lake Rousseau. To estimate the water surface elevation below Lake Rousseau for the staff's second scenario (failure of Inglis Dam coincident with PMF in Withlacoochee River Basin and failure of upstream dams), the staff conservatively assumed that the discharge from Lake Rousseu would be a combination of peak discharge estimated for the PMF event coincident with upstream dam failures and the peak discharge because of breach of Inglis Dam. Because the staff estimated that peak discharge from Lake Rousseau during the PMF event coincident with upstream dam failures is greater than the peak discharge from the single failure of Inglis Dam, the staff conservatively estimated the combined discharge by doubling the staff-estimated peak discharge from for the PMF event coincident with upstream dam failures. Therefore, the staff-estimated peak discharge for the second scenario is 3,502 m^3/s (123,702 cfs).

The staff performed a steady-state simulation using the HEC-RAS model provided by the applicant with an input discharge of 3,502 m^3/s (123,702 cfs). The staff determined that the maximum water surface elevation below Lake Rousseau for the second scenario would be approximately 9.7 m (31.8 ft) NGVD29. Therefore, the staff concluded that failure of Inglis Dam during the PMF event and coincident upstream dam failures would not result in a flood hazard at the LNP site.

2.4.4.4.2 Unsteady Flow Analysis of Potential Dam Failures

<u>Information Submitted by the Applicant</u>

The applicant did not perform an unsteady flow analysis of potential dam failures. The peak discharge following the failure of the Inglis Dam was used in a steady flow simulation to estimate water surface elevation downstream of the Inglis Dam.

NRC Staff's Technical Evaluation

The staff reviewed the methodology adopted by the applicant in its estimation of design basis floodwater surface elevations. To verify the conservativeness of the applicant's approach, the staff issued **RAI 02.04.04-03,** which states the following:

> To meet the requirements of GDC 2, 10 CFR 52.17, 10 CFR Part 100, and 10 CFR 100.23(d), an appropriate configuration of the cascade of dam failures and its potential to produce the largest flood adjacent to the plant site is needed. Flood waves produced by postulated dam failure scenarios should be routed to the proposed plant site to conservatively estimate the most severe floodwater surface elevation that may affect SSCs important to safety. Please clarify the steady flow methodology for analysis of the dam break-induced flood and to justify why the estimated flood water surface elevations are conservative.

The applicant responded to the staff's RAI 02.04.04-03 in a letter dated June 15, 2009 (ML091680038). The applicant stated that its steady-state analysis of the postulated Inglis Dam and Inglis Lock failure used a downstream water surface elevation specified by a 10 percent exceedance tide. The applicant stated that flood discharge and water surface elevations estimated by a steady-state approach are overestimated for a flow event that is transient. The staff's confirmatory analyses agree with the applicant's explanation. Therefore, the staff concludes that the steady-state simulation used by the applicant would result in a conservative estimate of the floodwater surface elevation.

2.4.4.4.3 Water Level at the Plant Site

Information Submitted by the Applicant

The applicant used the USACE HEC-RAS computer program to estimate water surface elevations downstream of the Inglis Dam after the failure of the dam. The applicant estimated the cross sections of the floodplain from downstream of the Inglis Dam to the Gulf of Mexico using USGS digital terrain data (Figure 2.4.4-1, adapted from FSAR Revision 0 Figure 2.4.4-201). The applicant estimated that the maximum water surface elevation downstream of the Inglis Dam due to its failure would be 7.53 m (24.72 ft) NGVD29. The applicant concluded that the LNP site would not be adversely affected by this flood.

LNP COL 2.4-2

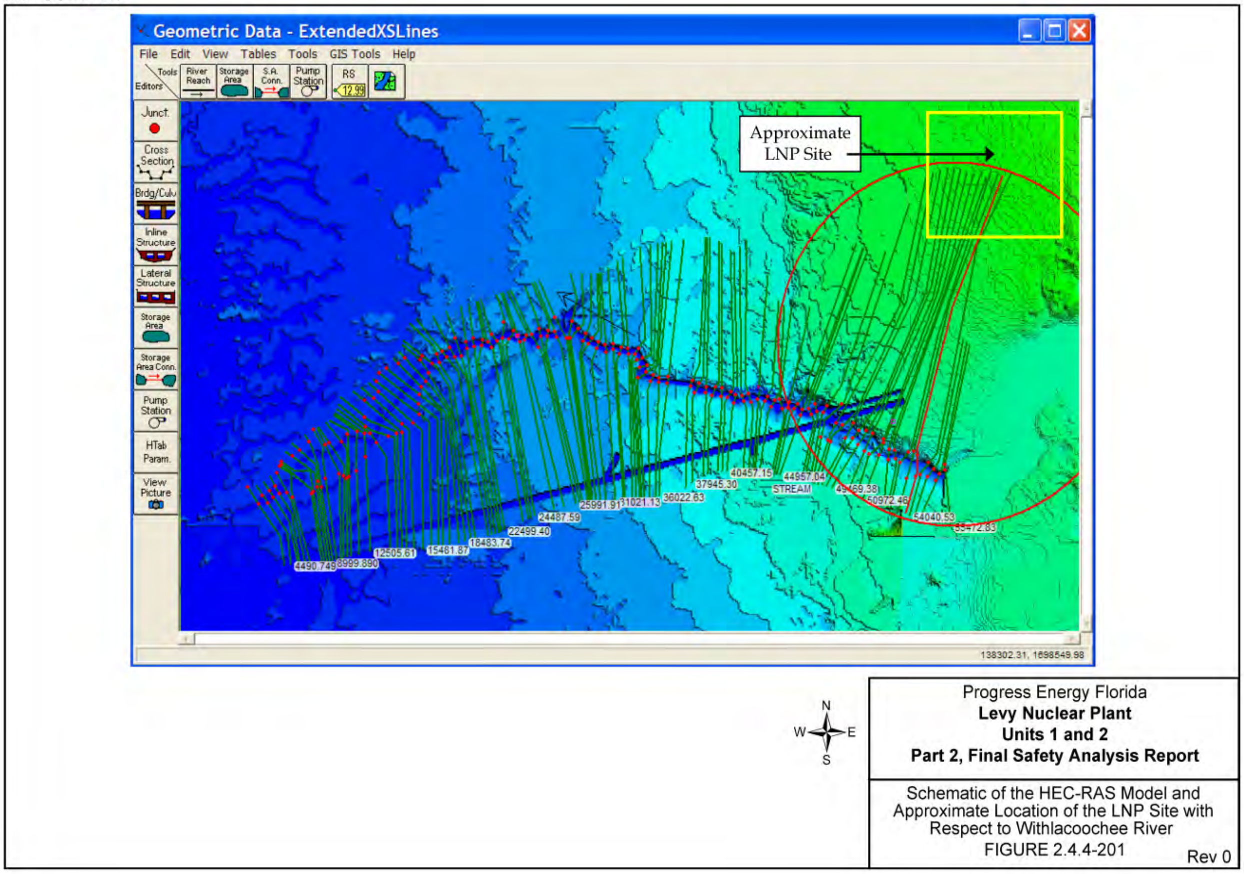


Figure 2.4.4-1. The Cross Sections Used in the HEC-RAS Simulation by the Applicant

NRC Staff's Technical Evaluation

The staff performed an independent analysis to estimate the sensitivity of floodwater surface elevations with respect to the applicant-selected parameters of the dam-failure scenario. The staff considered two cases: (1) a 50 percent increase in the peak discharge used in the applicant's HEC-RAS steady-state simulation and (2) an increase in Manning's n by 50 percent. The staff found that the maximum water surface elevation predicted by HEC-RAS is only minimally sensitive to the altered parameters. The maximum water surface elevation predicted by HEC-RAS for the two sensitivity simulations was 7.9 m (26 ft) NGVD29 compared to the applicant's estimate of 7.53 m (24.72 ft) NGVD29. Therefore, the staff concluded that it is unlikely that the LNP site could be inundated by a dam breach event postulated by the applicant.

The staff has independently assessed two issues in order to verify the applicant's conclusion that upstream dam failures in the Withlacoochee River Basin would not affect the LNP site. The first of these issues was described in RAIs 02.04.03-05 and 02.04.03-06 and addressed peaking of the unit hydrographs used in the PMF simulations. It is plausible that the inflow hydrograph into Lake Rousseau during the PMF would be more severe if peaked unit hydrographs were used in the PMF simulations, which may increase the discharge after the postulated breach of the Inglis Dam. The applicant addressed this issue in response to RAI 02.04.03-06. As stated in Section 2.4.3 of this SER, based on the applicant's response to the staff's RAI 02.04.03-06, the staff concluded that the applicant has used appropriate and conservative methods in the estimation of the PMF in the Withlacoochee River Basin upstream of the Inglis Dam. The second issue with regard to the effect of upstream dam failures on water surface elevations in Lake Rousseau stems from the plausible consideration that upstream dam failures could occur during PMF conditions in the Withlacoochee River Basin. The staff independently assessed the effects of increased water level in Lake Rousseau, as described in the applicant's responses to RAIs 02.04.03-05 and 02.04.03-06. The staff's independent assessment of dam failures in the Withlacoochee River Basin upstream of Lake Rousseau is described in Section 2.4.4.4.1 of this SER.

The staff performed an independent assessment of dam failures in the Withlacoochee River Basin upstream of Lake Rousseau after the applicant responded to staff's RAIs 02.04.03-05, 02.04.03-06, and 02.04.04-02. The staff's independent assessment is described in Section 2.4.4.4.1 of this SER. Based on its independent assessment, the staff concluded that failures of dams in the Withlacoochee River Basin upstream of Lake Rousseau would not result in flooding of the LNP site. The staff also concluded that failure of Inglis Dam coincident with a PMF event and upstream dam failures would not result in appreciable increase water surface elevations downstream of the dam to affect the LNP site. Therefore, the staff considers RAIs 02.04.03-05, 02.04.03-06, and 02.04.04-02 to be resolved.

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 COL FSAR related to this section.

As set forth above, the applicant has presented and substantiated information to establish the site description. The staff has reviewed the information provided and, for the reasons given above, concludes that the applicant has provided sufficient details about the site description for the staff to determine, as documented in Section 2.4.4 of this SER, that the applicant has met the relevant requirements of 10 CFR 52.79(a)(1)(iii) and 10 CFR Part 100 with respect to determining the acceptability of the site. This addresses part of COL information item 2.4-2.

2.4.5 Probable Maximum Surge and Seiche Flooding

2.4.5.1 *Introduction*

FSAR Section 2.4.5 of the LNP COL application addresses the probable maximum surge and seiche (PMSS) flooding to ensure that any potential hazard to the safety-related SSCs at the proposed site has been considered in compliance with the Commission's regulations.

Section 2.4.5 of this SER presents evaluation of the following topics based on data provided by the applicant in the 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 waterbody that may affect floodwater surface elevations near the site or cause a low water surface elevation affecting safety-related water supplies; (4) wind-induced wave run-up under PMH or PMWS winds; (5) effects of sediment erosion and deposition during a storm surge and seiche-induced waves that may result in blockage or loss of function of SSCs important to safety; (6) the potential effects of seismic and non-seismic information about the postulated design bases and how they relate to a surge and seiche in the vicinity of the site and the site region; (7) any additional information requirements prescribed in the "Contents of Application" sections of the applicable subparts to 10 CFR Part 52.

2.4.5.2 *Summary of Application*

This section of the COL FSAR addresses the site-specific information about PMSS flooding in terms of impacts on structures and water supply. The applicant addressed these issues as follows:

AP1000 COL Information Item

- LNP COL 2.4-2

In addition, this section addresses the following COL-specific information identified in Section 2.4.1.2 of Revision 19 of the AP1000 DCD.

> Combined License applicants referencing the AP1000 design will address the following site-specific information on historical flooding and potential flooding factors, including the effects of local intense precipitation.
>
> - Probable Maximum Flood on Streams and Rivers – Site-specific information that will be used to determine design basis flooding at the site. This information will include the probable maximum flood 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 of flood level.

No further action if required for sites within the bounds of the site parameter for flood level.

2.4.5.3 *Regulatory Basis*

The relevant requirements of the Commission regulations for the identification of floods, flood design considerations and potential dam failures, and the associated acceptance criteria, are described in Section 2.4.5 of NUREG-0800 (NRC 2007a).

The applicable regulatory requirements for identifying the effects of dam failures are:

- 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.

- 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.

Appropriate sections of the following RGs are used by the staff for the identified acceptance criteria:

- RG 1.59, “Design Basis Floods for Nuclear Power Plants” (NRC 1977a), as supplemented by best current practices; and

- RG 1.102, “Flood Protection for Nuclear Power Plants” (NRC 1976b).

2.4.5.4 *Technical Evaluation*

The NRC staff reviewed Section 2.4.5 of the LNP 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 probable maximum surge and seiche flooding. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

2.4.5.4.1 Probable Maximum Winds and Associated Meteorological Parameters

Information Submitted by the Applicant

The applicant stated that between the years 1851 and 2006, northwest Florida was struck by 57 hurricanes. Fourteen of these hurricanes were classified as major hurricanes but none were of Category 4 or 5.

The applicant estimated the meteorological parameters of the PMH from NOAA NWS Report 23. The applicant-estimated PMH parameters are listed in Table 2.4.5-1.

Table 2.4.5-1. Applicant-Estimated PMH Parameters

Parameter	Minimum Value	Maximum Value	Unit
Central pressure	88.9 (889)	89.1 (891)	kPa (millibar)
Peripheral pressure	102 (1,020)	102 (1,020)	kPa (millibar)
Radius of maximum winds	12.4 (6.7)	41.3 (22.3)	km (nautical mile)
Forward speed	25.7 (16)	37 (23)	km/hr (mi/hr)
Maximum wind speed	251 (156)	252.7 (157)	km/hr (mi/hr)
Track direction	200	245	degree from north

The applicant estimated the 10 percent exceedance high spring tide of 1.3 m (4.3 ft) mean low water from RG 1.59 (NRC 1977a). The applicant reported a maximum astronomical tide of 1.5 m (4.9 ft) mean lower-low water based on tide data at Cedar Key, Florida.

NRC Staff's Technical Evaluation

An accurate description of the assessment of PMSS events at the LNP site is needed for the staff to perform its safety assessment. To resolve inconsistencies observed in the information presented by the applicant with regard to observed hurricanes, tropical storms, and tropical depressions, staff issued **RAI 02.04.05-01**, which states:

> To meet the requirements of GDC 2, 10 CFR 52.17, and 10 CFR Part 100, estimates of the probable maximum hurricane (PMH) and the probable maximum storm surge, are needed. The PMH, as defined by NOAA NWS Report 23, should be estimated for coastal locations that may be exposed to these events. In the FSAR text, it is stated that FSAR Table 2.4.5-201 contains a list of hurricanes that came within 80.5 km (50 mi) of the LNP site during 1867–2004. The table contains a list of events that includes hurricanes, tropical storms, and tropical depressions. Please resolve this inconsistency.

The applicant responded to the staff's RAI 02.04.05-01 in a letter dated July 20, 2009 (ML092030128). The applicant agreed with the staff's observation regarding FSAR Table 2.4.5-201 and updated that table to include only a list of recorded hurricanes.

In RAI 2.4.5-2, the staff requested additional information related the applicant's use of Hsu's empirical equation for the estimation of PMH storm surge and why the applicant considered the estimated coastal storm surge elevations under PMH conditions to be conservative.

The applicant responded to the staff's RAI 02.04.05-02 in a letter dated July 20, 2009 (ML092030128). The applicant stated that Hsu's method (Hsu et al. 2006), which uses three key pieces of information—minimum sea level pressure, shoaling factor, and correction factor for storm motion—has been validated using data from recent hurricanes, including Katrina and Rita. The applicant used parameters of a PMH storm to estimate the PMSS at the coastline and compared it to the coastal storm surge elevations given in RG 1.59 (NRC 1977a). The applicant-estimated coastal storm surge including the 10 percent exceedance high tide using Hsu's method (Hsu et al. 2006) was slightly higher than that obtained by converting the value specified in RG 1.59 (NRC 1977a) to the same datum. The applicant concluded therefore, that Hsu's method (Hsu et al. 2006) is conservative.

The staff reviewed the applicant's response to RAI 02.04.05-02 and calculations to determine that Hsu's empirical method (Hsu et al. 2006) produced a higher storm surge estimate that that specified in RG 1.59 (NRC 1977a) at the coastline near the LNP site. Therefore, the staff agrees with the applicant that Hsu's empirical method (Hsu et al. 2006) is conservative insofar as it is used to estimate coastal storm surge near the LNP site.

2.4.5.4.2 Surge and Seiche Water Levels

Information Submitted by the Applicant

The applicant used three approaches for estimating the PMH storm surge at the LNP site. These methods are based on (1) guidance in RG 1.59 (NRC 1977a), (2) results obtained by NOAA NWS using its Sea, Lake, and Overland Surge from Hurricanes (SLOSH) model for several combinations of hurricane parameters, and (3) correlating the SLOSH estimates with an empirical equation.

Storm Surge Estimate from Regulatory Guide 1.59

The applicant assumed that the estimates of storm surge at Crystal River provided in Appendix C of RG 1.59 (NRC 1977a) are applicable for the LNP site because of the proximity of the site to this location. The applicant obtained the following PMH storm surge parameters on the open coast near Crystal River from RG 1.59 (NRC 1977a):

Wind setup	8.1 m (26.55 ft)
Pressure setup	0.8 m (2.65 ft)
Initial rise	0.2 m (0.6 ft)
10 percent exceedance high tide	1.3 m (4.3 ft) MLW
Total surge	10.4 m (34.1 ft) MLW

Storm Surge Estimate from NOAA NWS SLOSH Runs

The applicant stated that SLOSH model results are generally accurate to approximately 20 percent of the computed value. The applicant chose four coastal points near the LNP site and extracted the maximum of the maximum envelope of water (MOM) values from NOAA NWS pre-computed SLOSH model runs for hurricanes of Categories 1 through 5. The applicant also obtained the MOM values for the towns of Yankeetown and Inglis and for the location of the LNP site. The SLOSH model MOM scenarios predicted that the LNP site would be dry from storm surge caused by hurricanes of Categories 1 through 5.

Storm Surge Estimate for the PMH Using Hsu's Empirical Method

The applicant used an empirical equation proposed by Hsu et al. (2006) to estimate the open coast PMH storm surge. The equation uses two empirical coefficients, one called the shoaling factor and the other the storm motion factor, along with a minimum sea-level pressure for the hurricane. The applicant estimated the shoaling coefficient using the location of the coast near the LNP site, specifically the Cedar Key NOAA gauge site, along with a nomograph provided by Hsu et al. (2006). The storm motion factor was estimated using PHM storm track parameters, forward speed, and track direction (see Table 2.4.5-1), along with a nomograph provided by Hsu et al. (2006). The applicant reported that the maximum value of the storm motion factor was estimated to be 0.7.

The applicant estimated the storm surge heights induced by hurricanes of Categories 1 through 5 at the coast using Hsu's method (Hsu et al. 2006) and compared them to the average of the previously selected four coastal points' storm surge estimated by the SLOSH model. The applicant concluded that because storm surges estimated by Hsu's method (Hsu et al. 2006) were consistently higher than those from the SLOSH model, results obtained from Hsu's method (Hsu et al. 2006) were conservative.

The applicant obtained a relationship between inland storm surge heights and the coastal storm surge heights from NOAA NWS pre-computed SLOSH model runs for two locations: Yankeetown and Inglis. A similar relationship for storm surge at the LNP site could not be obtained because the LNP site location was dry in all SLOSH model runs. The applicant concluded that these two relationships, for Yankeetown and Inglis, could be used to estimate the storm surge height at the inland location if the storm surge height at the Gulf coast was known, irrespective of the intensity of the hurricane.

The applicant proposed that the storm surge at the LNP site be obtained from an extrapolation relationship based on the storm surge heights at Yankeetown and Inglis and the corresponding distances of the three locations from the Gulf coast. Using this relationship, the applicant estimated the storm surge height at the LNP site for hurricanes of Categories 1 through 5. All of these storm surges heights were reported as "(dry)" in FSAR Revision 0 Table 2.4.5-214.

The applicant performed a set of estimation of storm surge at the LNP site using 1000 randomly selected combinations of PMH parameters. The applicant did not provide any detail about how storm surge at the LNP site was obtained from these sets of PMH parameters. The maximum applicant-estimated stillwater storm surge at the LNP site was 12.60 m (41.33 ft).

The applicant did not consider seiches in Lake Rousseau as the controlling influence and stated that the potential for flooding at the site due to seiches in Lake Rousseau is insignificant.

NRC Staff's Technical Evaluation

The NRC staff reviewed the analysis and data provided by the applicant. To obtain clarification on the conversion of datums and tabular presentation of data used in the applicant's analysis, the staff issued **RAI 02.04.05-03**, which states:

> To meet the requirements of GDC 2, 10 CFR 52.17, and 10 CFR Part 100, estimates of the probable maximum hurricane (PMH) and the probable maximum storm surge are needed. The storm surge induced by the PMH should be estimated as recommended by Regulatory Guide 1.59, supplemented by current best practices. Please clarify the details of how the conversion from MSL to NGVD29 was made and provide details of how the Hsu method storm surge heights in FSAR Table 2.4.5-213 were obtained. Please clarify why the table is titled "PMH Analysis for the LNP Site," since it appears that the values reported in this table are for storm surges for hurricanes of categories 1 through 5 and not for the PMH.

The applicant responded to the staff's RAI 02.04.05-03 in a letter dated July 20, 2009 (ML092030128). The applicant stated that the Cedar Key tidal datum was used to convert water surface elevation from mean sea level to NGVD29 and NAVD88 datums. The applicant used the NOAA VERTCON tool to convert between NGVD29 and NAVD88 datums. The staff determined in its independent review that the Cedar Key NOAA tide gauge is located closest to the LNP site and therefore is the most appropriate location to use for antecedent tidal elevations.

The applicant stated that storm surge water surface elevations reported in FSAR Table 2.4.5-213 were obtained using Hsu's empirical equation (Hsu et al. 2006) along with parameters for hurricanes of Category 1 through 5 listed in FSAR Table 2.4.5-205, with the mean of the atmospheric pressure range used for each hurricane category in the equation. The staff reviewed Hsu's methodology (Hsu et al. 2006) along with the parameters listed in FSAR Table 2.4.5-205 and determined that the applicant has adequately used the empirical method.

The applicant stated that FSAR Table 2.4.5-213 was labeled "PMH Analysis for the LNP Site" because it represents on step in the process of estimating the PMSS at the LNP site. The applicant stated that the title of the table would be revised for clarity. To resolve inconsistencies in the application of the SLOSH model as presented in the FSAR, the staff issued **RAI 02.04.05-04**, which states:

> To meet the requirements of GDC 2, 10 CFR 52.17, and10 CFR Part 100, an estimate of wind-induced wave runup under PMH winds is needed. The controlling flood water surface elevations are estimated based on the combination of appropriate ambient water surface elevations, critical storm surge or seiche water surface elevations, and coincident wind-wave action as described in ANSI/ANS-2.8-1992.
>
> (1) The applicant stated in FSAR Revision 0, Section 2.4.5.2.3 page 2.4-37: "Since the datum used in the SLOSH model is NGVD, formerly known as the Sea Level Datum of 1929, an astronomical tide level above NGVD29 would add additional height to the values computed by the SLOSH model. Thus, the SLOSH model accounts for astronomical tides." Jelesnianski et al. (1992) clearly state that astronomical tide is ignored by the SLOSH model except for its superposition onto the computed surge. The applicant's statement conveys a broader interpretation of the capabilities of the SLOSH model in how it incorporates the effect of astronomical tide in surge computations.
>
> (2) The applicant stated in FSAR Revision 0, Section 2.4.5.2.3 page 2.4-37: "Generally, waves do not add significantly to the total area flooded by storm surge and can usually be ignored." The applicant also stated in FSAR Revision 0, Section 2.4.5.3.1 page 2.4-41: "As mentioned in FSAR Subsection 2.4.5.2.3, the SLOSH model does not include the additional heights generated by wind-driven waves on top of the stillwater storm surge. Therefore, wind-driven wave height needs to be determined." While the first statement may be true inasmuch as the area of inundation is concerned, it gives an impression that wind waves on top of storm surge stillwater elevation may be ignored, which is not the case, as stated by the second quote.
>
> Please resolve these inconsistencies, or explain why your statements are sufficient.

The applicant responded to the staff's RAI 02.04.05-04 in a letter dated July 20, 2009 (ML092030128). The applicant stated that the SLOSH model accounts for tides by specifying the initial tide level. The applicant stated that the SLOSH model results presented in FSAR Tables 2.4.5-206 through 2.4.5-209 used an initial tidal elevation of 0.8 m (2.5 ft) NGVD29, whereas the 10 percent exceedance tide for Cedar Key tidal gauge is 0.6 m (2.01 ft) NGVD29.

Therefore, the applicant concluded that its PMH analysis is based on a conservative estimate of the initial tidal elevation. The staff reviewed the applicant response and its calculation package to determine whether the initial tidal elevation is more conservative than the recommended 10 percent exceedance tide. Therefore, the staff determined that the applicant's PMSS estimates used a conservative value for initial tidal elevation.

The applicant stated that for clarity and to be more specific to site conditions, the statement "generally, waves do not add significantly to the total area flooded by storm surge and can usually be ignored" would be removed from the FSAR. The staff determined that the removal of the aforementioned phrase would clarify the contribution of wind driven waves to storm surge. The staff considers RAI 02.04.05-04 to be resolved.

To obtain clarification on the hydrodynamic basis of the analysis presented by the staff issued **RAI 02.04.05-05**, which states:

> To meet the requirements of GDC 2, 10 CFR 52.17, and 10 CFR Part 100, estimates of the probable maximum hurricane (PMH) and the probable maximum storm surge are needed. The storm surge induced by the PMH should be estimated as recommended by Regulatory Guide 1.59, supplemented by current best practices. Please clarify and justify the hydrodynamic basis for the extrapolation equation, FSAR Revision 0 Equation 2.4.5-5, used for estimation of storm surge at the LNP site.

The applicant responded to the staff's RAI 02.04.05-05 in a letter dated July 20, 2009 (ML092030128). The applicant provided an explanation of how three methods, based on RG 1.59 (NRC 1977a), NOAA pre-computed SLOSH model simulations for hurricanes of Category 1 through 5, and Hsu's empirical approach (Hsu et al. 2006), were used in the FSAR. The applicant stated that the mechanism of propagation of waves and consequent flooding of inland locations is based on the SLOSH model pre-computed results. The applicant stated that extrapolation of the SLOSH model pre-computed results to predict the PMSS at the LNP site is based on hydrodynamics of the model itself.

The staff disagreed with the applicant's assessment because it used an extrapolation technique. Coastal hydrodynamics, especially the interaction of storm surge with inland topography is a highly complex and nonlinear process. The staff disagreed that the extrapolation procedure used by the applicant can accurately be used to predict the storm surge resulting from a PMH by only using a few points in the modeling domain. The staff also determined that a technically sound and demonstrably conservative approach should be used to estimate the PMSS at the LNP site. To resolve this pending issue, the staff drafted **RAI 02.04.05-09**, which states:

> In response to the staff's RAI 2.4.5-05, the applicant stated that the extrapolation equation that was used to estimate PMSS at the LNP site is based on National Oceanic and Atmospheric Administration National Weather Service's Sea, Lake and Overland Surges from Hurricanes (SLOSH) modeling results for hurricanes of Categories 1 through 5 in the Gulf of Mexico near the LNP site. Through independent confirmatory analysis, the staff determined that the Probable Maximum Storm Surge (PMSS) water surface elevations obtained by using the

> extrapolation procedure described by the applicant may be conservative, but is not technically valid because there is no hydrodynamic basis that captures the complex interaction of the storm surge and inland topography within the equation.
>
> Provide the following information: (a) an analysis of the PMSS event using a technically sound and conservative approach such as those predicted by a storm surge model (e.g., SLOSH) with input from appropriate Probable Maximum Hurricane scenarios, (b) an estimate of sea level rise accounting for current climatic predictions, and (c) if factored into the PMSS analysis (i.e., application of margins), a detailed description of the process for determining uncertainty estimations.

The applicant's responses to RAIs 02.04.05-10 and 02.04.05-11 described below, document the applicant's use of the SLOSH model to simulate PMH conditions directly as opposed to extrapolating from pre-existing Category 1 through 5 results. Because the applicant no longer relies on pre-computed SLOSH model scenarios for hurricanes of Categories 1 through 5, the portion of the RAI 02.04.05-05 related to the extrapolation method used before is obsolete.

To ascertain whether the applicant has considered other mechanisms in addition to surge in the determination of flooding at the site, the staff issued **RAI 02.04.05-06**, which states:

> To meet the requirements of GDC 2, 10 CFR 52.17, and 10 CFR Part 100, estimates of seiche and resonance in waterbodies induced by meteorological causes, tsunamis, and seismic causes are needed. Please address the possibility of seiches of meteorological and seismic origin in Lake Rousseau; including, the possibility of resonance in Lake Rousseau that may amplify any potential seiche activity.

The applicant responded to the staff's RAI 02.04.05-06 in a letter dated July 20, 2009 (ML092030128). The applicant stated that Lake Rousseau is located approximately 4.8 km (3 mi) south of the LNP site and its operating pool elevation is maintained more than 6.1 m (20 ft) below the nominal plant grade floor elevation of safety-related structures to be built at the LNP site. Because of the significant difference in LNP nominal plant grade floor elevation and the operating pool elevation of Lake Rousseau and because of limited fetch due to the long and narrow shape of the lake, the applicant concluded that the possibility of a meteorologically induced seiche affecting LNP safety-related SSCs is insignificant. The applicant compared the runup and run-in induced by seismically generated tsunamis in the Gulf of Mexico—5.7 m (18.6 ft) and 0.89 km (0.55 mi), respectively—with the elevation and location of the LNP site and concluded that a seismically generated seiche would not affect the site. The applicant also stated that the possibility of resonance in Lake Rousseau due to a seismic event is insignificant.

The staff agrees with the applicant that a significant margin, greater than 6.1 m (20 ft), exists between the operating pool elevation of Lake Rousseau and the nominal plant grade floor elevation of safety-related SSCs. The staff reviewed the characteristics of Lake Rousseau and determined that it is a shallow lake, with an average depth of less than 3 m (10 ft). Also, because the lake is narrow and long in the east-west direction and the LNP site is located to its north, there is limited fetch available for waves to develop. Because of these characteristics, the staff determined that waves set up in Lake Rousseau would be limited by fetch and by water depth. The USACE Coastal Engineering Manual (CEM) (Scheffner 2008) suggests that waves are limited to 0.6 times water depth. The staff determined, therefore, that waves set up under most extreme meteorological conditions would not exceed approximately 1.8 m (6 ft) in height. Because the nominal plant grade floor elevation of safety-related SSCs at the LNP site is located more than 6.1 m (20 ft) above the operating pool elevation of Lake Rousseau, the staff concluded that meteorologically or seismically induced waves setup in the lake would not adversely affect the plant.

To ascertain that the applicant has considered all plausible PMH scenarios and used appropriate initial and boundary conditions in the analysis of surge staff issued **RAI 02.04.05-10**, which states:

> In RAI 2.4.5-09 (RAI ID 4629, Question 17567), the staff requested the applicant to provide the following information: (a) an analysis of the probable maximum storm surge (PMSS) event using a technically sound and conservative approach such as that predicted by a storm surge model (e.g., Sea, Lake, and Overland Surges from Hurricanes [SLOSH]) with input from appropriate Probable Maximum Hurricane (PMH) scenarios, (b) an estimate of sea level rise accounting for current climatic predictions, and (c) if factored into the PMSS analysis (i.e., application of margins), a detailed description of the process for determining uncertainty estimations. The applicant’s response, dated June 18, 2010, does not appear to describe an estimation of PMSS at and near the LNP site using PMH scenarios input into a currently accepted hydrodynamic storm surge model. NRC requests that the applicant:
>
> (1) utilize a set of plausible PMH scenarios consistent with National Oceanic and Atmospheric Administration (NOAA) National Weather Service (NWS) Report 23 (NWS 23)[11] as input to a currently accepted storm surge model (such as SLOSH)
>
> (2) use initial open-water conditions that are consistent with current understanding of long-term sea-level rise and are valid for the life of the proposed plant
>
> (3) provide estimates of coincident wind-wave runup
>
> (4) maps of highest PMSS water surface elevation at and near the LNP site, and

[11] Schwerdt et al., 1979.

(5) provide updates to FSAR Section 2.4.5 including descriptions of data, methods, model setup, PHM scenarios and how they are consistent with NWS 23, treatment of uncertainty in the analysis, and available margins.

The applicant responded to the staff's RAI 02.04.05-10 in a letter dated January 27, 2011 (ML110340018). The applicant stated that it performed a confirmatory analysis using SLOSH Version 3.95 for the estimation of the PMH surge elevation at the LNP site. The applicant used the Cedar Key Basin for the analysis. The applicant selected PMH parameters based on NWS Report 23. The applicant determined the PMH antecedent water levels including a 10 percent exceedance spring high tide elevation of 0.98 m (3.23 ft) NAVD88 and a 100-year sea level rise of 0.18 m (0.59 ft) for a combined antecedent initial water level of 1.16 m (3.82 ft) NAVD88. The applicant simulated 576 preliminary cases using the SLOSH model, which varied in terms of landfall location, radius to maximum winds, forward speed, and track direction. The applicant examined the preliminary results and selected the case that yielded the highest water level. Based on this case, the applicant developed a refined and simulated a collection of new SLOSH cases to more precisely determine the conditions leading to the highest water elevation associated with the PMH. The applicant finally determined that a PMH with a radius to maximum winds of 41.8 km (26 mi), a forwards speed of 37 km/hr (23 mph) coming from 225 degree clockwise from north, yielded a surge at the LNP site of 14.5 m (47.7 ft) NAVD88 where the ground level is about 12.8 m (42 ft) (no datum given). The applicant determined that PMH wave setup at the LNP is 0.18 m (0.6 ft) and the wave runup is 0.45 m (1.48 ft) yielding a PMSS of 15.17 m (49.78 ft) NAVD88 (14.54 m (47.70 ft NAVD88) + 0.18 m (0.6 ft) + 0.45 m (1.48 ft)). The applicant reasoned that in the analysis described in the RAI response yielded a PMSS (15.17 m (49.78 ft) NAVD88) that closely corresponded with that previously described in the FSAR (15.09 m (49.52 ft) NAVD88), that the value presented in the FSAR would be used as the characteristic PMH flood elevation at the site.

The staff reviewed the applicant's approach to estimation of the initial water elevation for a hydrodynamic storm surge model using tidal data presented in RG 1.59 (NRC 1977a) for the Cedar Key tide gauge, and NOAA's description of predicted tides. The staff determined that NOAA estimates harmonic constants at reference tide stations that are used to predict the harmonic component of tidal variations at the reference stations. Observed tide water levels also include the effects of wind-wave activity and initial rise. Both of these additional components manifest as random variations added to the harmonic component of the tidal variations. Because these random variations are independent of the harmonic forcings (mainly gravitational forces of the sun and the moon) and therefore can occur at any time, there is no assurance the "high" random variations of tides would be in phase with the highs of the predicted tides. Therefore, estimating the 10 percent exceedance tide from raw tide water level observations can result in the underestimation of the initial water level (represented by 10 percent exceedance of predicted tides plus initial rise). RG 1.59 (NRC 1977a) does not describe how the initial rise reported for various locations in Appendix C of the guide was estimated. The staff concluded that the applicant had not provided sufficient information. Therefore, the staff issued **RAI 02.04.05-11**, which states:

> In RAI 2.4.5-10, the staff requested the applicant to provide supplemental information; the staff stated that the applicant must (1) use a set of plausible

probable maximum hurricane (PMH) scenarios consistent with the National Oceanic and Atmospheric (NOAA) National Weather Service (NWS) Report 23 (NWS 23) as input to a currently accepted storm surge model (such as NWS Sea, Lake, and Overland Surges from Hurricanes [SLOSH]), (2) use initial open-water conditions that are consistent with current understanding of long-term sea-level rise and are valid for the life of the proposed plants, (3) provide estimates of coincident wind-wave runup, (4) provide maps of highest probable maximum storm surge (PMSS) water surface elevation at and near the LNP sites, and (5) provide updates to FSAR Section 2.4.5, including descriptions of data, methods, model setup, PMH scenarios and how they are consistent with NWS 23, treatment of uncertainty in the analysis, and available margins.

The applicant responded to RAI 2.4.5-10 on January 27, 2011. The staff's review of the applicant's response to RAI 2.4.5-10 has raised the following issues:

(1) Regulatory Guide (RG) 1.59 recommends that the following components of PMSS be estimated: (a) probable maximum surge (wind and pressure setups), (b) 10 percent exceedance tide, and (c) initial rise (forerunner or sea-level anomaly). The wind wave runup also needs to be added to obtain the PMSS. The applicant did not use an initial rise in its SLOSH simulations. RG 1.59 recommends an initial rise of 0.6 ft for Crystal River, FL. Because the value of initial water surface can have nonlinear effects on SLOSH predictions, 10 percent exceedance tide, initial rise, and long-term sea level rise should be combined to specify the initial water surface in SLOSH for simulation of the PMH scenarios.

    In a subsequent teleconference, the applicant stated its interpretation of RG 1.59 recommendations. The applicant stated that RG 1.59 recommends use of initial rise as an additional component of the initial water level if the 10 percent exceedance tide is estimated from predicted tides. The applicant stated that use of initial rise is not necessary because its approach used observations of tidal water levels that already contain the effects of initial rise.

(2) The applicant has not used the US Army Corps of Engineers Coastal Engineering Manual (CEM) for estimation of coincident wind wave activity. The CEM approach is recommended in SRP 2.4.5 as the currently accepted practice. The applicant did not provide justification why it used another approach. In a subsequent teleconference, the applicant stated that they did in fact use the CEM approach to estimate wind wave activity although this fact was not clearly stated in the response to RAI 2.4.5-10.

(3) The applicant states that the chosen PMSS maximum water surface elevation value for the LNP site is 49.52 ft NAVD88, not the higher estimate of 49.78 ft NAVD88 obtained from the SLOSH PMSS

simulations. The PMSS maximum water surface elevation of 49.52 ft NAVD88 reported in the FSAR was obtained using an approach that the staff disagreed with previously. Also, the applicant added long-term sea-level rise and initial rise estimates after estimating the PMSS; this approach would not account for the nonlinear effects of initial water surface elevation on the PMSS.

The NRC staff requests the following additional information:

(1) The staff reviewed the applicant's approach to estimation of initial water level for a hydrodynamic storm surge model. The staff also reviewed RG 1.59, tidal data at the Cedar Key tide gauge, and NOAA's description of predicted tides. The staff determined that NOAA estimates harmonic constants at reference tide stations that are used to predict the harmonic component of tidal variations at the reference stations. Observed tide water levels also include the effects of wind wave activity and initial rise. Both of these additional effects manifest as random variations added to the harmonic component of the tidal variations. Because these random variations are independent of the harmonic forcings (mainly gravitational forces of the sun and the moon) and therefore can occur at any time, there is no assurance that "high" random variations of tides would be in phase with the highs of the predicted tides. Therefore, estimating the 10 percent exceedance tide from raw tide water level observations can result in underestimation of the initial water level (represented by 10 percent exceedance of predicted tides plus initial rise). RG 1.59 does not describe how initial rise reported for various locations in Appendix C of RG 1.59 was estimated.

The staff needs the following information to complete its review of the PMSS at the LNP site:

a. A detailed description of the applicant's approach used to estimate the initial water level for use in the SLOSH model runs, an analysis of how this approach is consistent with the recommendations of RG 1.59, a statement of the difference in the numerical values of the initial water level obtained by the applicant's approach and that recommended by RG 1.59, and a detailed justification of why the difference between the two numerical values would result in an insignificant difference in the PMSS maximum water surface elevation at the LNP site, or

b. An updated PMSS maximum water surface elevation at the LNP site that is a combination of (i) maximum stillwater elevation from a SLOSH simulation carried out with an initial water surface elevation estimated following the guidelines of RG 1.59 and using more recent tide data and (ii) wind wave effects using the CEM approach (see (2) below).

(2) Provide an update to FSAR text that clearly describes how the CEM approach was used to estimate wind wave activity coincident with PMSS maximum water surface elevation at the LNP site.

(3) Provide updates to FSAR that describe appropriately selected PMSS characteristics at the LNP site. Provide a discussion of available margins between the DCD Maximum Flood Level site parameter (the design grade elevation or the DCD plant elevation of 100 ft) and the highest PMSS water surface elevation accounting for coincident wind-wave activity.

The applicant responded to the staff's RAI 02.04.05-11 in a letter dated June 21, 2011 (ML11175A300). To address part (1) of the staff's request, the applicant performed an updated PMSS maximum water surface elevation at the LNP site by estimating an initial water surface elevation for the SLOSH model following the guidance in RG 1.59 (NRC 1977a) and using more recent tide data. Because the applicant has followed guidance in RG 1.59 (NRC 1977a) and used more recently available tide data to specify an initial water surface elevation for the SLOSH model simulation, the staff concluded that the applicant's approach for estimating the PMSS maximum water surface elevation is appropriate. The applicant found that the two methods yielded values that were close, with the larger being 0.82 m (2.68 ft) NAVD88. The applicant used this larger value for subsequent analysis. The applicant determined an initial water level for use with the SLOSH model. The applicant's initial water level was 1.18 m (3.87 ft) NAVD88, which is based on an initial rise of 0.18 m (0.60 ft), a long-term sea level rise of 0.18 m (0.59 ft), and the 10 percent exceedance tide of 0.82 m (2.68 ft) NAVD88. The applicant stated that its initial water level was slightly larger than the one used previously (1.16 m [3.82 ft] NAVD88). The applicant applied the SLOSH model with the revised initial water elevation and found it has an insignificant effect on the SLOSH model predictions for the case producing the maximum surge elevation previously reported. The applicant reported a maximum surge elevation of 14.53 m (47.7 ft) NAVD88. The staff concluded that the applicant has adequately addressed the PMSS maximum stillwater surface elevation. The staff's evaluation of issues related to wave action is described below.

2.4.5.4.3 Wave Action

Information Submitted by the Applicant

The applicant estimated that the limiting wave period would be approximately 10 seconds assuming a deep water depth of 10 m (32.8 ft). The applicant also assumed the ground surface elevations would vary between 1.5 and 4.6 m (5 and 15 ft) and the storm surge elevations would vary from 6.1 to 10.7 m (20 to 35 ft). The applicant carried out 1,000 wave setup estimations from randomly selected combinations of ground surface and storm surge elevations. The applicant selected the maximum of these 1,000 simulated wave setups, 2.3 m (7.65 ft), as the wave setup value for the LNP site. The applicant stated that the surge boundary remains to the west of U.S. Highway 19, which is approximately 6.4 km (4 mi) from the LNP site. The applicant concluded, therefore, that the temporary increase in water level was highly unlikely to reach the LNP site.

The applicant reported the total water depth as the sum of Stillwater depth and wave setup. The applicant performed 1,000 simulations for the total water depth by combining the random selection of storm surge parameters and the wave setup parameters. The maximum of the 1,000 applicant-estimated total water depths was 14.93 m (48.98 ft) NGVD29 or 14.62 m (47.98 ft) NAVD88.

NRC Staff's Technical Evaluation

The staff requested additional information regarding the methodology used in the analysis of coincident wind-generated wave action and runup in **RAI 02.04.05-07**, which states:

> To meet the requirements of GDC 2, 10 CFR 52.17, and10 CFR Part 100, an estimate of wind-induced wave runup under PMH winds is needed. Criteria and methods of the USACE, as generally summarized in the USACE Coastal Engineering Manual, are used as a standard to evaluate the applicant's estimate of coincident wind-generated wave action and runup. These criteria are also used to evaluate flooding, including the static and dynamic effects of broken, breaking, and nonbreaking waves. Please add a reference in the FSAR for the methodology used to estimate wave action in Lake Rousseau, or explain why such a reference is not needed.

The applicant responded to the staff's RAI 02.04.05-07 in a letter dated July 20, 2009 (ML092030128). The applicant stated that due to the narrow and irregular shape of Lake Rousseau, the fetch length in the lake would be too short to generate a wave that would affect the LNP site. As stated above, the staff determined the meteorologically or seismically generated waves in Lake Rousseau would be limited by fetch and by water depth and would not reach the LNP site.

To ensure that the applicant has considered wave runup during PMH storm surge flooding, the staff issued **RAI 02.04.05-08**, which states:

> To meet the requirements of GDC 2, 10 CFR 52.17, and 10 CFR Part 100, an estimate of wind-induced wave runup under PMH winds is needed. The applicant added the estimated wave setup to the estimated stillwater PMH storm surge to obtain total water depth at the LNP site during the PMH conditions. Please provide an estimate of wave runup during the PMH storm surge at the LNP site.

The applicant responded to the staff's RAI 02.04.05-08 in a letter dated July 20, 2009 (ML092030128). The applicant provided an estimate of wave runup under PMH conditions using the procedures described by the USACE CEM (Scheffner 2008). The applicant estimated that the maximum wave runup would be 0.26 m (0.85 ft). The applicant stated that the FSAR would be updated to include the runup analysis.

The staff reviewed the applicant's response to RAI 02.04.05-08 and its calculations to determine that the applicant has used the USACE CEM (Scheffner 2008) guidance for estimation of wave runup during PMH conditions. The staff determined that the USACE CEM (Scheffner 2008) guidelines are widely used in engineering practice and are suitable for use in estimation of site characteristics for an FSAR. The staff finds that the applicant appropriately considered wave runup during PMH conditions at the LNP site.

To determine whether the applicant has followed an approach that is consistent with the regulatory guidance in National Weather Service Report 23, the staff issued **RAI 02.04.05-11**, which states:

> In RAI 2.4.5-10, the staff requested the applicant to provide supplemental information; the staff stated that the applicant must (1) use a set of plausible probable maximum hurricane (PMH) scenarios consistent with the National Oceanic and Atmospheric (NOAA) National Weather Service (NWS) Report 23 (NWS 23) as input to a currently accepted storm surge model (such as NWS Sea, Lake, and Overland Surges from Hurricanes [SLOSH]), (2) use initial open-water conditions that are consistent with current understanding of long-term sea-level rise and are valid for the life of the proposed plants, (3) provide estimates of coincident wind-wave runup, (4) provide maps of highest probable maximum storm surge (PMSS) water surface elevation at and near the LNP sites, and (5) provide updates to FSAR Section 2.4.5, including descriptions of data, methods, model setup, PMH scenarios and how they are consistent with NWS 23, treatment of uncertainty in the analysis, and available margins.
>
> The applicant responded to RAI 2.4.5-10 on January 27, 2011. The staff's review of the applicant's response to RAI 2.4.5-10 has raised the following issues:
>
> (4) Regulatory Guide (RG) 1.59 recommends that the following components of PMSS be estimated: (a) probable maximum surge (wind and pressure setups), (b) 10 percent exceedance tide, and (c) initial rise (forerunner or sea-level anomaly). The wind wave runup also needs to be added to obtain the PMSS. The applicant did not use an initial rise in its SLOSH simulations. RG 1.59 recommends an initial rise of 0.6 ft for Crystal River, FL. Because the value of initial water surface can have nonlinear effects on SLOSH predictions, 10 percent exceedance tide, initial rise, and long-term sea level rise should be combined to specify the initial water surface in SLOSH for simulation of the PMH scenarios.
>
> In a subsequent teleconference, the applicant stated its interpretation of RG 1.59 recommendations. The applicant stated that RG 1.59 recommends use of initial rise as an additional component of the initial water level if the 10 percent exceedance tide is estimated from predicted tides. The applicant stated that use of initial rise is not necessary because its approach used observations of tidal water levels that already contain the effects of initial rise.

(5) The applicant has not used the US Army Corps of Engineers Coastal Engineering Manual (CEM) for estimation of coincident wind wave activity. The CEM approach is recommended in SRP 2.4.5 as the currently accepted practice. The applicant did not provide justification why it used another approach. In a subsequent teleconference, the applicant stated that they did in fact use the CEM approach to estimate wind wave activity although this fact was not clearly stated in the response to RAI 2.4.5-10.

(6) The applicant states that the chosen PMSS maximum water surface elevation value for the LNP site is 49.52 ft NAVD88, not the higher estimate of 49.78 ft NAVD88 obtained from the SLOSH PMSS simulations. The PMSS maximum water surface elevation of 49.52 ft NAVD88 reported in the FSAR was obtained using an approach that the staff disagreed with previously. Also, the applicant added long-term sea-level rise and initial rise estimates after estimating the PMSS; this approach would not account for the nonlinear effects of initial water surface elevation on the PMSS.

The NRC staff requests the following additional information:

(4) The staff reviewed the applicant's approach to estimation of initial water level for a hydrodynamic storm surge model. The staff also reviewed RG 1.59, tidal data at the Cedar Key tide gauge, and NOAA's description of predicted tides. The staff determined that NOAA estimates harmonic constants at reference tide stations that are used to predict the harmonic component of tidal variations at the reference stations. Observed tide water levels also include the effects of wind wave activity and initial rise. Both of these additional effects manifest as random variations added to the harmonic component of the tidal variations. Because these random variations are independent of the harmonic forcings (mainly gravitational forces of the sun and the moon) and therefore can occur at any time, there is no assurance that "high" random variations of tides would be in phase with the highs of the predicted tides. Therefore, estimating the 10 percent exceedance tide from raw tide water level observations can result in underestimation of the initial water level (represented by 10 percent exceedance of predicted tides plus initial rise). RG 1.59 does not describe how initial rise reported for various locations in Appendix C of RG 1.59 was estimated.

The staff needs the following information to complete its review of the PMSS at the LNP site:

a. A detailed description of the applicant's approach used to estimate the initial water level for use in the SLOSH model runs, an analysis of how this approach is consistent with the recommendations of RG 1.59, a statement of the difference in the numerical values of the initial water level obtained by the applicant's approach and that recommended by RG 1.59, and a detailed justification of why the difference between the two numerical values would result in an insignificant difference in the PMSS maximum water surface elevation at the LNP site, or

b. An updated PMSS maximum water surface elevation at the LNP site that is a combination of (i) maximum stillwater elevation from a SLOSH simulation carried out with an initial water surface elevation estimated following the guidelines of RG 1.59 and using more recent tide data and (ii) wind wave effects using the CEM approach (see (2) below).

(5) Provide an update to FSAR text that clearly describes how the CEM approach was used to estimate wind wave activity coincident with PMSS maximum water surface elevation at the LNP site.

(6) Provide updates to FSAR that describe appropriately selected PMSS characteristics at the LNP site. Provide a discussion of available margins between the DCD Maximum Flood Level site parameter (the design grade elevation or the DCD plant elevation of 100 ft) and the highest PMSS water surface elevation accounting for coincident wind-wave activity.

The applicant responded to the staff's RAI 02.04.05-11 in a letter dated June 21, 2011 (ML11175A300). The applicant's response to part (1) of the staff's request and the staff's review of the applicant's response to part (1) are described above in Section 2.4.5.4.2 of this SER.

To address part (2) of the staff's request, the applicant used the Automated Coastal Engineering Systems (ACES) software to compute wave action at the LNP site. The applicant states that the software is designed to use the methods outlined in the USACE CEM (Scheffner 2008). The applicant states that due to the shallowness of water at the LNP embankment and the high wind conditions the waves at the LNP site will break. The applicant then uses breaking-wave calculations to estimate wave runup. The applicant estimated a wind-wave setup of 0.18 m (0.6 ft). Using the SLOSH-predicted PMSS maximum water elevation of 14.5 m (47.7 ft) NAVD88 combined with the wind setup of 0.18 m (0.6 ft), the applicant estimated that the water depth at the toe of an affected structure located at a grade elevation of 14.3 m (47.0 ft) NAVD88 would be 0.4 m (1.3 ft). The applicant used USACE CEM (Scheffner 2008) guidance the water depth to compute a wave period of 1.96 seconds and, along with the wave-breaking assumption, estimated a maximum wave height of 0.3 m (1.0 ft). The applicant found that for these conditions, ACES yielded a 0.45–m (1.48–ft) maximum wave runup. The applicant stated that updates to the FSAR based on the approach outlined in the RAI response will be made. The staff concluded that the applicant has adequately addressed the issue related to the estimation of PMH wind-wave action at the site. The staff is tracking future FSAR updates as **Confirmatory Item 2.4.5-1**.

Resolution of Confirmatory Item 2.4.5-1

Confirmatory Item 2.4.5-1 is an applicant commitment to update Section 2.4.5 of its FSAR. The staff verified that LNP COL FSAR Section 2.4.5 was appropriately updated. As a result, Confirmatory Item 2.4.5-1 is now closed.

The applicant responded to part (3) of this request with a discussion of the available margin between the DCD maximum flood level and the maximum estimated PMH surge level. The applicant stated that the maximum flood level as the sum of the maximum PMH surge level (14.54 m [47.7 ft] NAVD88), the initial rise (0.18 m [0.6 ft]), and the maximum wave runup (0.45 m [1.48 ft]) or 15.17 m (49.78 ft) NAVD88. The applicant stated that the LNP DCD plant elevation is 15.54 m (51 ft) NAVD88, leaving a margin of 0.37 m (1.22 ft).

The staff reviewed the methods used by the applicant in estimation of the maximum PMSS water surface elevation and concluded that it is acceptable because the applicant has used current guidance supplemented with more recently available data and used conservative assumptions. Therefore, the staff has determined that the applicant has adequately addressed the effects of the PMH on the water surface elevation at the LNP site.

2.4.5.4.4 Resonance

Information Submitted by the Applicant

The applicant stated that adverse effects from resonance in Lake Rousseau and the Gulf of Mexico on safety-related SSCs at the LNP site appear to be unlikely because the resonance will be quickly dissipated.

NRC Staff's Technical Evaluation

The staff reviewed the applicant's response to RAI 02.04.05-06 to evaluate the effects of resonance in Lake Rousseau and any induced flood wave that may travel from the lake towards the LNP site. As stated above, the staff determined the meteorologically or seismically generated waves set up in Lake Rousseau would be limited by fetch and by water depth and would not reach the LNP site. The staff considers RAI 02.04.05-06 to be resolved.

2.4.5.4.5 Protective Structures

Information Submitted by the Applicant

The applicant stated that all safety-related SSCs are protected from adverse effects of water up to an elevation of 51 ft NAVD88, which is higher than the design basis flood at the LNP site.

NRC Staff's Technical Evaluation

The staff evaluated the highest floodwater elevations during PMH conditions resulting from storm surge, wave setup, and wave runup to determine if all safety-related SSCs are adequately protected after the review of the applicant's responses to RAIs 02.04.05-09, 02.04.05-10, and 02.04.05-11. The staff has accepted the applicant's conclusion that the design-basis flood elevation at the LNP site is caused by a PMH and results in a combined effects maximum water surface elevation of 15.17 m (49.78 ft) NAVD88, which is lower than the LNP site grade elevation of 15.24 m (50 ft) NAVD88 and the corresponding DCD plant elevation of 15.54 m (51 ft) NAVD88 with an available margin of 0.37 m (1.22 ft).

The staff has completed its review of the maximum water surface elevations near the LNP site after the applicant's PMH analysis was completed as documented by the responses to RAIs 02.04.05-09, 02.04.05-10, and 02.04.05-11. Therefore, the staff considers these RAIs to be resolved.

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 COL FSAR related to this section.

As set forth above, the applicant has presented and substantiated information to establish the site description. The staff has reviewed the information provided and, for the reasons given above, concludes that the applicant has provided sufficient details about the site description for the staff to determine, as documented in Section 2.4.5, of this SER, that the applicant has met the relevant requirements of 10 CFR 52.79(a)(1)(iii) and 10 CFR Part 100 with respect to determining the acceptability of the site. This addresses part of COL information item 2.4-2.

2.4.6 Probable Maximum Tsunami Hazards

2.4.6.1 *Introduction*

The probable maximum tsunami hazards are addressed to ensure that any potential tsunami hazards to the SSCs important to safety are considered in plant design. The specific areas of review are as follows: (1) historical tsunami data, including paleotsunami mappings and interpretations, regional records and eyewitness reports, and more recently available tide gauge and real-time bottom pressure gauge data, (2) probable maximum tsunami (PMT) that may pose hazards to the site, (3) tsunami wave propagation models and model parameters used to simulate the tsunami wave propagation from the source towards the site, (4) extent and duration of wave runup during the inundation phase of the PMT event, (5) static and dynamic force metrics, including the inundation and drawdown depths, current speed, acceleration, inertial component, and momentum flux that quantify the forces on any safety-related SSCs that may be exposed to the tsunami waves, (6) debris and water-borne projectiles that accompany tsunami currents and may impact safety-related SSCs, (7) effects of sediment erosion and deposition caused by tsunami waves that may result in blockage or loss of function of safety-related SSCs, (8) potential effects of seismic and non-seismic information on the postulated design bases and how they relate to tsunami in the vicinity of the site and the site region, and (9) any additional information requirements prescribed within 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 COL FSAR addresses the site-specific information about potential dam failures. The applicant addressed the information as follows:

AP1000 COL Information Item

- LNP COL 2.4-6

In addition, this section addresses the following COL-specific information identified in Section 2.4.1.2 of Revision 19 of the AP1000 DCD.

> Combined License applicants referencing the AP1000 design will address the following site-specific information on historical flooding and potential flooding factors, including the effects of local intense precipitation.
>
> - Probable Maximum Flood on Streams and Rivers – Site-specific information that will be used to determine design basis flooding at the site. This information will include the probable maximum flood 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 of flood level.

2.4.6.3 *Regulatory Basis*

The relevant requirements of the Commission regulations for the identification of tsunami floods, tsunami flood design considerations and the associated acceptance criteria, are described in Section 2.4.6 of NUREG-0800 (NRC 2007a).

The applicable regulatory requirements for identifying the effects of tsunami flooding are:

- 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.

- 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.

Appropriate sections of the following RGs are used by the staff for the identified acceptance criteria:

- RG 1.59, "Design Basis Floods for Nuclear Power Plants" (NRC 1977a), as supplemented by best current practices; and

- RG 1.102, "Flood Protection for Nuclear Power Plants" (NRC 1976b).

2.4.6.4 *Technical Evaluation*

The NRC staff reviewed Section 2.4.6 of the LNP 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 probable maximum tsunami hazards. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

2.4.6.4.1 Probable Maximum Tsunami

Information Submitted by the Applicant

Because the applicant did not include a summary of the PMT assessment in Section 2.4.6.1 of the FSAR, information from other sections of the FSAR was used to determine which sources were considered and what the applicant determined were the water levels associated with each source. Three tsunami source regions were considered by the applicant to determine the PMT: (1) far-field sources outside the Gulf of Mexico and Caribbean region, (2) seismogenic sources along the Caribbean plate boundary, and (3) earthquake and landslide tsunami sources in the Gulf of Mexico. For the far-field sources, the applicant appears to consider that the maximum wave height would be from an event similar to the 1755 Lisbon seismogenic tsunami (<1 m wave heights in the Gulf of Mexico). For Caribbean sources, the worst-case scenario is determined by the applicant to be a seismogenic tsunami offshore Venezuela (in the Caribbean Sea), with a maximum wave height of 0.65 m offshore of the site (FSAR pg. 2.4-58). For Gulf of Mexico tsunami sources, the applicant considered the East Breaks slump in the northwest Gulf of Mexico as the worst-case scenario, with a maximum wave height of 1.68 m offshore of the site (FSAR pg. 2.4-53). The applicant stated that the controlling source of the PMT appears to be the East Breaks landslide.

To obtain clarification on the most reasonably severe geo-seismic activity possible and corresponding tsunami analysis, the staff issued **RAI 02.04.06-01**, asking the applicant for a summary of the PMT assessment for the Levy County site, including the controlling source for the PMT and corresponding tsunami water level determination. The applicant responded to the staff's RAI 02.04.06-01 in a letter dated July 22, 2009 (ML092080077). The applicant refers to the responses of RAI 02.04.06-08 and 02.04.06-10, suggesting that the Mississippi Canyon slide is the controlling source for the PMT. The PMT runup indicated in the response to RAI 02.04.06-01 does not agree with either the uncorrected or corrected PMT runup values indicated in the applicant's responses to RAI 02.40.6-06 (Tables 1 and 2), RAI 02.04.06-08 (Table 3), and RAI 02.04.06-10 (Table 1).

The applicant responded to the staff's RAI 02.04.06-11 in a letter dated March 25, 2010. The applicant states that the PMT runup and run-in values for a Mississippi Canyon-like slide moving down slope at a velocity of 50 m/s (164 ft/s) were incorrectly presented as 23.5 m (77.1 ft) NAVD88 and 2.19 km (1.36 mi), respectively. The correct PMT runup and run-in values are 22.5 m (73.8 ft) NAVD88 and 2.07 km (1.29 mi), respectively, as presented in the response to RAI 2.4.6-10 (Table 1). The associated LNP COL in FSAR Subsection 2.4.6, Rev. 1 was revised to incorporate clarification of the PMT analysis and text presented in LNP calculation package LNG-0000-X7C-043, Revision 0. The correct PMT runup and run-in values presented above was also included in this revision. Therefore, the staff considers RAIs 02.04.06-01 and 02-04-06-11 to be resolved.

To obtain information on the generation of tsunami-like waves from hill-slope failures and the stability of the coastal area, the staff issued **RAI 02.04.06-02**, asking the applicant to provide a discussion of the generation of tsunami-like waves from hill-slope failures and the stability of the coastal area in the updated FSAR with reference to the findings in Section 2.5 of the FSAR. The applicant responded to the staff's RAI 02.04.06-02 in letters dated July 22, 2009 (ML092080077) and August 09, 2010 (ML102290085). The applicant stated that no permanent slopes or hill slopes are present near the site or within the coastal areas near the site. Therefore, the staff considers RAI 02.04.06-02 to be resolved.

NRC Staff's Technical Evaluation

The NRC Staff conducted an independent confirmatory analysis to determine the PMT at the Levy County site that is described in detail in the sections that follow. In summary, numerical hydrodynamic modeling of three different types of tsunami sources have been performed to determine their impact on the Levy County site. The three source types are (1) distant earthquake sources; (2) a regional earthquake source in the Gulf of Mexico; and (3) regional submarine landslide sources in the Gulf of Mexico. Most of the analysis is focused on source type (3) for determination of the PMT. For all conditions, the most conservative source parameters were employed, even when arguably unphysical, to provide an absolute upper limit on the possible tsunami effects at the Levy County site.

The Staff found that the applicant did not use any of the standard methods of tsunami propagation and inundation modeling. In RAI 2.4.6-08, the staff requested additional information regarding the applicant's analysis procedure used to calculate tsunami wave height and period at the site, including the theoretical bases of the models, their verification and the conservatism of all input parameters. In a letter dated July 22, 2009, the applicant describes a procedure in which an estimated source amplitude is multiplied by three factors: (1)propagation loss, (2) shoaling correction, and (3) "beaching" amplification. Each of the multiplicative factors is determined from analytic expressions—variations in water depth along the propagation path between the source and the site were not explicitly accounted for. The results of their analysis indicate that the PMT is from a Mississippi Canyon landslide source, with a maximum water level of 21.4 m (Response to RAI 02.04.06-8). Including sea-level rise, sea-level anomaly, and high tide, their PMT maximum water level is 22.5 m (NAVD88) (Response to RAI 02.04.06-10), substantially above the plant grade elevation of 15.5 m (NAVD88).

Using conservative source parameters and neglecting the radial spreading of wave energy, the staff's 1HD simulations indicate that the Mississippi Canyon source clearly has the greatest potential to bring at large wave to the Levy site, with 1HD water elevations near the site in excess of +30 m. The staff's 2HD simulations of this source and the WORST CASE Florida Slope landslide source that include radial spreading predict a maximum wave elevation of 7 m offshore of the site (30 m water depth). However, the Mississippi Canyon wave is longer in period and has a longer train of large waves, and thus is designated as the PMT for the Levy site. The staff's highly refined nearshore simulations show that this source results in a maximum water level of +3 m. Because of nonlinear effects during wave propagation, one cannot simply add an antecedent sea level that includes 10 percent exceedance high tide, sea level anomaly, and sea-level rise to this maximum water to the +3m maximum water level. A separate simulation that includes the nonlinear propagation effects and a +1.2 m (NAVD88) antecedent sea level results in a maximum water level of +6.1 m. Thus, the results from the staff's independent analysis indicate that the PMT does not reach the Levy site plant grade elevation. Therefore, the staff considers RAI 2.4.6-8 to be resolved.

2.4.6.4.2 Historical Tsunami Record

Information Submitted by the Applicant

The applicant reviews tsunami catalogs for the Caribbean and the Gulf of Mexico regions and determines that there were three events that affected the Gulf coast: two seismogenic tsunamis and one seismic seiche. The sources of information primarily include the NOAA/NGDC Historical Tsunami Database (internet) and the published report of Lander et al. (2002).

The first seismogenic tsunami was caused by the 1918 Mona Passage earthquake, located northwest of Puerto Rico. Maximum runup from the tsunami was reported to be 6 m local to the source. In the Gulf of Mexico, the tsunami was recorded at the Galveston tide gauge station, but the maximum amplitude of the wave was not indicated by the applicant.

The second seismogenic tsunami was caused by an earthquake near Vieques Island in 1922. In the Gulf of Mexico, a maximum amplitude of 0.6 m was recorded at the Galveston tide gauge station, with a dominant period of 45-minutes.

A seiche was observed in the Gulf of Mexico in 1964 that was set up by seismic waves emanating from the 1964 Gulf of Alaska earthquake. The applicant did not indicate the maximum amplitude of the seiche in the Gulf of Mexico.

To obtain clarification with respect to the historical tsunami record, the staff issued RAIs 02.04.06-03, 02.04.06-04 and 02.04.06-05. In RAI 02.04.06-03, the staff asked the applicant to provide clarification in the updated FSAR of the meaning of the descriptor "impact" as used on pg. 2.4-45 of the FSAR: "…historically no Caribbean tsunami has *impacted* the United States Gulf Coast." The applicant responded to the staff's RAI 02.04.06-03 in letters dated July 22, 2009 (ML0920800771), and August 09, 2010 (ML1022900851). The applicant explains in their response that the descriptor "impact" means "no tsunamis are known to have originated in the Caribbean Sea and generated a runup exceeding 1.0 m at any location along the United States Gulf Coast." Therefore, the staff considers RAI 02.04.06-03 to be resolved.

The staff issued RAI 02.04.06-04 to provide clarification in the updated FSAR whether any of the Maximum Water Height measurements listed in FSAR Table 2.4.6-202 are located in the Gulf of Mexico. The applicant responded to the staff's RAI 02.04.06-04 in a letter dated July 22, 2009 (ML0920800771). The applicant indicates that none of the locations of Maximum Water Height measurements are located in the Gulf of Mexico. It should be noted that the Maximum Water Height measurements are typically located near the source—not necessarily in the Caribbean as the applicant indicates in their response to RAI 2.4.6-04. Therefore, the staff considers RAI 02.04.06-04 to be resolved.

The staff issued RAI 02.04.06-05, asking the applicant to provide clarification in the updated FSAR whether there is any geologic evidence of tsunami deposits at the Levy County site or at nearby regions. Additionally, indicate whether there are geologically conducive locations for the deposition and preservation of tsunami deposits in the vicinity of the Levy County site. If such paleo-tsunami evidence exists, indicate how they are distinguished from storm wash-over deposits. The applicant responded to the staff's RAI 02.04.06-05 in a letter dated July 22, 2009 (ML0920800771). The applicant indicates that site-specific borings lead them to conclude that there is no geologic evidence of paleo-tsunami or tsunami-like deposits in the vicinity of the Levy County site. However, the applicant needs to provide additional details of the sedimentological analysis used to arrive at this conclusion, including the thickness of sand layers that the methods used were capable of detecting, and cross reference to applicable parts of FSAR Section 2.5. The applicant responded to the staff's RAI 02.04.06-12 in a letter dated March 25, 2010 (ML100910299), with additional details of the sedimentological analysis. Based on the applicant's detailed response, the staff considers RAIs 02.04.06-05 and 02.04.06-12 to be resolved.

NRC Staff's Technical Evaluation

The Staff reviewed the applicant's primary references of historical observations and measurements of tsunami and seismic seiche waves occurring along the Gulf Coast and finds the applicant's assessment of the historical tsunami record to be acceptable.

The closest locations of interpreted paleotsunami deposits to the Levy County site are in southern Alabama, as shown in FSAR Figure 2.4.6.4.2-1. The deposits are thought to be part of a regional tsunami event in the Gulf of Mexico at or near the time of the Cretaceous-Tertiary (K-T) boundary.

The common interpretation of this deposit is that it was emplaced by a tsunami generated from Chicxulub asteroid impact, owing to its date and the existence of impact ejecta at the Brazos site and elsewhere. However, the tsunami deposit was discovered by Bourgeois et al. (1988) prior to the discovery of the Chicxulub impact crater (Hildebrand and others, 1991). An important alternate hypothesis related to possible tsunamigenic sources in the Gulf of Mexico is provided by Bourgeois et al. (1988):

> "If the tsunami were produced by a major submarine landslide, it should not occur precisely at the K-T boundary unless the landslide were caused by an earthquake related to boundary events, which is a possibility" (pg. 569)

Bourgeois et al. (1988) suggested that a tsunami wave 50-100 m high was necessary to explain this deposit. The published wave heights and flow speeds of the Brazos tsunami deposit are reasonable, representing order-of-magnitude estimates. It is not conceivable that the wave that created these deposits was generated by any landslide source that would be of relevance to the present-day PMT determination. As the staff demonstrates in independent analysis, any landslide wave generated at the present-day continental shelf break would not be able to maintain a large wave height across such a long propagation distance over very shallow water. The depth-limiting dissipation effect, in which large amplitude waves are dissipated much faster than small amplitude waves during long propagation over shallow depth, would necessarily reduce any landslide generated wave located at the shelf break to a minimal event at the shoreline. It is still possible that this deposit was generated by a paleo-landslide source, but this landslide event would have been local to the Brazos site. It is considerably more likely that a wave of the estimated height would be caused by a relatively nearby large impact event. Waves emanating from such a source would have the needed extreme wave heights and long periods to be able to propagate significant wave energy this far inland.

Over the last 20 years, the Brazos deposit has been extensively sampled from out crops and subsurface cores at sites near the banks of the Brazos River. Recently, studies have both corroborated and disputed whether the Brazos deposit was emplaced by a tsunami, whether it occurred exactly at the geologic boundary between the Cretaceous and Tertiary periods (i.e., at the K-T boundary), and whether the trigger was the Chicxulub impact (e.g., Smit and others, 1996; Gale, 2006; Schulte and others, 2006; Keller and others, 2007). Conflicting interpretations of the deposits at the southern Alabama locations are described in earlier studies (Mancini and others, 1989; Liu and Olsson, 1992; Savrda, 1993; Keller and Stinnesbeck, 1996). The exact age and hydrologic process that formed the regional tsunami deposit remain controversial. However, in light of these studies over the last 20 years, the lead author of original study identifying the deposit maintains that it was emplaced by a tsunami (J. Bourgeois, pers. comm., 2009).

The Staff examined primary references of historical observations and measurements of tsunami and seismic seiche waves occurring along the Gulf Coast were examined.

The applicant did not provide evidence that an adequate investigation was conducted for tsunami deposits at or near the proposed site. Additionally, the applicant does not consider the existence of a possible paleotsunami (Bourgeois and others, 1988) that occurred along the ancient Gulf Coast shoreline, including locations in southern Alabama. The common interpretation of this deposit is that it was emplaced by a tsunami generated by the Chixulub impact or by landslide or earthquake activity associated with the impact. Although arguments have been presented against this interpretation, this deposit, along with the historical record, should be considered as possible evidence of tsunami occurrence along the Gulf Coast. However, the staff finds that the flow speeds and wave heights inferred from the deposit are not relevant to determination of the present-day PMT.

2.4.6.4.3 Source Generator Characteristics

Information Submitted by the Applicant

The applicant identifies possible tsunami sources from three general regions: (1) far-field sources outside of the Gulf of Mexico and Caribbean Sea, (2) the Caribbean plate boundary, and (3) inside the Gulf of Mexico.

Far-field source scenarios initially considered include the 1964 Gulf of Alaska seismic seiche, the 1755 Lisbon seismogenic tsunami, and far-field landslide sources in the Atlantic Ocean. The applicant appears to consider only the 1755 Lisbon seismogenic in determining water levels from a far-field source.

Caribbean sources include earthquakes along the boundary of the Caribbean plate. Specific earthquake and tectonic segments considered by the applicant include the North Panama Deformation Belt, the northern South America convergence zone, the northern Caribbean subduction zone, and the Cayman transform fault system.

Gulf of Mexico tsunami sources considered include intra-plate earthquakes and landslides. For intra-plate earthquakes, the applicant indicates the historical occurrence of the Mw=5.8 September 10, 2006 Gulf of Mexico earthquake, but does not include a seismogenic source in this region of the Gulf of Mexico in their tsunami analysis. The applicant does include the results from a scenario by Knight (2006) offshore Veracruz, Mexico, that the applicant links to present-day seismic activity. For landslides in the Gulf of Mexico, the applicant primarily considers the East Breaks landslide offshore Texas, but not other possible landslide sources in the Gulf of Mexico. All of the aforementioned information was obtained by the applicant from published journal articles and web sites.

In order to obtain a more comprehensive picture of tsunami source generators, the staff issued RAIs 02.04.06-06 and 02.04.06-07. In RAI 02.04.06-06, the staff asked the applicant to provide a discussion in the updated FSAR of submarine landslides in the Gulf of Mexico, other than East Breaks, as potential tsunami generators, including the Mississippi Canyon landslide, and landslides along the Florida Escarpment and along the slope above the Florida Escarpment. In addition, clarify text in the FSAR indicating whether the East Breaks landslide is considered as the PMT source, in relation to discussion of the north Venezuela seismogenic tsunami as having "the most severe impacts for the Gulf Coast" (pg. 2.4-58).

The applicant responded to the staff's RAI 02.04.06-06 in a letter dated July 22, 2009 (ML0920800771). In their response to RAI 02.04.6-06, the applicant is inconsistent in their characterization of the Mississippi Canyon and Florida Escarpment tsunami sources. On page 9-10 of their response, the applicant appears to discount the tsunami potential based on the date of the last landslides in those regions. In the rest of their response, they indicate that these sources are used for PMT determination (and, in fact, the Mississippi Canyon slide is the applicant's controlling PMT source). The applicant needs to clarify whether the Mississippi Canyon and Florida Escarpment are considered to be significant potential sources for PMT determination. In addition, the applicant indicates identical source parameters for "Florida Escarpment" and "Slope above the Florida Escarpment" in Table 1 of their response to RAI 02.04.6-06. However, the water depth in these two regions is different. The applicant needs to explain this apparent discrepancy, or justify why the entries in Table 1 are correct. The applicant responded to the staff's RAI 02.04.06-13 in a letter dated March 25, 2010 (ML1009102991), with additional details and a revised Table 1. Based on the applicant's detailed response and FSAR revision, the staff considers RAIs 02.04.06-06 and 02.04.06-13 to be resolved.

The staff issued RAI 02.04.06-07, asking the applicant to provide clarification in the updated FSAR regarding seismologic characterization of the region offshore Veracruz, Mexico, relative to the generation of tsunamis. The applicant responded to the staff's RAI 02.04.07 in a letter dated July 22, 2009 (ML0920800771). The applicant's explanation provides additional details of the source parameters considered, although the staff is not aware of 15-20 earthquakes > M7 near Veracruz Mexico. The applicant needs to clarify the location of "15-20 earthquakes of magnitude 7 or greater…near Veracruz" indicated in the applicant's response to RAI 02.04.06-07, in terms of tsunami potential for the Gulf of Mexico versus the Pacific Ocean. The applicant should also provide the information source for this statement. The staff issued RAI 02.04.06-14 to obtain additional information related to the "15-20 earthquakes of magnitude 7 or greater…near Veracruz" described in the applicant's response to RAI 02.04.06-07. The applicant responded to the staff's RAI 02.04.06-14 in a letter dated March 25, 2010 (ML1009102991), with additional geo-seismic descriptions of controlling distant tsunami generators, including location, source dimensions, fault orientation, and maximum displacement. Based on the applicant's detailed response, which conforms to the guidance in section C.I.2.4.6.3 of RG 1.206, the staff considers RAIs 02.04.06-07 and 02.04.06-14 to be resolved.

NRC Staff's Technical Evaluation

In this section, tsunami sources used for the independent confirmatory analysis are described in terms of their identification, characteristic, and tsunami generation parameters. Potential tsunamigenic sources are first discussed below, including parameters associated with the maximum submarine landslides in the Gulf of Mexico. At the end of this section, we briefly discuss seismic seiches.

Potential tsunami sources that are likely to determine the PMT at the Levy County site are submarine landslides in the Gulf of Mexico. Subaerial landslides, volcanogenic sources, near-field intra-plate earthquakes and inter-plate earthquakes along Caribbean plate boundary faults are unlikely to be the causative tsunami generator for the PMT at the Levy County site as discussed below.

With regard to subaerial landslides, there are no major coastal cliffs near the site that would produce tsunami-like waves that exceed the amplitude of those generated by other sources.

Volcanogenic Sources

According to the Global Volcanism Program of the Smithsonian Institution (http://www.volcano.si.edu/), there are three general regions of volcanic activity that have the potential to generate localized wave activity in the Gulf of Mexico and Caribbean Sea: (1) two Mexican volcanoes near the Gulf of Mexico coastline; (2) two volcanoes in the western Caribbean; and (3) volcanic activity along the Lesser Antilles island arc. Two Mexican volcanoes, (Cerro el Abra/Los Atlixos and San Martin) associated with the eastern Trans-Mexican Volcanic Belt, are located near the Gulf of Mexico coastline. Basaltic flows associated with Los Atlixcos have reached as far as the coast. Also in the eastern Caribbean, Volcán Azul on the coast of Nicaragua is composed of three small cinder cones, but these are unlikely to generate significant failures. There are many active volcanoes along the Lesser Antilles island arc, some of which have historically caused local tsunamis (Pelinovsky and others, 2004). However, catastrophic failures associated with volcanoes along the eastern coasts of Mexico and Central American are either too far inland or too small in size to generate significant wave activity in the Gulf of Mexico near the Levy County site. Based on existing evidence, volcanoes along the Lesser Antilles or in the eastern Atlantic Ocean are too far away and/or unfavorably situated to generate significant wave activity in the Gulf of Mexico.

Intra-Plate Earthquakes

Because there are no tectonic plate boundaries in the Gulf of Mexico region, earthquakes *local* to the Levy County site occur in an intra-plate tectonic environment, limiting the maximum magnitude these earthquakes can attain. According to the documentation for the 2008 update of the United States National Seismic Hazard Maps (Petersen and others, 2008), the maximum magnitude (M_{max}) for the Florida Gulf coast is estimated to be approximately M_{max}=7.5. See Wheeler (2009) and Mueller (2010) for further details. Because the maximum slip, and consequently the maximum sea floor displacement, associated with an earthquake scales with its magnitude, the initial tsunami wave amplitude associated with an intra-plate earthquake would therefore be less than that used for local, submarine landslides under the conservative hot-start conditions as described in Section 2.4.6.4.5. Empirical evidence from global earthquakes indicates that the maximum local tsunami runup from M_w=7.5 earthquakes is approximately 6 m (Geist, 2002). This maximum is related to an earthquake along an island arc (Kuril Islands) without a broad continental shelf.

Inter-Plate Earthquakes

In the far-field, offshore tsunami amplitudes from Carribbean inter-plate earthquakes are estimated in Chapter 8 of ten Brink and others (2008), using the linear-long wave equations. The description of major plate boundary faults and specific source parameters are described in that study. The tsunami propagation model presented in ten Brink and others (2008) has been refined during our confirmatory analysis for two of the principal sources (the northern South America Convergent Zone and the northern Caribbean Subduction Zone) using the COMCOT tsunami model discussed in Sections 2.4.6.4.4 and 2.4.6.4.5. Tsunami amplitudes at the Florida Gulf coast from these seismogenic sources are generally small (i.e., < 1 m) compared to tsunami amplitudes determined for submarine landslides in establishing the PMT. Tsunami amplitudes from earthquakes along the Azores-Gibraltar oceanic convergence boundary are also likely to be small (i.e., < 1 m) in the Gulf of Mexico (Mader, 2001; Barkan and others, 2009). For the remainder of this section, we focus on submarine landslide sources as the principal generator for the PMT at the Levy County site.

Submarine Landslides in the Gulf of Mexico

Submarine landslides in the Gulf of Mexico are considered a potential tsunami hazard for the Levy County site for several reasons: (1) some dated landslides in the Gulf of Mexico have post-glacial ages (Coleman and others, 1983), suggesting that triggering conditions for these landslides are still present, (2) the size and shallow initiation depth of landslides in the Gulf of Mexico, and (3) analysis of recent seismicity suggest the presence of small-scale energetic landslides in the Gulf of Mexico.

With regard to (1), the Mississippi Canyon landslide is dated 7,500-11,000 years before present (ybp) (Coleman and others, 1983; Chapter 3 in ten Brink and others, 2007) and the East Breaks landslide is dated 15,900 ± 500 ybp (Piper and Behrens, 2003). Both landslides, which are among the largest landslides in the Gulf of Mexico, occurred after the end of the last glacial maximum, during post-glacial transgression. Although landslide activity along the passive margins of North America may be decreasing with time since the last glacial period, the 1929 Grand Banks landslide is a historic example of such an event that produced a destructive tsunami (Fine and others, 2005). In addition, the Mississippi River continues to deposit large quantities of water-saturated sediments on the continental shelf and slope, making them vulnerable to over-pressurization and slope failure.

With regard to (2), several submarine landslide characteristics have been found to be significant in determining tsunami generation potential of the landslide, headwall depth including landslide volume, initial acceleration of the slide mass, and slide velocity (Ward, 2001; Harbitz and others, 2006). The volume of failed material for each of several of the landslides in the Gulf of Mexico (see below) and the shallow headwall depths (< 300 m) of the East Breaks and Mississippi Canyon landslides suggest that these landslides had the potential to generate tsunamis.

Finally, with regard to (3), seismograms of an event that occurred on February 10, 2006 (i.e., the Green Canyon event, FSAR Figure 2.4.6.4.3-2) that occurred offshore southern Louisiana (Dewey and Dellinger, 2008) suggest that energetic landslides continue to occur in the Gulf of Mexico (Nettles, 2007). Most landslides affected by salt tectonics are small in size (e.g., in comparison to the East Breaks landslide; Chapter 3 of ten Brink and others, 2007) and unlikely to be tsunamigenic. However, in terms of the failure duration, the 2006 event must have occurred rapidly enough to have generated seismic energy. While source analyses of this event cannot definitively distinguish between a fault and landslide source and evidence of significant sediment failure has not yet been found (Dellinger and Blum, 2009) this event reveals the potential for present-day slope failure.

Maximum Submarine Landslides

The NRC Staff defines four provinces in the Gulf of Mexico that are likely to be the origin of submarine landslides that control the determination of the PMT. Three additional provinces defined in Chapter 3 of ten Brink and others (2007) are not likely to be sites of major tsunamigenic landslides. The four provinces defined for PMT analysis are the Florida Escarpment and Slope region (immediately off the Levy County site), Mississippi Canyon, Northwest Gulf of Mexico, and Campeche Escarpment and Slope. The Northwest Gulf of Mexico is a mixed canyon/fan and salt province consisting of terrigenous and hemipelagic sediment, the Mississippi Canyon a canyon/fan province consisting of terrigenous and hemipelagic sediment and the Campeche and Florida margins are carbonate provinces formed from reef structures and characterized by having steep slopes. Above these escarpments a broad gentle slope comprised of carbonate sediment separates the escarpments from the shelf.

The primary landslide parameters that are used in the tsunami models include the excavation depth and slide width, which can be directly measured from sea floor mapping of the largest observed slide in the four geologic provinces. The other necessary parameter is downslope landslide length, interpreted from the runout distance. The runout distance measured from sea floor mapping is a combination of fast plug flow (low viscosity, non-turbulent), creeping plug flow (high viscosity/viscoplastic, non-turbulent) and turbidity currents (turbulent boundary layer fluid). The latter two likely have little to no tsunami-generating potential. Also, turbidity currents often involve entrainment of material during flow, such that the deposition volume may be greater than the excavation volume. Finally, hydroplaning may increase the runout of submarine landslides. The landslide lengths indicated below are intended to represent the main tsunami-generating phase. The amplitude of the initial negative wave above the excavation region is linked to the maximum excavation depth. The amplitude of the initial positive wave above the deposition region is determined from a conservation of landslide volume. The excavation volume can be well determined using GIS techniques (see below). Setting the deposition volume equal to the excavation volume, the positive amplitude is determined for a given landslide length. For a fixed volume, increasing the landslide length decreases the initial positive amplitude of the landslide tsunami.

Landslide volume calculations are based on measuring the volume of material excavated from the landslide source area using a technique similar to that applied by ten Brink and others (2006) and Chaytor and others (2009). Briefly stated, the approach involves using multibeam bathymetry to outline the extent of the excavation area, interpolating a smooth surface through the polygons that define the edges of the slide to provide an estimate of the pre-slide slope surface, and subtracting this surface from the present seafloor surface.

The maximum observed landslide from multibeam surveys is taken as the maximum landslide for a given region. It may be possible that larger landslides could occur in a given region, however this determination of the maximum landslide is consistent with the overall definition of PMT as "the most severe of the natural phenomena that have been historically reported or determined from geological and geophysical data for the site and surrounding area". In this case, the maximum landslide is taken from geologic observations spanning tens of thousands of years. Moreover, because landslide volumes appear to follow a power-law or log-normal distribution (ten Brink and others, 2006; Chaytor and others, 2009), there may be no mathematical or physical constraints on the definition of the theoretical maximum landslide (other than the dimensions of the entire continental slope). These calculations were only completed for part of the East Breaks landslide, the Mississippi Canyon landslide, and a landslide from the slope above the Florida Escarpment. No calculations were made for failures above the Campeche Escarpment because currently available bathymetric data are inadequate.

East Breaks Landslide

Geologic Setting: River delta that formed at the shelf edge during the early Holocene

Post Failure Sedimentation: Landslide source area appears to be partially filled (predominantly failure deposits with some post-failure sedimentation)

Age: 10,000 – 25,000 years (Piper, 1997; Piper and Behrens, 2003)

Maximum Single Event (East Breaks landslide): Maximum and minimum parameters are taken from different interpretations of the digitized failure scar surrounding the excavation region (Chaytor and others, 2009).

Volume	Area	Width	Length	Excavation Depth	Runout Distance
Max: 21.95 km^3 Min: 20.80 km^3	519.52 km^2 420.98 km^2	~ 12 km	~ 50 km	~160 m	91 km

Run out distance: 91 km from end of excavation and 130 km from headwall based on GLORIA mapping (Rothwell and others, 1991) (See FSAR Figure 2.4.6.4.3-7). Multibeam bathymetry is not available for the entire run-out area

Trabant and others (2001) have reported volumes of 50-60 km^3 and a run-out distance of 160 km. Trabant and others (2001) derived their volume estimate from the size of debris lobes in the deposition region, using a 3D seismic reflection dataset that is proprietary. The staff cannot confirm their result for that reason and because we lack the necessary bathymetry coverage that far downslope to identify the extent of the debris lobes. Debris lobes are often the result of multiple events that are difficult to distinguish (Chaytor and others, 2009; Twichell and others, 2009) and may include sediment entrainment during flow. Our volume estimate above is for the amount excavated at the source (within the landslide scarp) and is more representative of a single failure event.

Mississippi Canyon

Geologic Setting: River delta and fan system

Age: 7,500 to 11,000 years (Coleman and others, 1983; Chapter 3 in ten Brink and others, 2007)

Maximum Single Event

Volume	Area	Excavation Depth	Runout Distance
425.54 km^3	3687.26 km^2	~300 m	297 km

Other reported volumes are1500-2000 km^3 (Coleman and others, 1983). As with the East Breaks landslide, this estimate is from landslide deposits that most likely represent multiple failure episodes. The volume given above is the staff's best estimate of a maximum single-event volume.

Florida Escarpment and Slope

Geologic Setting: The slope above the edge of a carbonate platform

Post Failure Sedimentation: None visible on multibeam images or on available high-resolution seismic profiles (Twichell and others, 1993).

Age: Early Holocene or older (Doyle and Holmes, 1985). Because the deposits from these carbonate failures accumulate along the base of the Florida escarpment are buried by Mississippi Fan deposits, they are older than the youngest fan deposits dated at about 11,500 years old.

Maximum Single Event

Volume	Area	Excavation Depth	Runout Distance
16.2 km^3	647.57 km^2	~150 m but quite variable	Uncertain.

Runout distance: The landslide deposit is at the base of the Florida Escarpment buried under younger Mississippi Fan deposits.

Campeche Escarpment

Geologic Setting: Carbonate platform

One of the persistent issues during the independent confirmatory analysis is acquiring sufficient geologic information about the Campeche Escarpment with which to estimate the maximum landslide parameters as with the other Gulf of Mexico landslide provinces. Plans to conduct multibeam bathymetry surveys are pending. Presently, there is no published information showing the detailed bathymetry or distribution of landslides on or above the Campeche Escarpment.

Seismic Seiches

Seismic seiches are fundamentally a different type of wave than tsunamis. Rather than being impulsively generated by displacement of the sea floor, seismic seiches occur from resonance of seismic surface waves (continental Rayleigh and Love waves) within enclosed or semi-enclosed bodies of water. The harmonic periods of the oscillation are dependent on the dimensions and geometry of the body of water. In 1964, seiches were set up along the Gulf Coast from seismic surface waves emanating from the M=9.2 Gulf of Alaska earthquake. The efficiency at which the seiches occurred at great distance from the earthquake is primarily explained by amplification of surface wave motion from the thick sedimentary section along the Gulf Coast (McGarr, 1965). Because the propagation path from Alaska to the Gulf Coast is almost completely continental (McGarr, 1965) and because the magnitude of the 1964 earthquake is close to the maximum possible for that subduction zone (e.g., Bird and Kagan, 2004), it is likely that the historical observations of 1964 seiche wave heights are the maximum possible and less than the PMT amplitudes from landslide sources.

In summary, the NRC Staff list the following findings of our independent confirmatory analysis of the tsunami source characteristics:

- There is sufficient evidence to consider submarine landslides in the Gulf of Mexico as a present-day tsunami hazard for the purpose of defining the PMT at the Levy County Site.

- Four landslide provinces are defined in the Gulf of Mexico that are applicable for determining the PMT: Northwest Gulf of Mexico, Mississippi Canyon, slope above the Florida Escarpment, and Campeche Escarpment.

- Parameters for the maximum submarine landslide were determined for each of the provinces, except for the Campeche Escarpment where we are awaiting additional data.

- It is likely that seismic seiche waves resulting from the 1964 Gulf of Alaska earthquake are nearly the highest possible, owing to a predominantly continental ray path for seismic surface waves from Alaska to the Gulf Coast, However, they are smaller than the PMT amplitudes from submarine landslides in the region.

2.4.6.4.4 Tsunami Analysis

Information Submitted by the Applicant

The applicant's tsunami analysis primarily consists of using past studies to ascertain the tsunami propagation characteristics from the three source regions discussed in Section 2.4.6.3 to estimate tsunami amplitudes offshore of the Levy County Nuclear Plant site. Different types of tsunami analyses were used to estimate tsunami water levels for each of the three source regions.

For tsunami sources located in the far-field, the applicant only considers a source with characteristics similar to the 1755 Lisbon tsunami in their tsunami analysis. To determine tsunami amplitudes in the Gulf of Mexico from this far-field earthquake, the applicant cites the results of Mader (2001). The applicant indicates that Mader (2001) uses the nonlinear long wave equations and a 10-minute bathymetric grid to calculate tsunami amplitudes.

For tsunami sources located in the Caribbean region, the applicant cites analysis of open-ocean propagation presented by Knight (2006) (FSAR reference 2.4.6-225) and the USGS Administrative Report (2007) describing tsunami sources affecting U.S. Atlantic and Gulf Coasts (FSAR reference 2.4.6-214). The tsunami analysis method used by Knight (2006) is not indicated by the applicant. The Caribbean sources used in the analysis by Knight (2006) include earthquakes along the northern Caribbean subduction zone (i.e., the "Puerto Rico Trench" as termed by Knight, 2006), a source possibly related to the Cayman transform fault system (i.e., the "Swan fault" offshore Cancun, Mexico as termed by Knight, 2006), and the northern South America convergence zone (incorrectly called the "North Panama Deformed Belt" by Knight (2006) and by the applicant). The tsunami analysis method used in the USGS Administrate Report (2007) is a finite-difference approximation to the linear-long wave equations. Tsunami propagation across the continental shelf and tsunami runup were not modeled in this study. The Caribbean sources used in the USGS (2007) analysis as indicated by the applicant include earthquakes along the northern Caribbean subduction zone, the Cayman transform fault system, the North Panama Deformation Belt, and the northern South America convergence zone.

For tsunami sources located in the Gulf of Mexico region, the applicant considers both earthquake and landslide sources. Although intra-plate sources in the vicinity of the Mw=5.8 September 10, 2006 Gulf of Mexico earthquake are not further considered for tsunami analysis by the applicant, an offshore Veracruz tsunami scenario from Knight (2006) is considered, which the applicant links to intra-plate seismicity. As with the Caribbean tsunami sources where the applicant cites the work of Knight (2006), the applicant does not indicate the tsunami analysis method used for the Veracruz tsunami scenario. For landslide sources in the Gulf of Mexico, the applicant uses a tsunami attenuation function (FSAR equation 2.4.6-1) derived by Zahibo et al. (2003) (FSAR reference 2.4.6-222) for tsunamis originating in the Caribbean region. The theoretical basis for this attenuation function and evidence of its applicability for tsunamis in the Gulf of Mexico is not included in the FSAR. The applicant uses a Monte Carlo analysis to establish the maximum wave height near the Levy County Nuclear Plant from this attenuation function.

In order to obtain a complete description of the analysis procedure used to calculate tsunami wave height and period at the site, including the theoretical bases of the models, including the applicant's verification and the conservatism of all input parameters, the staff issued RAIs 02.04.06-08 and 02.04.06-09. In RAI 02.04.06-08, the staff asked the applicant to provide theoretical basis, assumptions (e.g., source parameterization), and applicability to the Levy County site for the tsunami attenuation function discussed on pg. 2.4-53 (Equation 2.4.6-1) and make available the details of the Monte Carlo analysis used to estimate the maximum wave height and where the maximum wave height estimate is geographically located. In addition, for this and other methods of tsunami analysis indicated in the FSAR, provide the procedure use to calculate tsunami propagation, runup, and inundation (i.e., tsunami water levels) at the Levy County site from offshore tsunami amplitude.

The applicant responded to the staff's RAI 02.04.08 in letters dated July 22, 2009 (ML0920800771) and August 10, 2010 (ML1022900851). The applicant provided a substantial new effort regarding analysis for tsunami generation, propagation, and runup. However, there are several unresolved issues in the applicant's response: (1) the formulas for source amplitude are poorly documented (they are not contained in Silver et al., 2009); (2) water depths listed in Table 1 seem arbitrary (its 300-800 m for East Breaks); (3) it is unclear how source "diameter" is determined; (4) there are typographic errors in the numbers for the Veracruz and Venezuela source diameters (Table 4); (5) the assumption that "wave amplitude onshore cannot exceed its estimated runup height at shore," is incorrect but this may be an issue with the terminology; and (6) variable *Co* in equations 17 and 18 is undefined. The applicant needs to provide additional details regarding the method for tsunami analysis in reference to the aforementioned items. In RAI 02.04.06-15, the staff requested additional information related to these six unresolved issues.

The applicant responded to the staff's RAI 02.04.06-15 in a letter dated March 25, 2010, with additional details. However, the revised equations are now incorrect, according to the most recent review article of Ward (2010). The staff issued RAI 02.04.06-16, asking the applicant to provide additional details regarding the new methodology for tsunami analysis described in response to RAI 02.4.06-08 and RAI 02.04.06-15. This discussion should specifically include: (1) the basis for source amplitude formulae; (2) clarify what is meant by "wave amplitude onshore cannot exceed its estimated runup height at shore" (statement is incorrect using standard tsunami terminology); and (3) definition of variable C_o in equations 17 and 18. The applicant responded to the staff's RAI 02.04.06-16 in a letter dated November 30, 2010 (ML1034206451). The application of the equations and understanding of the assumptions and approximations behind the method were still incorrect.

The staff issued RAI 02.04.06-17, asking the applicant to provide the following:

> An analysis of the PMT event using a technically sound and conservative approach such as those predicted by a site and region specific model approach applicable to tsunami waves to calculate tsunami water levels at or near the site. Such a model avoids approximations of source geometry, bathymetry between the source and offshore of site, and topography near the site inherent in the applicant's current approach. For example, shallow water wave equation models (COMCOT, ComMIT. Delft3D) and Boussinesq-type Models (COULWAVE, FUNWAVE, Geowave) for earthquake and earthquake/landslide/ impact generated tsunamis, respectively.
>
> If a numerical model is used, provide a clear presentation of all equations used, discussion of assumptions inherent in these equations and the associated conservatism, and the procedure to calculate the water-level values. Please provide all input data sources, calculation packages, and any associated modeling input files.

(a) If the existing approach which relies on the Ward et al publication is used, proper usage of these methods must be checked, and a complete presentation of the theoretical assumptions, as relevant to propagation modeling of a landslide-generated wave and runup/inundation, should be provided. The applicant must provide site-specific justification as to why the Ward (2010) equations are applicable and conservative for the Levy site. This would typically involve presenting the theoretical assumptions behind the generation, attenuation, shoaling, and runup equations, and why these assumptions are valid and conservative with respect to site-specific conditions. Specifically:

> Tsunami Generation: (1) Provide the reference for wave amplitude Equation 2.4.6-3, along with relevant assumptions used to develop that equation. (2) Provide references for the expressions of slide velocity and a clear indication as to which expressions were used to calculation the slide velocities listed in FSAR Table 2.4.6- 206. (3) Provide the rationale and justification for using Equation 2.4.6-8 derived for impact tsunami sources to model landslide tsunamis, particularly with regard to difference in wave characteristics between landslide and impact tsunamis. (4) Explain how diameter listed for each source in FSAR Table 2.4.6-206 relates to landslide parameters.
>
> Tsunami Propagation: (1) Explain how the "measurement point" is chosen to determine R, the distance of measurement point from the source. (2) Because the "measurement point" is a nearshore location, justify the use of Equation 2.4.6-11 that is derived for constant water depth, considering the broad continental shelf

offshore western Florida. (3) If in a revised procedure applicant applies the propagation and shoaling terms at the edge of the continental shelf, provide an expression for propagation across the continental shelf. (4) The equation for the attenuation curves (2.4.6-8) is miss-cited. Provide the correct reference, domain of applicability of these fitted curves, and assumptions used to derive these curves.

Tsunami Runup: (1) Definition of *h* in Equation 2.4.15 is inconsistent with the definition indicated in FSAR References 2.4.6-228 and 2.4.6-237, from which this equation was taken. In the revised FSAR, applicant indicates that *h* represents "shoreline wave height" whereas it is intended to represent runup as described in the aforementioned References. Provide clarification of the use of Equation 2.4.15. (2) Provide the theoretical assumptions behind the equation 2.4.15, and why these assumptions are valid and conservative with respect to site-specific conditions. (3) If revised Equation 2.4.15 is used to calculate runup, confirm that revised section 2.4.6.6.3.5 is not necessary. (4) Provide the geographic location (lat, long) and water depth where the shoaled amplitude A(R) in FSAR Table 2.4.6-207 is calculated. (5) Provide location information for revised figure 2.4.6-230 "Landward Topographic Profile", for example, in a map figure.

The applicant responded to the staff's RAI 02.04.06-17 in letters dated February 28, 2011, April 19, 2011, and July 14, 2011. Using the FUNWAVE-TVD tsunami model, the applicant provided a detailed, site-specific, technically sound and conservative approach to calculate tsunami propagation, runup, and inundation (i.e., tsunami water levels) at the Levy County site, including proposed FSAR revisions. Therefore, the staff considers RAI 02.04.06-08, RAI 02.04.06-15, RAI 02.04.06-16 and RAI 02.04.06-17 to be resolved.

The staff issued RAI 02.04.06-09, asking the applicant to provide clarification in the updated FSAR to resolve the inconsistency of the statement that the Gulf of Mexico contains no sources of reverse faults (1st sentence, section 2.4.6.4.1.2, pg. 2.4-52) given the mechanism of the September 10, 2006 Mw=5.8 in the NE Gulf of Mexico (third sentence). The applicant responded to the staff's RAI 02.04.09 in a letter dated July 22, 2009 (ML0920800771). The applicant clarifies that they meant to indicate that there are no subduction zone faults in the Gulf of Mexico, without adding specific explanation for the possibility of intra-plate reverse faults, such as the September 20, 2006 earthquake. Therefore, the staff considers RAI 02.04.06-09 to be resolved.

NRC Staff's Technical Evaluation

Numerical simulations of tsunami propagation have made great progress in the last thirty years. Several tsunami computational models are currently used in the National Tsunami Hazard Mitigation Program, sponsored by the National Oceanic and Atmospheric Administration, to produce tsunami inundation and evacuation maps for the states of Alaska, California, Hawaii, Oregon, and Washington. The computational models include MOST (Method Of Splitting Tsunami), developed originally by researchers at the University of Southern California (Titov and Synolakis, 1998); COMCOT (Cornell Multi-grid Coupled Tsunami Model), developed at

Cornell University (Liu and others, 1995); and TSUNAMI2, developed at Tohoko University in Japan (Imamura, 1996). All three models solve the same depth-integrated and 2D horizontal (2DH) nonlinear shallow-water (NSW) equations with different finite-difference algorithms. There are a number of other tsunami models as well, including the finite element model ADCIRC (ADvanced CIRCulation Model For Oceanic, Coastal And Estuarine Waters) (e.g., Myers and Baptista, 1995).

Earthquake generated tsunamis, with their very long wavelengths, are ideally matched with NSW for transoceanic propagation. Models such as Titov & Synolakis (1995) and Liu et al. (1995) have been shown to be reasonably accurate throughout the evolution of a tsunami, and are in widespread use today. However, when examining the tsunamis generated by submarine mass failures, the NSW can lead to significant errors (Lynett and others, 2003). The length scale of a submarine failure tends to be much less than that of an earthquake, and thus the wavelength of the created tsunami is shorter. To correctly simulate the shorter wave phenomenon, one needs equations with excellent shallow to intermediate water properties, such as the Boussinesq equations. While the Boussinesq model too has accuracy limitations on how deep (or short) the landslide can be (Lynett and Liu, 2002), it is able to simulate the majority of tsunami generating landslides. Thus, for the work proposed here, the Boussinesq-based numerical model COULWAVE (Lynett and Liu, 2002) will be used. (See Appendix for reprints of peer-reviewed papers that form the foundation of COULWAVE.) This model solves the fully nonlinear extended Boussinesq equations on a Cartesian grid. COULWAVE has the capability of accurately modeling the wind waves with both nonlinear and dispersive properties. A particular advantage of the model is the use of fully non-linear equations for both deep and shallow water. This avoids the common problem of "splitting" the analysis when the wave reaches shallow water. Applications for which COULWAVE has proven very accurate include wave evolution from intermediate depths to the shoreline, including parameterized models for wave breaking and bottom friction. For technical details on wave propagation, breaking, runup, inundation, and overtopping of sloping structures see Geist et al., (2009) (including the references).

In response to **RAI 02.04.06-17**, the applicant models a tsunami from the Mississippi Canyon landslide using a FUNWAVE. FUNWAVE is a phase-resolving, time-stepping Boussinesq model for ocean surface wave propagation in the nearshore. For confirmatory analysis, the NRC staff used a higher-order Boussinesq hydrodynamics model (COULWAVE), which is more specifically suited to landslide tsunamis. As described above, the staff considers **RAI 02.04.06-17** to be resolved**.**

2.4.6.4.5 Tsunami Water Levels

Information Submitted by the Applicant

The various methods of tsunami analysis used by the applicant to estimate tsunami water levels at the Levy County Nuclear Plant site are described at the beginning of Section 2.4.6.4.4. Most of the water level estimates are taken directly from previously published studies. The exception is the analysis for the East Breaks landslide in the Gulf of Mexico, where the applicant uses a tsunami attenuation function and Monte Carlo analysis to establish the maximum water level.

The applicant provided the following table summarizing the water level estimates for each of the sources considered:

Location	Mechanism	Magnitude	Offshore Wave Height	Estimated Runup	Validation of Source as Potential Tsunami Generator	Analysis Reference
West Cayman oceanic transform fault (also known as Swan Island fault)	Earthquake	Mw 8.35	13 cm (5.1 in)	39 cm (15.4 in)	Bird (2003)	USGS (2007)
East Cayman fault (also known as Oriente fault)	Earthquake	Mw 8.45	12 cm (4.72 in)	36 cm (14.2 in)	Bird (2003)	USGS (2007)
Northern Puerto Rico/Lesser Antilles	Earthquake	Mw 8.84	14 cm (5.5 in)	42 cm (16.5 in)	Bird (2003)	USGS (2007)
North Panama deformation belt	Earthquake	Mw 8.28	25 cm (9.8 in)	75 cm (29.5 in)	Bird (2003)	USGS (2007)
North Venezuela subduction zone	Earthquake	Mw 8.5	65 cm (25.6 in)	195 cm (76.8 in)	Bird (2003)	USGS (2007)
Puerto Rico trench (66W, 18N)	Earthquake	Mw 9.0	25 cm (9.8 in)	75 cm (29.5 in)	Bird (2003)	Knight (2006)
Caribbean Sea (85W, 21N) (translated from the Swan fault to mouth of Gulf near Cancun)	Earthquake	Mw 8.2	30 cm (11.8 in)	90 cm (35.4 in)	Bird (2003)	Knight (2006)
North Panama Deformed Belt (66W, 12N)	Earthquake	Mw 9.0	15 cm (5.9 in)	45 cm (17.7 in)	Bird (2003)	Knight (2006)
Gulf of Mexico, offshore of Veracruz (95W, 20N)	Earthquake	Mw 8.2	35 cm (13.8 in)	105 cm (41.3 in)	hypothetical	Knight (2006)
East Breaks Slump	Landslide	50 to 60 cubic kilometers (km^3)	1.68 m (5.5 ft)	5.04 m (16.5 ft)	Trabant (2001); tsunami claim not further supported	Trabant (2001), Zaibo (2003)

As indicated previously, the “North Panama Deformed Belt” is incorrectly identified by Knight (2006) and the applicant and is not the same region defined as the North Panama deformation belt by USGS (2007). Knight’s (2006) “North Panama Deformed Belt” source is geographically located along the northern South America convergence zone (also known as the north Venezuela subduction zone). The “Estimated Runup” values indicated in the applicants table above were determined by applying an amplification factor of 3 to the “Offshore Wave Height” values, as indicated by the applicant during the site audit. Not included in this table is the applicant’s Gulf of Mexico offshore wave height estimate of “less than one meter” from the 1755 Lisbon far-field seismogenic tsunami (Mader, 2001) as cited on pg. 2.4-55 of the FSAR. It is unclear whether high tide and long-term sea-level rise are included in determining these water levels.

The applicant indicates that the nominal plant grade elevation is 15.2 m (NAVD88) and therefore the water level from the Probable Maximum Tsunami will not impact safety-related facilities at the Levy County Nuclear Plant site.

In order to obtain a complete description of the ambient water levels assumed to be coincident with the tsunami, the staff issued RAI 02.04.06-10, asking the applicant to provide a discussion in the updated FSAR of the value for 10% exceedance high-tide and long-term sea-level rise coincident with maximum tsunami water levels at the Levy County site. The applicant responded to the staff’s RAI 02.04.10 in a letter dated July 22, 2009 (ML0920800771). The applicant provided details of high spring tide, sea-level anomaly and sea-level rise in the calculation of

PMT water levels. Based on the applicant's response, the staff considers RAI 02.04.06-10 to be resolved.

NRC Staff's Technical Evaluation

Numerical modeling of three different types of tsunami sources has been performed to determine their impact on the Levy County site. The three source types are: (1) distant earthquake sources; (2) a regional earthquake source in the Gulf of Mexico; and (3) regional submarine landslide sources in the Gulf of Mexico. Most of the analysis described in this section is focused on source type (3) for determination of the PMT. For all conditions, the most conservative source parameters were employed, even when arguably unphysical, to provide an absolute upper limit on the possible tsunami effects at the Levy County site.

a. Distant Earthquake Sources

Regional tsunami propagation patterns in the Gulf of Mexico have been computed for a number of distant earthquake sources located in the Caribbean as reported in ten Brink et al. (2008). In Chapter 8 of that study, earthquake scenarios along five fault systems were examined: (1) west Cayman oceanic transform fault (OTF); (2) east Cayman OTF; (3) northern Caribbean subduction zone; (4) north Panama Oceanic Convergence Boundary; and (5) the northern South America convergent zone. In that report, tsunami propagation was modeled using the leap-frog, finite-difference approximation to the linear-long wave equations computed using Cartesian coordinates. Bottom friction, wave breaking, and runup were not modeled—computations were restricted to water depths of 250 m or greater. Results for the western Gulf of Mexico indicate that offshore tsunami amplitudes were less than 1.0 m for each earthquake scenario.

For comparative purposes, we re-compute here the offshore tsunami water levels for earthquake scenarios (3) and (5) using the COMCOT model. The COMCOT model is more accurate than the model used in ten Brink et al. (2008) since it includes non-linear terms in the propagation equations (hence, the computations can be carried into shallower water than in ten Brink et al., 2008), a moving boundary condition at the shoreline, and is computed in spherical coordinates. Bottom friction is also included, but is set at a low, conservative value ($f = 10^{-4}$) in this case.

These results confirm that tsunami amplitudes from distant Caribbean earthquakes are less than 1.0 m near the Levy County site. Tsunami amplitudes from earthquakes along the Azores-Gibraltar oceanic convergence boundary are also likely to be less than 1 m in the Gulf of Mexico (Mader, 2001; Barkan and others, 2009).

b. Regional Earthquake Sources

Regional tsunami propagation patterns in the Gulf of Mexico have been computed for a local earthquake near the location of the September 10, 2006 M=6.0 earthquake. For this scenario, probable maximum fault dimensions and slip similar to an M_{max}=7.5 earthquake (Petersen and

others, 2008; Wheeler, 2009; Mueller, 2010) was determined from the empirical scaling relationships for intra-plate earthquakes of Wells and Coppersmith (1994). Conservative values were allowed within 1 standard deviation of the empirical estimates of all fault types (empirical relationships for reverse faults only are not statistically reliable). This resulted in the following rupture parameters: length=150 km; width=30 km, average slip= 5m. The corresponding magnitude, assuming a shear modulus of 30 GPa, is M_w=7.8—slightly greater than M_{max}=7.5 because of the conservative assumptions. The geometric parameters of the earthquake were taken from the nodal plane of the September 10, 2006 *M*=6.0 earthquake that optimized the radiation of tsunami energy toward the site: dip = 47°; strike=346°; latitude=27.3°N; longitude 86.3°W.
The offshore tsunami water levels for this local earthquake scenario was computed using the COMCOT model as described for the distant earthquake sources above. Bottom friction is also included, but is set at a low, conservative value ($f = 10^{-4}$) in this case. In general, tsunami amplitudes from the local M_w=7.8 sources are larger than the distant M~9 earthquake sources, with peak tsunami amplitudes near 1 m. These amplitudes are significantly less than the tsunami amplitudes produced by the regional submarine landslide sources described below.

c. Regional Submarine Landslide Sources

Five different landslide tsunami sources in the GOM are investigated to determined their impact at the Levy site. First, all sources are simulated as one-horizontal-dimension (1HD) transects, and thus conservatively neglect radial spreading of wave energy. Additionally, each source is simulated with a wide range of frictional coefficients, from no friction to likely in-situ friction, to provide both an upper limit and a realistic estimate of the runup. From these 1HD simulations, the Mississippi Canyon source clearly has the greatest potential to bring at large wave to the Levy site, with 1HD water elevations near the site in excess of +30 m. This source and a local Florida Shelf landslide source are chosen for additional analysis by means of two-horizontal-dimension (2HD) simulations, where radial spreading is explicitly included. Interestingly, both of these sources predict a wave of similar maximum elevation at the 30 m depth offshore of the site, approximately 7 m. However, the Mississippi Canyon wave is longer in period and has a longer train of large waves, and thus is designated as the PMT for the Levy site. Highly refined nearshore simulations show that this source, even when including high tide and future sea level rise, does not produce a tsunami that reaches the Levy site ground elevation.

Numerical Grid Development

The bathymetry/topography grid required by the hydrodynamic model is created via three main sources: 1) the Smith and Sandwell (SS) 2-minute global elevation database; 2) a recent GOM grid created by the U.S. Army Corps of Engineers for use with the storm surge model ADCIRC; and 3) a blend of available bathymetry and topography for the west coast of Florida. Sources 2) and 3) are a combination of numerous databases including recent lidar surveys and digitized elevation maps. These two sources were used for bathymetry and topography at locations with bottom elevations greater than -500 m. For depths greater than this (or elevations lower), the SS was primarily used.

Figure 2.4.6.4.5-1 shows the entire GOM grid coverage, with the five tsunami landslide source locations outlined. The high level of detail in the full resolution image is not evident in this reproduced image, but the staff's review addressed the detailed GOM grid.

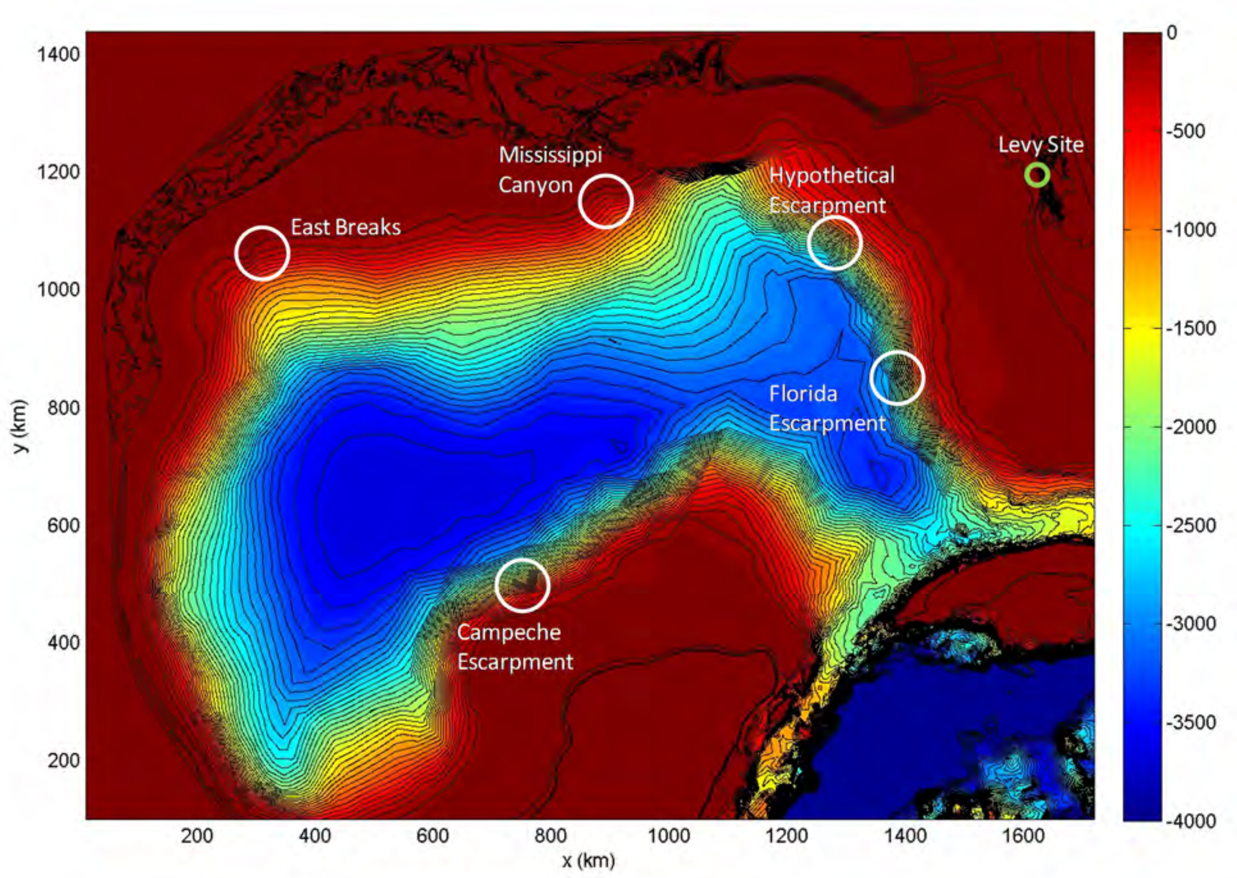


Figure 2.4.6.4.5-1. Bathymetry/topography contour surface of the GOM domain used for the tsunami hydrodynamic modeling. General locations of the five potential tsunami sources are shown by the white circles and the Levy site by the green circle. Bottom elevations are indicated by colors following the colorbar, with units in meters.

Initial Numerical Simulations – Physical Limits

The purpose of these initial simulations is to provide an absolute upper limit of the tsunami wave height that could be generated by the potential tsunami sources. Note that these limiting simulations use physical assumptions that are arguably unreasonable; the results of these simulations will be used to filter out tsunami sources that are incapable of adversely impacting the Levy site under even the most conservative assumptions. Specifically, these assumptions are:

1. Time scale of the seafloor motion is very small compared the period of the generated water wave (tsunami)
2. Bottom roughness, and the associated energy dissipation, is negligible in locations that are initially wet (i.e. locations with negative bottom elevation, offshore)

Assumption 1 simplifies the numerical analysis considerably. With this assumption, the free water surface response matches the change in the seafloor profile exactly. This type of approximation is used commonly for subduction-earthquake-generated tsunamis, but is known to be very conservative for landslide tsunamis (Lynett & Liu, 2002). The modeling simplification arises because need to include the landslide time evolution is removed. The initial pre-landslide bathymetry profile, as estimated by examination of neighboring depth contours, is subtracted by the post (existing) landslide bathymetry profile. This difference surface is smoothed and then used directly as a “hot-start” initial free surface condition in the hydrodynamic model.

Assumption 2 does not simplify the analysis significantly; however it does prevent the use of an overly high bottom roughness coefficient, which could artificially reduce the tsunami energy reaching the shoreline. Note that while the offshore regions are assumed to be without bottom friction, such an assumption is too physically unrealistic to accept for the inland regions where the roughness height may be the same order as the flow depth. For tsunami inundation, particularly for regions such as this project location where the wave would need to inundate long reaches of densely vegetated land to reach the site, inclusion of some measure of bottom roughness is necessary.

If any of these initial simulations indicate the need for more precise description of the source motion, such will be incorporated into a subsequent analysis. Source physics description and modeled motion will be given only if needed for this analysis. The most likely reason for needed higher precision would be if one of the initial simulation shows flooding at the site in exceedance of the PMF elevation determined elsewhere.

One-Horizontal Dimension (Transect) Simulations

First, one-horizontal-dimension (1HD) simulations are performed for all potential sources. The 1HD simulations require a small fraction of the CPU time of the 2HD runs, but do not include the radial spreading and refraction effects. Lack of radial spreading will lead to a conservative result in 1HD, while refraction can be either a constructive or destructive effect on the wave height, depending on the shallow water depth contours. 1HD simulations will provide an upper limit on the inundation distance and information on the relative importance of overland bottom friction, while the 2HD simulations provide insight into radial spreading and refraction. Results from the 1HD simulations will be used to filter all the sources down to a few possible candidates for the PMT; then a 2HD simulation will be run for each of these candidates.

East Breaks Landslide Source:

As provided in the landslide characterization section, the excavation depth of this slide is approximately 160 m. This length provides the trough elevation (i.e. -160 m) of the hot-start initial water surface condition. The horizontal dimensions of the slide source region are ~12 km in width and 50 km in length. With this information and knowledge of characteristic slide-generated waves taken from the literature (Lynett & Liu, 2002; Lynett & Liu, 2005), the hot-start initial condition is constructed.

1HD Results (No friction): The depth transect is taken from the source location directly to the Levy site. A constant spatial grid size of 200 m is used across the transect for the 1HD cases. Predictions from three 1HD simulations are given for **A)** no bottom friction, **B)** bottom friction due to moderate roughness characteristic of grass/turf (f=0.01), and **C)** bottom friction due to large roughness characteristic of the trees and dense shrub-like vegetation currently existing seaward of the Levy site (f=0.05). Note that the three different bottom friction values are only applied over initially dry land; for all simulations the initially submerged portions of the transect use no bottom friction.

In model simulations, the offshore evolution of the East Breaks wave can be seen with clearly dispersive effects, as shown by the long train of waves that reaches the Florida shelf. All of the simulations provide identical results for the tsunami prior to reaching the shoreline, as all the simulations start with the same wave, use the same bathymetry, and are frictionless offshore. This is most evident as the tsunami approaching the site.

1HD Results (Friction): As the wave starts inundating dry land, friction becomes important. The no-friction case A) shows a fast moving bore front that easily reaches the Levy site ground elevation, with maximum water surface elevations approaching +25 m at the site. Despite the modest friction value used in case B), here the tsunami wave front is slowed significantly but does reach the site, and maximum water elevations at the site are approximately +22 m. Finally, for case C), the large, realistic friction retards the flow considerably, and the tsunami wave front is stopped 3 km seaward of the site. Note that in all these figures, the horizontal and vertical scales are distorted, and that the realistic friction tsunami case still does manage to travel 15 km inland. A conclusion of this 1HD East Breaks study is that a tsunami approaching the site, with a bore height up to +12 m at the still water shoreline, will not adversely impact the site if the vegetation roughness is properly accounted for.

Campeche Landslide Source:

As noted in the landslide description section, there is no available data with which to constrain this source. In the absence of any quantitative guidance, it is assumed that a slide in this region will share geometric properties with the slope above the Florida Escarpment. As provided in the landslide characterization section, the excavation depth of this slide is approximately 150 m. This length provides the trough elevation (i.e. -150 m) of the hot-start initial water surface condition. The horizontal dimensions of the slide source region are assumed to be ~20 km in width and 50 km in length, inferred from the various scarps visible in the multibeam bathymetric data. With this information and knowledge of characteristic slide-generated waves taken from the literature (Lynett & Liu, 2002; Lynett & Liu, 2005), the hot-start initial condition is constructed.

1HD Results (No friction): The depth transect is taken from the source location directly to the Levy site. A constant spatial grid size of 200 m is used across the transect for the 1HD cases.

Predictions from three 1HD simulations are given for **A)** no bottom friction, **B)** bottom friction due to moderate roughness characteristic of grass/turf (f=0.01) , and **C)** bottom friction due to large roughness characteristic of the trees and dense shrub-like vegetation currently existing seaward of the Levy site (f=0.05). Note that the three different bottom friction values are only applied over initially dry land; for all simulations the initially submerged portions of the transect use no bottom friction.

In model simulations, the offshore evolution of the Campeche wave can be seen with clearly dispersive effects as shown by the long train of waves that reaches the Florida shelf. All of the simulations provide identical results for the tsunami prior to reaching the shoreline, as all the simulations start with the same wave, use the same bathymetry, and are frictionless offshore.

1HD Results (Friction): As the wave starts inundating dry land, friction becomes important and the results of the three simulations diverge. The no-friction case A) shows a fast moving bore front that easily reaches the Levy site ground elevation, with maximum water surface elevations approaching +23 m at the site. Despite the modest friction value used in case B), the tsunami wave front is slowed significantly but does reach the site, and maximum water elevations at the site are approximately +14 m. Finally, for case C), the large, realistic friction retards the flow considerably, and the tsunami wave front is stopped 15 km seaward of the site. Note that in all these figures, the horizontal and vertical scales are distorted, and that the realistic friction tsunami case still does manage to travel 15 km inland. A conclusion of this 1HD Campeche study is that a tsunami approaching the site, with a bore height up to +14 m at the still water shoreline, will not adversely impact the site if the vegetation roughness is properly accounted for.

Florida Slope Landslide Source:

As provided in the landslide characterization section, the excavation depth of this slide is approximately 150 m. This length provides the trough elevation (i.e. -150 m) of the hot-start initial water surface condition. The horizontal dimensions of the slide source region are assumed to be ~20 km in width and 50 km in length, inferred from the various scarps visible in the multibeam bathymetric data. With this information and knowledge of characteristic slide-generated waves taken from the literature (Lynett & Liu, 2002; Lynett & Liu, 2005), the hot-start initial condition is constructed.

1HD Results (No Friction): The depth transect is taken from the source location directly to the Levy site. A constant spatial grid size of 200 m is used across the transect for the 1HD cases. Predictions from three 1HD simulations are given for A) no bottom friction, B) bottom friction due to moderate roughness characteristic of grass/turf (f=0.01), and **C)** bottom friction due to large roughness characteristic of the trees and dense shrub-like vegetation currently existing seaward of the Levy site (f=0.05). Note that the three different bottom friction values are only applied over initially dry land; for all simulations the initially submerged portions of the transect use no bottom friction.

In the staff simulations, the large, nonlinear wave immediately steepens and forms a bore-front once on the shallow shelf. All of the simulations provide identical results for the tsunami prior to reaching the shoreline, as all the simulations start with the same wave, use the same bathymetry, and are frictionless offshore.

<u>1HD Results (Friction)</u>: As the wave starts inundating dry land, friction becomes important and the results of the three simulations diverge. The no-friction case A) shows a fast moving bore front that barely reaches the Levy site ground elevation, with maximum water surface elevations approaching +14 m at the site. With the modest friction value used in case B), the tsunami wave front is slowed significantly and does not reach the site. Finally, for case C), the large, realistic friction retards the flow considerably, and the tsunami wave front is stopped 25 km seaward of the site. A conclusion of this 1HD Florida Slope study is that a tsunami approaching the site, with a bore height up to +6 m at the still water shoreline, will not adversely impact the site if the vegetation roughness is properly accounted for.

It should also be noted that one of the reasons for the relatively small wave height produced by this source, as compared to the Campeche source, is the longer length of shelf that the wave must travel over before reaching the shoreline. With the Florida Slope transect, the shelf length is 150 km longer than that for the Campeche source. A second reason for a smaller tsunami, again as compared to Campeche, is the wave orientation. For a slide on the Florida shelf, the wave approaching Florida would have a leading depression. For a slide coming from Campeche, the wave approaching Florida would have a leading elevation. Once a leading depression wave is on the shelf, nonlinear effects will cause the trailing elevation wave to overrun and partially absorb the depression, equating to a decrease in the absolute elevation of the elevation wave front.

<u>Florida Slope WORST CASE Landslide Source:</u>

As mentioned in the previous Florida Slope section, the very long shelf length required by drawing the transect from the existing landslide source to the site might diminish the tsunami impacts considerably. In the section, a landslide source, identical to the Florida Slope, is hypothesized to exist immediately offshore of the Levy site. By minimizing the travel time to the coast and time over the shallow shelf, this simulation will provide an upper limit of the tsunami impact at the Levy site due to a Florida Slope-type slide anywhere along the west Florida shelf.

As provided in the landslide characterization section, the excavation depth of this slide is approximately 150 m. This length provides the trough elevation (i.e. -150 m) of the hot-start initial water surface condition. The horizontal dimensions of the slide source region are assumed to be ~20 km in width and 50 km in length, inferred from the various scarps visible in the multibeam bathymetric data. With this information and knowledge of characteristic slide-generated waves taken from the literature (Lynett & Liu, 2002; Lynett & Liu, 2005), the hot-start initial condition is constructed.

<u>1HD Results (No Friction)</u>: The depth transect is taken from the source location directly to the Levy site. A constant spatial grid size of 200 m is used across the transect for the 1HD cases. Predictions from three 1HD simulations are given for **A)** no bottom friction, **B)** bottom friction due to moderate roughness characteristic of grass/turf (f=0.01), and **C)** bottom friction due to large roughness characteristic of the trees and dense shrub-like vegetation currently existing

seaward of the Levy site (f=0.05). Note that the three different bottom friction values are only applied over initially dry land; for all simulations the initially submerged portions of the transect use no bottom friction.

In the offshore evolution of the Florida Slope wave, the large, nonlinear wave immediately steepens and forms a bore-front once on the shallow shelf. All of the simulations provide identical results for the tsunami prior to reaching the shoreline, as all the simulations start with the same wave, use the same bathymetry, and are frictionless offshore.

<u>1HD Results (Friction)</u>: As the wave starts inundating dry land, friction becomes important and the results of the three simulations diverge. The no-friction case A) shows a fast moving bore front that reaches the Levy site ground elevation, with maximum water surface elevations approaching +15 m at the site. With the modest friction value used in case B), the tsunami wave front is slowed significantly and does not reach the site. Finally, for case C), the large, realistic friction retards the flow considerably, and the tsunami wave front is stopped 15 km seaward of the site. A conclusion of this 1HD Florida Slope WORST CASE study is that a tsunami approaching the site, with a bore height up to +9 m at the still water shoreline, will not adversely impact the site if the vegetation roughness is properly accounted for. Despite the 50 percent larger nearshore wave elevation from the Florida Slope WORST CASE, as compared to the Florida Slope, the impact at the Levy site is not considerably different.

<u>Mississippi Canyon Landslide Source:</u>

As provided in the landslide characterization section, the excavation depth of this slide is approximately 300 m. However, this excavation, in the upper canyon, occurs near the shelf break, where the water depths away from the scarp are ~150 m. Thus the initial depression is set to the water depth at the head of the scarp, 150 m. The horizontal dimensions of the slide source region are assumed to be ~30 km in width and 160 km in length, inferred from the multibeam bathymetric data. With this information and knowledge of characteristic slide-generated waves taken from the literature (Lynett & Liu, 2002; Lynett & Liu, 2005), the hot-start initial condition is constructed.

<u>1HD Results (No Friction)</u>: The depth transect is taken from the source location directly to the Levy site. A constant spatial grid size of 200 m is used across the transect for the 1HD cases. Predictions from three 1HD simulations are given for **A)** no bottom friction, **B)** bottom friction due to moderate roughness characteristic of grass/turf (f=0.01), and **C)** bottom friction due to large roughness characteristic of the trees and dense shrub-like vegetation currently existing seaward of the Levy site (f=0.05). Note that the three different bottom friction values are only applied over initially dry land; for all simulations the initially submerged portions of the transect use no bottom friction.

In the offshore evolution of the Florida Slope wave the large, nonlinear wave immediately steepens and forms a bore-front once on the shallow shelf. All of the simulations provide identical results for the tsunami prior to reaching the shoreline, as all the simulations start with the same wave, use the same bathymetry, and are frictionless offshore.

1HD Results (Friction): The no-friction case A) shows a fast moving bore front that easily reaches the Levy site ground elevation, with maximum water surface elevations approaching +40 m at the site. Even with the modest friction value used in case B), the tsunami wave front is not slowed significantly and also easily reaches the site with water elevations of +33 m. Finally, for case C), the large, realistic friction retards the flow considerably, but still, the tsunami reaches the site, although the site is near the inundation limit. A conclusion of this 1HD Mississippi Canyon study is that a tsunami approaching the site, with a bore height up to +20 m at the still water shoreline, may impact the site. A more detailed, 2HD analysis of this site is clearly needed.

Two-Horizontal Dimension Simulations

From the 1HD simulations, it is possible to reduce the number of tsunami sources that need additional attention. The Mississippi Canyon source gives the largest heights at the shoreline, twice as large as the nearest source, and is also the closest non-Florida slope source to the site, so radial spreading effects should also be relatively minor for Mississippi Canyon. Thus, it can be reasonable expected that, if detailed 2HD simulation show that the Mississippi Canyon source has no impact at the site, then all other non-Florida slope sources (East Breaks, Campeche) can also be eliminated.

While it is likely that elimination of the Mississippi Canyon source as impacting the Levy site would also eliminate the Florida Slope WORST CASE source, because the Florida Slope WORST CASE is on the immediate shelf, radial spreading effects may not act to decrease the incoming wave height significantly. 2HD wave heights may be quite similar to those predicted by the 1HD simulation, which showed the tsunami reaching the site for the no-friction case. Therefore, two sources, Mississippi Canyon and Florida Slope WORST CASE, are discussed further in this SER.

Florida Slope WORST CASE Landslide Source

The slide and initial water surface condition properties for this source are described above in the corresponding 1HD section, but are given again here for completeness. As provided in the landslide characterization section, the excavation depth of this slide is approximately 150 m. This length provides the trough elevation (i.e. -150 m) of the hot-start initial water surface condition. The horizontal dimensions of the slide source region are assumed to be ~20 km in width and 50 km in length, inferred from the various scarps visible in the multibeam bathymetric data. With this information, and knowledge of characteristic slide-generated waves taken from the literature (Lynett & Liu, 2002; Lynett & Liu, 2005), the hot-start initial condition is constructed. A constant spatial grid size of 500 m is used in the numerical simulation.

The 2HD evolution, within 15 minutes from the landslide, it is clear that radial spreading effects are important offshore of the shelf, but on the shelf, where the wave is approaching the Levy site, this is not the case. Spreading is minor, and the wave energy remains in a laterally compact front. The elevation component of the landward traveling wave forms into a bore about 30 minutes after the slide and quickly overtakes the leading depression. The bore front height continues to diminish and by the time the front reaches a depth of about 30 m its elevation is approximately 7 m. Note that for the 1HD simulation, the wave height at this depth was 10 m, a relatively minor reduction. Results from this simulation will be analyzed further and compared with the 2HD Mississippi Canyon results in a later section.

Mississippi Canyon Landslide Source

The slide and initial water surface condition properties for this source are described above in the corresponding 1HD section, but are given again here for completeness. The initial depression is set to the water depth at the head of the scarp, 150 m. The horizontal dimensions of the slide source region are assumed to be ~30 km in width and 160 km in length, inferred from the multibeam bathymetric data. With this information and knowledge of characteristic slide-generated waves taken from the literature (Lynett & Liu, 2002; Lynett & Liu, 2005), the hot-start initial condition is constructed. A constant spatial grid size of 500 m is used in the numerical simulation.

In the 2HD evolution, within 20 minutes from the landslide, it is clear that radial spreading effects are important for the wave approaching the site. By the time the wave has reached the shelf break the leading elevation wave height is ~15 m, a significant reduction from the hot start elevation of 120 m. The elevation component of the landward traveling wave forms into a bore once on the shelf. The bore front height continues to diminish, and by the time the front reaches a depth of about 30 m, its elevation is approximately 7 m. Note that for the 1HD simulation, the wave height at this depth was 25 m.

<u>Local Evolution of the Tsunami in the Nearshore Areas of the Site</u>

Finally, propagation over the shallow, nearshore bathymetry at the site is examined. The purpose of these simulations is to provide very refined 2HD inundation using the best available bathymetry and topography near the site. This subdomain is nested inside the large-scale 2HD domains discussed above for the Florida Slope WORST CASE and Mississippi Canyon sources. The offshore boundary, situated at a depth of 30 m, is forced with results from the large-scale 2HD simulations. Interestingly, the peak elevations of the wave trains are nearly identical, with the peak Mississippi Canyon crest elevation of 7.2 m, and the peak Florida Slope WORST CASE crest elevation of 6.9 m. The periods of the wave components in these two wave trains are slightly different, with the period from the Mississippi Canyon source at 45 minutes and that from the Florida Slope WORST CASE at 38 minutes. The most significant difference between the two trains is the number of large waves in the train. The Mississippi Canyon wave train has four distinct waves with crest elevation greater than 2 m, while the Florida Slope WORST CASE train has just one. With these comparisons in mind, is it evident that the Mississippi Canyon source produces the PMT for this site, and will be the only source used to simulate the refined, nearshore tsunami impact.

A subdomain, approximately 200 km by 150 km, centered 75 km offshore is used here.. A constant grid size of 100 m is used, and both the seafloor and initially dry land is assumed smooth, with no bottom friction dissipation. This is the most conservative assumption, and provides an upper physical limit for the inundation distance. As mentioned above, the offshore boundary is forced with the Mississippi Canyon sea surface time series. The interaction with the coastline is complex, owing to the complex bathymetry and topography, and the runup elevation is highly variable across the shoreline. In the lower (southern) part of the domain, where relatively steep topography is located close to the shoreline, the maximum runup elevation is +8 m and the inundation distance is ~ 8 km. However, immediately seaward of the site, where a wide, coastal plan exists, the runup elevation is +3 m, but the inundation distance is ~18 km. Thus, the tsunami does not come close to the site ground elevation.

The above simulation assumes that the tsunami event occurs at mid-tide with current sea levels. Independent analysis of the 10% exceedance high tide was conducted for 16 years of NOAA NOS CO-OPS data at the Clearwater Beach, FL tide gauge station (years 1973-2006). The 10 percent exceedance high tide was determined to be 0.75 m (NAVD88) for these years, compared to 0.82 m indicated in the applicant's response to RAI 2.4.6-10. The long-term sea-level rise at the Clearwater Beach, FL station is 2.43±0.80 mm/yr according to NOAA NOS-CO-OPS data. Therefore our estimated antecedent water level is 0.75 m (high tide) + 0.18 m (sea level anomaly) + 0.32 m (100-year sea level rise + 1s.d.) = 1.2 m (NAVD88). The applicant's estimated antecedent water level is 1.1 m (NAVD88) as indicated in their response to RAI 2.4.6-10.

A final simulation, using the identical numerical configuration described in the preceding paragraph is run, with the higher water levels. The maximum runup offshore of the site, using the water level increased by 1.2 m, is +6.1 m. Thus, by increasing the water depth by 1.2 m, the runup elevation was increased by 3.1 m. Clearly, the process of bore evolution is highly nonlinear, and the increase in the water depth allows for a measurably larger wave to reach the shoreline and push farther inland than would be expected by a simple linear addition of the water depth increase (1.2 m) to the previous runup prediction (+3.0 m). However, even when considering this, the maximum tsunami runup in the vicinity of the site does not approach the Levy site ground elevation.

Summary

Numerical modeling of three different types of tsunami sources has been performed to determine their impact on the Levy County site. The three source types are (1) distant earthquake sources; (2) a regional earthquake source in the Gulf of Mexico; and (3) regional submarine landslide sources in the Gulf of Mexico. For the latter source type that defines source for the PMT, water levels from five different submarine landslide scenarios were calculated using COULWAVE to determine the PMT.

Using conservative source parameters and neglecting the radial spreading of wave energy, the 1HD simulations indicate that the Mississippi Canyon source clearly has the greatest potential to bring a large wave to the Levy site, with 1HD water elevations near the site in excess of +30 m. 2HD simulations of this source and the WORST CASE Florida Slope landslide source that include radial spreading predict a maximum wave elevation of 7 m offshore of the site (30 m water depth). However, the Mississippi Canyon wave is longer in period and has a longer train of large waves, and thus is designated as the PMT for the Levy site. Highly refined nearshore simulations show that this source results in a maximum water level of +3 m. Because of nonlinear effects during wave propagation, one cannot simply add an antecedent sea level that includes 10% exceedance high tide, sea level anomaly, and sea-level rise to this maximum water to the +3m maximum water level. A separate simulation that includes the nonlinear propagation effects and a +1.2 m (NAVD88) antecedent sea level results in a maximum water level of +6.1 m. Thus, the PMT does not reach the Levy site plant grade elevation.

2.4.6.4.6 Hydrography And Harbor Or Breakwater Influences On Tsunami

Information Submitted by the Applicant

The applicant indicates that routing of the controlling tsunami, including breaking wave formation and resonance effects, is expected to be minor and limited to shorelines. In addition, the applicant indicates that hydrography and harbor or breakwater influences are not expected to be severe enough to impact safety-related structures.

NRC Staff's Technical Evaluation

The NRC Staff concurs with the applicant in that the hydrography and harbor or breakwater influences are not expected to be severe enough to impact safety-related structures. The offshore hydrography and harbor or breakwater influences are specifically accounted for in the numerical modeling performed during the independent confirmatory analysis.

2.4.6.4.7 Effects On Safety-Related Facilities

Information Submitted by the Applicant

The applicant indicates that the effects of the Probable Maximum Tsunami are not expected to be severe enough to impact the operation of safety-related structures. The applicant further indicates that measures to protect the site against the effects of tsunami are not included in the design criteria.

NRC Staff's Technical Evaluation

The NRC Staff concurs with the applicant in that the effects of the Probable Maximum Tsunami are not expected to be severe enough to impact the operation of safety-related structures

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 COL application and confirmed that the COL applicant has addressed the information relevant to design basis for tsunami flooding. The staff also confirmed that there is no outstanding information required to be addressed in the COL FSAR related to this section.

The staff reviewed the information provided and, for the reasons given above, concludes that the COL applicant has provided sufficient details about the site description to allow a staff evaluation, as documented in Section 2.4.6 of this report. Based on the above, the staff concludes that the identified site characteristics meet the requirements of 10 CFR 52.79(a)(1)(iii) and 10 CFR 100.20(c), with respect to establishing the design basis for SSCs important to safety. The information addressing the COL Information Item 2.4.6 is adequate and acceptable.

2.4.7 Ice Effects

2.4.7.1 *Introduction*

FSAR Section 2.4.7 addresses ice effects to ensure that safety-related facilities and water supply are not affected by ice-induced hazards.

Section 2.4.7 of this SER presents an evaluation of the following topics based on data provided by the applicant in the 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 a surface ice sheet to reduce the volume of available liquid water in safety-related water reservoirs; (4) potential effects of ice in producing forces on, or causing blockage of, safety-related facilities; (5) potential effects of seismic and non-seismic data on the postulated worst-case icing scenario for the proposed plant site; (6) any additional information requirements prescribed in the “Contents of Application” sections of the applicable subparts to 10 CFR Part 52.

2.4.7.2 *Summary of Application*

This section of the COL FSAR addresses the site-specific information about ice effects. The applicant addressed the information as follows:

AP1000 COL Information Item

- LNP COL 2.4-2

In addition, this section addresses the following COL-specific information identified in Section 2.4.1.2 of Revision 19 of the AP 1000 DCD.

> Combined License applicants referencing the AP1000 design will address the following site-specific information on historical flooding and potential flooding factors, including the effects of local intense precipitation:
>
> - Probable Maximum Flood on Streams and Rivers – Site-specific information that will be used to determine design basis flooding at the site. This information will include the probable maximum flood 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 of flood level.

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 the Commission regulations for the identification and evaluation of ice effects, and the associated acceptance criteria, are described in Section 2.4.7 of NUREG-0800 (NRC 2007a).

The applicable regulatory requirements for identifying ice effects are as follows:

- 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), as it sets forth the criteria to determine the siting factors for plant design bases with respect to water levels at the site.

- 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.

The related acceptance criteria are provided in the following RGs:

- RG 1.59, "Design Basis Floods for Nuclear Power Plants" (NRC 1977a), as supplemented by best current practices

- RG 1.102, "Flood Protection for Nuclear Power Plants" (NRC 1976b).

2.4.7.4 *Technical Evaluation*

The NRC staff reviewed Section 2.4.7 of the LNP 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 site-specific ice effects. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

2.4.7.4.1 Ice Conditions and Historical Ice Formation

Information Submitted by the Applicant

The applicant reviewed the historical temperature records from the NWS Cooperative Observer Station in Ocala, Florida. The monthly average minimum temperatures for the months of December, January, and February for the period 1971–2000 were 8.5, 7.6 and 8.3 ºC (47.3, 45.7, and 47 ºF), and the corresponding monthly mean temperatures were 15.3, 14.5, and 15.5 ºC (59.5, 58.1, and 59.9 ºF). The applicant concluded that ice formation on large bodies of water in the vicinity of the LNP site is unlikely and would not be severe enough to adversely affect the operation of safety-related SSCs.

NRC Staff's Technical Evaluation

The staff reviewed air temperature data from NOAA Cooperative Stations near the LNP site to evaluate the possibility of ice formation in the vicinity of the LNP site. The staff found several first-order stations located near the LNP site as listed in Table 2.4.7-1.

Table 2.4.7-1. First-Order NOAA NWS Cooperative Stations Located near the LNP Site

Name	County	Start Date	End Date
Inglis 3E	Levy	August 1, 1948	September 30, 1951
Morriston	Levy	March 1, 1940	February 28, 1942
Rockwell	Marion	August 1, 1899	June 30, 1919
Inverness 3 SE	Citrus	February 1, 1899	April 30, 2010
Ocala	Marion	January 1, 1892	February 28, 2010
Ocala 2NE	Marion	January 1, 1946	January 31, 1966

Of the stations near the LNP site, only those at Ocala and Inverness have long-term and current observations. The staff used these two meteorological stations to estimate characteristics of air temperature near the LNP site (Table 2.4.7-2).

Table 2.4.7-2. Statistics of Low Air Temperatures near the LNP Site

Statistics	Inverness	Ocala
Lowest daily mean air temperature	-4.4 ºC (24 ºF) on 2/14/1899	-3.6 ºC (25.5 ºF) on 12/24/1989
Number of days with daily mean air temperature below freezing	14 of 31,983	19 of 40,189
Longest period with daily mean air temperature at or below 0 ºC (32 ºF)	2 (three times)	2 (twice)
Longest period with daily mean air temperature at or below -7.8 ºC (18 ºF)	none	none

The staff independently determined that mean daily air temperature rarely (once in 2000 days) falls below freezing at the Inverness and Ocala stations. The longest duration over which mean daily air temperature was at or below freezing was 2 days at both Inverness and Ocala stations. There were no periods when mean daily air temperature fell below -7.8 ºC (18 ºF). Frazil ice forms in turbulent, supercooled water that is not covered by an ice layer but is directly in contact with the atmosphere with air temperature below -7.8 ºC (18 ºF) (USACE 2002). The staff concluded that ice formation, including frazil formation near the LNP site, is an unlikely event.

The LNP sites would host AP1000 units, which do not rely on an external safety-related source of water for safe shutdown. Therefore, the staff concluded that ice formation at the LNP site would not adversely affect safety-related SSCs for Units 1 and 2.

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 COL FSAR related to this section.

As set forth above, the applicant has presented and substantiated information to establish the site description. The staff has reviewed the information provided and, for the reasons given above, concludes that the applicant has provided sufficient details about the site description for the staff to determine, as documented in Section 2.4.7 of this SER, that the applicant has met the relevant requirements of 10 CFR 52.79(a)(1)(iii) and 10 CFR Part 100 with respect to determining the acceptability of the site. This addresses COL Information Item 2.4-2.

2.4.8 Cooling-Water Canals and Reservoirs

2.4.8.1 *Introduction*

FSAR Section 2.4.8 addresses the cooling-water canals and reservoirs used to transport and impound water supplied to the safety-related SSCs. Section 2.4.8 of this SER presents an evaluation of the following topics to verify their hydraulic design basis: (1) design bases postulated and used by the applicant to protect structures such as riprap, inasmuch as they apply to safety-related water supply; (2) design bases of canals pertaining to capacity, protection against wind waves, erosion, sedimentation, and freeboard and the ability to withstand a PMF (surges, etc.), inasmuch as they apply to a safety-related water supply; (3) design bases of reservoirs pertaining to capacity, PMF design basis, wind-wave and run-up protection, discharge facilities (e.g., low-level outlet, spillways, etc.), outlet protection, freeboard, and erosion and sedimentation processes inasmuch as they apply to a safety-related water supply; (4) potential effects of seismic and non-seismic information about the postulated hydraulic design bases of canals and reservoirs for the proposed plant site; and (5) any additional information requirements prescribed in the "Contents of Application" sections of the applicable subparts to 10 CFR Part 52.

2.4.8.2 *Summary of Application*

Safety systems for the AP1000 are designed to function without safety-related support systems such as component cooling water and service water. None of the safety-related equipment requires cooling water to affect a safe shutdown or mitigate the effects of design basis events. Heat transfer to the ultimate heat sink is accomplished by heat transfer through the containment shell to air and water flowing on the outside of the shell supplied by a passive containment cooling water tank. Therefore, the AP1000 design does not rely on service water and component cooling water systems to provide safety-related safe shutdown. There are no COL items related to cooling-water canals and reservoirs.

2.4.8.3 *Regulatory Basis*

The relevant requirements of the Commission regulations for the identification of design considerations for cooling-water canals and reservoirs, and the associated acceptance criteria, are described in Section 2.4.8 of NUREG-0800 (NRC 2007a).

The applicable regulatory requirements for cooling-water canals and reservoirs are as follows:

- 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), as it sets forth the criteria to determine the siting factors for plant design bases with respect to water levels at the site.

- 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.

The related acceptance criteria are provided in the following RGs:

- RG 1.59, “Design Basis Floods for Nuclear Power Plants” (NRC 1977a), as supplemented by best current practices

- RG 1.102, “Flood Protection for Nuclear Power Plants” (NRC 1976b).

2.4.8.4 *Technical Evaluation*

<u>Information Submitted by the Applicant</u>

The applicant stated that safety systems of the AP1000 reactor are designed to function without safety-related support systems such as component cooling water and service water. Heat transfer to the ultimate heat sink (UHS) occurs through the containment shell to the atmosphere and water supplied from a passive containment cooling-water tank. The applicant concluded, therefore, that no design bases for cooling-water canals or reservoirs are needed.

NRC Staff's Technical Evaluation

The staff reviewed the function of the AP1000 UHS and concluded that no external source of safety-related water is needed apart from the initial filling and occasional makeup water to the passive containment cooling-water storage tank located above the containment vessel and the passive containment cooling ancillary water storage tank located at ground level near the auxiliary building. Therefore, no safety-related cooling-water canals or reservoirs are needed at the LNP site with a permanent external source of water supply.

2.4.8.4.1 Cooling-Water Canals

Information Submitted by the Applicant

The applicant stated that safety systems of the AP1000 reactor are designed to function without safety-related support systems such as component cooling water and service water. Heat transfer to the UHS occurs through the containment shell to the atmosphere and water supplied from a passive containment cooling-water tank. The applicant concluded, therefore, that no design bases for cooling-water canals or reservoirs are needed.

NRC Staff's Technical Evaluation

The staff reviewed the function of the AP1000 UHS and concluded that no external source of safety-related water is needed apart from the initial filling and occasional makeup water to the passive containment cooling-water storage tank located above the containment vessel and the passive containment cooling ancillary water storage tank located at ground level near the auxiliary building. Therefore, no safety-related cooling-water canals or reservoirs are needed at the LNP site with a permanent external source of water supply.

2.4.8.4.2 Reservoirs

Information Submitted by the Applicant
The applicant stated that safety systems of the AP1000 reactor are designed to function without safety-related support systems such as component cooling-water and service water. Heat transfer to the UHS occurs through the containment shell to the atmosphere and water supplied from a passive containment cooling-water tank. The applicant concluded, therefore, that no design bases for cooling-water canals or reservoirs are needed.

NRC Staff's Technical Evaluation
The staff reviewed the function of the AP1000 UHS and concluded that no external source of safety-related water is needed apart from the initial filling and occasional makeup water to the passive containment cooling-water storage tank located above the containment vessel and the passive containment cooling ancillary water storage tank located at ground level near the

auxiliary building. Therefore, no safety-related cooling-water canals or reservoirs are needed at the LNP site with a permanent external source of water supply.

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 scope of Section 2.4.8 is not relevant to the LNP COL.

2.4.9 Channel Diversions

2.4.9.1 *Introduction*

LNP FSAR Section 2.4.9 addresses channel diversions. It evaluates plant and essential water supplies used to transport and impound water supplies to ensure that they will not be adversely affected by stream or channel diversions. The evaluation includes stream channel diversions away from the site (which may lead to a loss of safety-related water) and stream channel diversions toward the site (which may lead to flooding). In addition, in such an event, it must be ensured that alternate water supplies are available to safety-related equipment.

Section 2.4.9 of this SER presents an evaluation of the following specific areas: (1) historical channel migration phenomena including cutoffs, subsidence, and uplift; (2) regional topographic evidence that suggests a future channel diversion may or may not occur (used in conjunction with evidence of historical diversions); (3) thermal causes of channel diversion, such as ice jams, which may result from downstream ice blockages that may lead to flooding from backwater or upstream ice blockages that can divert the flow of water away from the intake; (4) potential for forces on safety-related facilities or the blockage of water supplies resulting from channel migration-induced flooding (flooding not addressed by hydrometeorological-induced flooding scenarios in other sections); (5) potential of channel diversion from human-induced causes (i.e., land-use changes, diking, channelization, armoring, or failure of structures); (6) alternate water sources and operating procedures; (7) potential effects of seismic and non-seismic information about the postulated worst-case channel diversion scenario for the proposed plant site; (8) any additional information requirement prescribed in the “Contents of Application” sections of the applicable subparts to 10 CFR Part 52.

2.4.9.2 *Summary of Application*

Safety systems for the AP1000 are designed to function without safety-related support systems such as component cooling water and service water. None of the safety-related equipment requires cooling water to affect a safe shutdown or mitigate the effects of design basis events. Heat transfer to the ultimate heat sink is accomplished by heat transfer through the containment shell to air and water flowing on the outside of the shell supplied by a passive containment cooling water tank. Therefore, the AP1000 design does not rely on service water and component cooling water systems to provide safety-related safe shutdown. There are no COL items related to cooling-water canals and reservoirs. There are no COL items related to channel diversions.

2.4.9.3 *Regulatory Basis*

The relevant requirements of the Commission regulations for the identification and evaluation of channel diversions, and the associated acceptance criteria, are described in Section 2.4.9 of NUREG-0800 (NRC 2007a).

The applicable regulatory requirements for identifying and evaluating channel diversions are as follows:

- 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), as it sets forth the criteria to determine the siting factors for plant design bases with respect to water levels at the site.

- 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.

The related acceptance criteria are provided in the following RGs:

- RG 1.59, “Design Basis Floods for Nuclear Power Plants” (NRC 1977a), as supplemented by best current practices

- RG 1.102, “Flood Protection for Nuclear Power Plants” (NRC 1976b).

2.4.9.4 *Technical Evaluation*

Information Submitted by the Applicant

The applicant stated that the CFBC is a man-made drainage structure that is not susceptible to migration or cutoff. The applicant concluded, based on gauge height data at two stations that no channel diversion of significance has occurred in approximately 35 years of record. The applicant concluded, based on the size of the Gulf of Mexico, that complete diversion of the Gulf is unlikely. The applicant stated, based on topographic characteristics, geological features, and low seismic activity in the drainage basin, that there is no possibility of a landslide-induced blockage that might limit flow of water into the CFBC from the Gulf of Mexico or from Lake Rousseau. The applicant also stated that because ice effects in the vicinity of the LNP site are considered unlikely, ice-induced diversion during winter months is also unlikely. The applicant stated that a potential for anthropogenic diversion of CFBC exists; however, because it is located in a relatively unpopulated area, the potential for such an event is unlikely.

The applicant stated that the AP1000 design does not have a safety-related cooling-water system and therefore, does not rely on service water and component cooling-water systems for safe shutdown.

NRC Staff's Technical Evaluation

The staff reviewed the function of the AP1000 UHS and concluded that no external source of safety-related water is needed apart from the initial filling and occasional makeup water to the passive containment cooling-water storage tank located above the containment vessel and the passive containment cooling ancillary water storage tank located at ground level near the auxiliary building. Therefore, the LNP units will not rely on any external source of water for safety-related use. The NRC staff concluded that any potential channel migration in the vicinity of the site would not affect safe shutdown of the plant.

The staff evaluated the possibility of a channel diversion-induced flood near the LNP site. The staff determined that the safety-related SSCs of the LNP units would be located in the Waccasassa River Basin, specifically in the Spring Run and Thousandmile Creek-Halverson Creek subbasins. Surface drainages in both of these subbasins drain directly to the Gulf, so they do not contribute flow to the Waccasassa River. The safety-related SSCs of the LNP units would be located near the upper portion of these two subbasins, where there are no named streams or watercourses and overland flow during large precipitation events is drained toward the west and southwest. Based on this review of topography and hydrology in the vicinity of the LNP site, the NRC staff determined that a future channel diversion is unlikely in the vicinity of the LNP site. The staff concluded therefore that the safety-related SSCs of the LNP units would be safe from adverse effects of any potential channel diversion.

The staff reviewed Section 2.4.9 of the LNP 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 related to this review topic. Because the AP1000 reactor design does not require makeup water from offsite for safety-related purposes, the staff determined that the scope of FSAR 2.4.9 is not relevant for the LNP COL.

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 scope of Section 2.4.9 is not relevant to the LNP COL.

2.4.10 Flooding-Protection Requirements

2.4.10.1 *Introduction*

FSAR Section 2.4.10 addresses the 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 SER presents an evaluation of the following specific areas:
(1) safety-related facilities exposed to flooding; (2) type of flood protection (e.g., "hardened facilities," sandbags, flood doors, bulkheads, etc.) provided to the SSCs exposed to floods; (3) emergency procedures needed to implement flood protection activities and warning times available for their implementation reviewed by the organization responsible for reviewing issues related to plant emergency procedures; (4) potential effects of seismic and non-seismic information about the postulated flood protection for the proposed plant site; and (5) any additional information requirements prescribed in the "Contents of Application" sections of the applicable subparts to 10 CFR Part 52.

2.4.10.2 *Summary of Application*

This section of the COL FSAR addresses the needs for site-specific information about flood protection requirements. The applicant addressed the information as follows:

COL Information Items

- LNP COL 2.4-2

In addition, this section addresses the following COL-specific information identified in Section 2.4.1.2 of Revision 19 of the AP1000 DCD.

> Combined License applicants referencing the AP1000 design will address the following site-specific information on historical flooding and potential flooding factors, including the effects of local intense precipitation.
>
> - Probable Maximum Flood on Streams and Rivers – Site-specific information that will be used to determine design basis flooding at the site. This information will include the probable maximum flood 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 of flood level.

No further action if required for sites within the bounds of the site parameter for flood level.

This section of the SER relates to historical flooding and local intense precipitation.

- LNP COL 2.4-6

In addition, this section addresses the following COL-specific information identified in Section 2.4.1.6 of Revision 19 of the DCD.

2.4.10.3 *Regulatory Basis*

The relevant requirements of the Commission regulations for the identification and evaluation of flooding protection requirements, and the associated acceptance criteria, are described in Section 2.4.10 of NUREG-0800 (NRC 2007a).

The applicable regulatory requirements for identifying and evaluating flooding protection requirements are as follows:

- 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), as it sets forth the criteria to determine the siting factors for plant design bases with respect to water levels at the site.

- 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.

The related acceptance criteria are provided in the following RGs:

- RG 1.59, “Design Basis Floods for Nuclear Power Plants” (NRC 1977a), as supplemented by best current practices

- RG 1.102, “Flood Protection for Nuclear Power Plants” (NRC 1976b).

2.4.10.4 *Technical Evaluation*

Information Submitted by the Applicant

The applicant stated that the AP1000 site parameters bound the LNP site flood levels.

NRC Staff's Technical Evaluation

The NRC staff reviewed the applicant's FSAR and related RAI responses to determine that the maximum floodwater surface elevation at the LNP site is 15.17 m (49.78 ft) NAVD88. This results from a probable maximum storm surge combined with wind-induced setup, as described in Section 2.4.2 of this SER. The maximum floodwater surface elevation is below the nominal plant grade floor elevation of 15.5 m (51 ft) NAVD88. The staff concluded therefore, that the DCD maximum flood level parameter would not be exceeded. Therefore, no flood protection is required for LNP Units 1 and 2.

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 DC rule, and that there is no outstanding information required to be addressed in the 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 LNP Units 1 and 2. The staff finds that the applicant has considered the appropriate site phenomena in establishing the flood protection measures for SSCs. The staff has reviewed the information provided and, for the reasons given above, concludes that the applicant has provided sufficient details about the site description for the staff to determine, as documented in Section 2.4.10 of this SER, that the applicant has met the relevant requirements of 10 CFR 52.79(a)(1)(iii) and 10 CFR Part 100 with respect to determining the acceptability of the site.

2.4.11 Low-Water Considerations

2.4.11.1 *Introduction*

FSAR Section 2.4.11 addresses 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 SER presents an evaluation of the following specific areas: (1) low water conditions due to the worst drought considered reasonably possible in the region; (2) effects of low water surface elevations caused by various hydrometeorological events and a potential blockage of intakes by sediment, debris, littoral drift, and ice because they can affect the safety-related water supply; (3) effects of low water on the intake structure and pump design bases in relation to the events described in SAR Sections 2.4.7, 2.4.8, 2.4.9, and 2.4.11, which consider the range of water supply required by the plant (including minimum operating and shutdown flows during anticipated operational occurrences and emergency conditions) compared with availability (considering the capability of the UHS to provide adequate cooling water under conditions requiring safety-related cooling); (4) use limitations imposed or under discussion by Federal, State, or local agencies authorizing the use of the water; (5) potential effects of seismic and non-seismic information about the postulated worst-case low water scenario for the proposed plant site; and (6) any additional information requirements prescribed in the "Contents of Application" sections of the applicable subparts to 10 CFR Part 52.

2.4.11.2 *Summary of Application*

This section of the COL FSAR addresses the impacts of low water on water supply. The applicant addressed the information as follows:

<u>AP1000 COL Information Item</u>

- LNP COL 2.4-3

In addition, this section addresses the following COL-specific information identified in Section 2.4.1.3 of Revision 19 of the AP1000 DCD.

> 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 the Commission regulations for the low water considerations, and the associated acceptance criteria, are described in Section 2.4.11 of NUREG-0800 (NRC 2007a).

The applicable regulatory requirements for identifying the effects of low water are as follows:

- 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), as it 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.

- 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.

2.4.11.4 *Technical Evaluation*

The NRC staff reviewed Section 2.4.11 of the LNP 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 low water considerations. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

Information Submitted by the Applicant regarding Low Flow in Rivers and Streams

The applicant provided an analysis of low flow in the Withlacoochee River using observed data at five USGS streamflow gauging stations.

Information Submitted by the Applicant regarding Historical Low Water

The applicant provided an analysis of low flow in the Withlacoochee River using observed data at five USGS streamflow gauging stations. The applicant compared the dates of the lowest observed water levels with those of hurricane occurrences but did not find any relationship between the two. The applicant concluded that low flow events are more likely to be caused by other effects, such as droughts.

Information Submitted by the Applicant regarding Heat Sink Dependability Requirements

The applicant stated that the UHS for the AP1000 design would not be affected by any low flow events because it does not rely on service water and component cooling-water systems. Water withdrawn from the CFBC would only be used to provide normal operational needs.

NRC Staff's Technical Evaluation

The staff reviewed the AP1000 DCD to evaluate the impact of low water conditions in the vicinity of the LNP site on the safety of the LNP units. Since no external water source is required for safe emergency shutdown, the staff determined that low water conditions would have no impact on the safety of the LNP units. There are no site characteristics in the DCD associated with low water conditions.

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 required information, that there are no site characteristics in the DCD associated with low water conditions, and that there is no outstanding information required to be addressed in the COL FSAR related to this section..

As set forth above, the applicant has presented and substantiated information related to the low water effects important to the design and siting of this plant. The staff finds that the applicant has considered the appropriate site phenomena in establishing the design bases for SSCs. The staff has reviewed the information provided and, for the reasons given above, concludes that the applicant has provided sufficient details about the site description for the staff to determine, as documented in Section 2.4.11 of this SER, that the applicant has met the relevant requirements of 10 CFR 52.79(a)(1)(iii) and 10 CFR Part 100 with respect to determining the acceptability of the site. This addresses COL information item 2.4-3.

2.4.12 *Groundwater*

2.4.12.1 *Introduction*

FSAR Rev. 4 Section 2.4.12 describes the hydrogeological characteristics of the site. The most significant objective of groundwater investigations and monitoring at this site is to evaluate the effects of groundwater on plant foundations. The evaluation is performed to ensure that the maximum groundwater elevation remains below the DCD site parameter value. The other significant objectives are to examine whether groundwater provides any safety-related water supply; to determine whether dewatering systems are required to maintain groundwater elevation below the required level; to measure characteristics and properties of the site needed to develop a conceptual site model of groundwater movement; and to estimate the direction and velocity of movement of potential radioactive contaminants.

This section presents an evaluation of the following specific areas: (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 the movement of accidental contaminants in groundwater, groundwater levels beneath the site, seasonal and climatic fluctuations, monitoring and protection requirements, and manmade changes that have the potential to cause long-term changes in local groundwater regime; (2) effects of groundwater levels and other hydrodynamic effects of groundwater on the design bases of plant foundations and 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 on the postulated worst-case groundwater conditions for the proposed plant site; and (6) any additional information requirements prescribed in the "Contents of Application" sections of the applicable subparts to 10 CFR Part 52.

2.4.12.2 *Summary of Application*

This section of the COL FSAR addresses groundwater conditions in terms of impacts on structures and water supply. The applicant addressed these issues as follows:

AP1000 COL Information Item

- LNP COL 2.4-4

This COL item is addressed by FSAR Section 2.4.12. In particular, this section addresses the site-related parameter for groundwater level that is specified in Table 2-1 of Revision 19 of the DCD, and is defined and discussed in Section 2.4.1.4 of Revision 19 of the DCD.
Section 2.4.1.4 states:

> 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.

2.4.12.3 *Regulatory Basis*

The relevant requirements of the Commission regulations for groundwater, and the associated acceptance criteria, are described in Section 2.4.12 of NUREG-0800.

The applicable regulatory requirements are as follows:

- 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), as it 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.

- 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.

2.4.12.4 *Technical Evaluation*

The NRC staff reviewed Section 2.4.12 of the LNP 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 groundwater. The results of the NRC staff's evaluation of the information

incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

2.4.12.4.1 Hydrogeological Description and Onsite Use of Groundwater

Information Submitted by the Applicant

The applicant stated that the LNP site is located on the Floridan platform, which consists of a sequence of Mesozoic and Cenozoic age shallow marine carbonate and evaporite sediments approximately 5,000 m (16,000 ft) thick. The site is located in the Gulf Coastal Lowlands, a subdivision of Florida's mid-peninsular physiographic zone. Much of the Gulf Coastal Lowlands has karst topography, an irregular terrain caused when near-surface carbonate rocks are dissolved by infiltrating rainwater.

The applicant described aquifers at the LNP site as consisting of a surficial aquifer, composed of unconsolidated Quaternary age sediments, and the deeper Floridan aquifer system found in the deeper predominately carbonate rocks of Miocene to Paleocene age. The Floridan aquifer system is extensive and receives recharge from a large area extending into Georgia, Alabama, and South Carolina. The Floridan aquifer system in Florida ranges in thickness from about 150 m (500 ft) to over 550 m (1800 ft) and consists of the Upper and Lower Floridan aquifers. The Upper Floridan and Lower Floridan aquifers are separated by low-permeability evaporite deposits and dense dolostones that form the middle confining unit (MCU). The MCU can be up to 122 m (400 ft) thick in the vicinity of the LNP site.

The Upper Floridan aquifer was described as the main source of potable water and spring flow in west-central Florida. The underlying Lower Floridan aquifer contains saline water and is not used as a potable water source near the LNP site. Site investigation boreholes drilled to as much as 152 m (500 ft) bgs (below ground surface) did not encounter the MCU (the bottom of the Upper Floridan aquifer) because it is below this depth.

The applicant described the local surficial aquifer as composed of sands. The applicant described the surficial aquifer as being recharged by wetlands mainly associated with cypress tree growth areas. The surficial aquifer in turn provides substantial recharge to the underlying Floridan aquifer system. Sands of the surficial aquifer grade into the carbonate-derived silty sediments at the top of the underlying Avon Park Formation (the uppermost geological formation within the Floridan aquifer that is present locally). The applicant stated that the thickness of the surficial aquifer at the LNP site varies from less than 3 m (10 ft) to about 60 m (200 ft) and the average thickness is approximately 15 m (50 ft). The applicant further described the surficial aquifer as being hydraulically connected to the Floridan aquifer. The water table in the surficial aquifer was generally found at depths of less than 1.5 m (5 ft). The water table varies seasonally depending on the amount of rainfall.

The applicant stated that the Upper Floridan aquifer is highly productive with transmissivity (thickness multiplied by hydraulic conductivity) estimated to range from approximately 4,645 to 9,290 m^2/d (50,000 to 100,000 ft^2/d) in the vicinity of the LNP site.

The reported site investigations included the drilling of geotechnical borings; installation and monitoring of wells completed in the surficial and upper bedrock aquifers; performance of slug tests and pumping tests; and analysis of water and soil samples. The applicant stated that there is no current onsite use of groundwater at the LNP site. The applicant indicated that general plant water supply for the new units, including service water tower drift and evaporation, potable water supply, raw water to the demineralizer, fire protection, and media filter backwash, will be provided by water supply wells completed in the Upper Floridan aquifer. The average flow rate needed was predicted to be 3,337 L/min (881.5 gpm).

NRC Staff's Technical Evaluation

The staff reviewed the information provided by the applicant in the FSAR regarding regional and site hydrogeology, groundwater conditions, and onsite groundwater use. The staff found the applicant's regional information to be comparable to the description provided in the "Ground Water Atlas of the United States" (USGS 1990) and in reports published by the Florida Geological Survey (Rupert 1988; Arthur et al. 2001). The staff confirmed that freshwater aquifers at the site include the uppermost surficial aquifer and the thicker and more extensive Upper Floridan aquifer. The staff also confirmed that no confining unit exists between the surficial and Upper Floridan aquifer systems in this area, and that these two aquifers are hydraulically connected. The staff found that hydraulic conductivity of the surficial aquifer is generally lower than that of the Upper Floridan aquifer. However, karst features that may be associated with some of the wetlands on the LNP site could result in areas of enhanced vertical hydraulic conductivity and connection between the surface and the Upper Floridan aquifer (White 1988). Neither of the aquifers is classified as a sole-source aquifer. The closest sole-source aquifer is the Volusia Sole-Source Aquifer, located approximately 80 mi east of the LNP site (EPA 2011).

The staff issued RAI 2.4.12-01 requesting additional information about groundwater chemistry as it relates to the transport properties of the subsurface. In response, the applicant provided groundwater chemistry data from the site monitoring wells and information related to the effects of groundwater chemistry on the transport of potential radioactive contaminants (ML092150960). The staff reviewed the information and determined that the information was adequate to support the analysis of transport from a hypothetical spill to groundwater presented in Section 2.4.13 of this report.

The staff found that there is no current onsite use of groundwater at the LNP site. Fresh groundwater from the Upper Floridan aquifer would be used for general plant water supply at LNP Units 1 and 2, but not for reactor cooling water. Groundwater will be withdrawn at an average of 4,153 L/min (1,097 gpm, or 1.58 mgd) to provide makeup water for service water tower drift and evaporation, potable water supply, raw water to the demineralizer, fire protection, and media filter backwash. The staff determined that the groundwater supply's lack of safety function is consistent with the uses stated for groundwater, and with provisions for safety-related water supply from other sources, as described in the FSAR Revision 2.

2.4.12.4.2 Groundwater Sources, Present and Future Groundwater Use

Information Submitted by the Applicant

The applicant determined that within 40.2 km (25 mi) of the LNP site, the SWFWMD has issued approximately 53,670 well permits, and the Suwanee River Water Management District (SRWMD) has issued 918 well permits. The applicant also determined that there are 268 public water supply systems within a 40.2 km (25-mi) radius of the LNP site. Of these, 46 public water supply systems serving 10,300 customers and having total design capacity of approximately 25 MLd (6.6 Mgd) are within 16 km (10 mi) of the LNP site. A total of 64 wells draw water from the Upper Floridan aquifer for these 46 public water supply systems. The applicant also found that three municipal/city systems account for approximately 7.2 MLd (1.9 Mgd), or 30 percent of the total public water supply design capacity within 16.1 km (10 mi) of the LNP site. The numbers and types of permitted wells were tabulated by Township Range and Section in FSAR Revision 4. Information about public water supply wells was also presented in the FSAR.

The applicant indicated that SWFWMD projected an increase in water demand within Levy County from approximately 49.6 MLd (13.1 Mgd) in 1994 to approximately 68.5 MLd (18.1 Mgd) in 2020, an increase of 18.9 MLd (5.0 Mgd) or 38 percent (SWFWMD 1997). However, the applicant also found that water use actually decreased in Levy County between 1994 and 2005, when it was reported as approximately 35.9 MLd (9.5 Mgd).

The applicant conducted a land-use survey covering the area within 8 km (5 mi) of the LNP site to identify the nearest residents and collect information including the number and use of wells. The results showed that all of the residents within this area use groundwater to supply their potable water needs, and that the depths of these private water wells range from 6 m (20 ft) to 137 m (450 ft) bgs. The nearest residential well was found to be about 2.6 km (1.6 mi) northwest of the LNP site.

NRC Staff's Technical Evaluation

The staff reviewed the information provided in FSAR Rev. 4 on current groundwater use and checked the provided data through queries of electronic databases available from the SWFWMD (2011) and SRWMD. The staff found that information provided in the FSAR was accurate, but, as noted by the applicant, some wells in the database may no longer be in use. This would result in an over-estimate of groundwater users. The staff issued RAI 2.4.12-03 to request an explanation for why, as shown in FSAR Figures 2.4.12-206 to 2.4.12-210, the density of wells in the SWFWMD was apparently much greater than in the SRWMD. In response, the applicant indicated that the SWFWMD requires registration of all wells, including domestic wells, but the SRWMD does not require registration of domestic wells.

The staff found that information provided in the FSAR was accurate, but, as noted by the applicant, some wells in the database may no longer be in use. This would result in an over-estimate of groundwater users. The staff checked the documents (SWFWMD 1997; SWFWMD 2009b) cited in the FSAR and verified information presented regarding future water use. There is uncertainty in the projections of groundwater use because previously published projections indicate steadily increasing population and water use. However, groundwater use in the area has decreased since 1994. The staff determined that the projected future water use provided in LNP FSAR Rev 2 of approximately 68.5 MLd (18.1 Mgd) in the year 2020 is conservatively higher than the likely actual future use. Projected water use in the SRWMD through 2030 was presented in a Water Supply Assessment (SRWMD 2010b). The purpose of the assessment was to determine whether water supplies in the district will satisfy water demands for all uses in the 2010 to 2030 planning period while protecting the environment. The SRWMD assessment estimated a range of 17 to 45 percent increase in demand for public water supply over the 20-year period. The applicant's estimation of projected increase in groundwater use through 2020 is within this range.

The staff issued RAI 2.4.12-02 requesting additional information about the planned plant water supply wells, including the design of the wellfield and the projected impacts of pumping on transport pathways, surrounding surface waters, and adjacent offsite groundwater users. In response, the applicant provided details on the plant water supply wells, including location, number of wells, and peak and average expected flow rates (ML092150960). The applicant also referred to the results of a site groundwater model (ML092240668). However, this model was subsequently revised by the applicant based on staff's environmental RAI 5.2.2-4 (ML093620182) related to the LNP environmental impact statement. The new revision of the groundwater model was documented by the applicant (ML093620211). The staff reviewed the revised groundwater model (ML093620211) and found that it did achieve the goals of matching groundwater levels measured on the LNP site and in four other wells measured in the area by the USGS. Results from the predictive model simulations showed that annual average LNP groundwater usage is relatively small compared to the overall model water balance. The LNP average operational usage of 5.98 MLd (1.58 Mgd) represents only 0.8 percent of the total water flux (787 MLd [208 Mgd]) through the model domain. At this withdrawal rate, the LNP wellfield is predicted to decrease the surficial and Upper Floridan aquifer discharge to surface waterbodies within the model domain by approximately 1.5 MLd (0.4 Mgd), or about 2 percent of the total simulated groundwater discharge to rivers and lakes.

Based on the information provided on the planned water supply wells, expected pumping rates, and the revised model calculation of water level impacts, the staff determined that pumping of the water supply wells will have little effect on offsite groundwater users or surface waterbodies. Significant problems have resulted from overuse of groundwater in upland northeastern portion of the SRWMD (SRWMD 2010a). However, the location of the LNP site in the lower portion of the drainage basin results in adequate recharge of the aquifer to meet demand.

The staff also determined that the planned groundwater supply for the proposed units does not have a safety function, so a loss of the groundwater supply will not compromise plant safety.

2.4.12.4.3 Groundwater Levels and Movement

Information Submitted by the Applicant

The applicant characterized the hydrogeology of the LNP site using groundwater observations, well tests, laboratory tests, and examination of site topography and geology.

The applicant described the observation well network installed to monitor water levels and determine hydraulic gradients and groundwater flow paths for the surficial and bedrock aquifers in the vicinity of the LNP site. Nested well sites with shallow, intermediate, and deep monitoring wells were installed and monitored to determine vertical gradients between the surficial and bedrock aquifers and variations over time.

The applicant installed a pumping test well and 23 observation and monitoring wells in 2007. The pumping test well and 15 of the observation and monitoring wells were screened within the silt and sand of the surficial aquifer directly above the bedrock interface at depths of 4 to 10.4 m (13 to 34 ft) bgs. Seven wells were installed at depths of 37.2 to 46.9 m (122 to 154 ft) in the limestone of the Upper Floridan aquifer and two wells were installed at an intermediate depth of 20.7 to 24.1 m (68 to 79 ft) within the limestone bedrock of the Upper Floridan aquifer. Water levels were measured in the wells in March, June, September, and December of 2007 to determine the configuration of the potentiometric surface in the immediate vicinity of the LNP site. The applicant found that the depth to groundwater was between 0 and 2.4 m (0 and 8 ft) with the shallowest groundwater levels occurring during the spring. The applicant determined that the groundwater is shallow and unconfined, and that groundwater conditions are influenced by the topography of the LNP site. They described the groundwater as flowing from a topographic high of approximately 18.3 m (60 ft) NGVD29 in the eastern portion of the site toward a topographic low of approximately 10.7 m (35ft) NGVD29 in the southwest portion of the site. In the center portion of the site, where the topography is relatively flat, the groundwater surface also becomes relatively flat. The applicant found that no significant differences were observed in groundwater flow direction or gradient during the quarterly measurements or between the surficial and bedrock aquifer.

The applicant installed pressure transducers in two wells screened in the surficial aquifer and collected groundwater elevation data every 12 hours for more than a year. These wells were located at the approximate center of the footprints for each of the two new units. The applicant found that maximum groundwater elevations were observed during March 2007 and March 2008 at both wells. They also found that groundwater elevations were more than 2.1 m (7 ft) below nominal plant grade elevation and more than 2.4 m (8 ft) below nominal plant floor elevation between March 2007 and March 2008.

The applicant calculated horizontal gradients of 0.0003 to 0.0007 between pairs of upgradient and downgradient monitoring wells based on March 2007 water level measurements. The applicant found slightly greater hydraulic heads within the surficial aquifer compared to the bedrock Floridan aquifer based on measurements at the six nested well sites. Measured vertical gradients in March 2007 for all sets of wells ranged from 0.0003 to 0.006 based on the vertical distance between the mid-point of the well screens. The two well pairs (MW-15S/MW-16D and MW-13S/MW14D) located within the footprint of LNP 1 and LNP 2 had slight downward vertical gradients with elevation head differences of 0.17 and 0.08 m (0.55 and 0.27 ft), respectively, in September 2007. The applicant found that the vertical gradients between the surficial and bedrock aquifers remained consistent for all nested well sets during each quarterly gauging event. However, groundwater levels in both aquifers were found to be higher in the spring and lower in the fall.

NRC Staff's Technical Evaluation

The staff issued RAI 2.4.12-05 requesting the site groundwater elevation monitoring data (including the monitoring locations) and the available historical seasonal groundwater elevations in the vicinity of the LNP site. In response, the applicant provided a map of site monitoring locations and also provided the measured groundwater elevation data for the onsite monitoring wells, including quarterly monitoring events and hourly measurements collected using pressure transducers (ML092190616). The applicant's response also included electronic links to other nearby water level records available from the USGS.

The staff issued RAI 2.4.12-06 requesting that the applicant clarify the description of groundwater discharge areas in the FSAR. The applicant's response referred to the response to RAI 2.4.12-08 discussed below (ML092150960).

In RAI 2.4.12-07, the staff asked the applicant to clarify "the significance of vertical hydraulic gradients in relation to the selection of the most conservative plausible conceptual model for transport of radioactive liquid effluents in the subsurface." The applicant responded with an explanation that the observed downward gradients between the surficial and bedrock aquifer indicate that effluents would migrate downward into the bedrock aquifer (Upper Floridan aquifer) and that this assumption is appropriately conservative because permitted water supply wells are only completed in the Upper Floridan aquifer and not in the surficial aquifer (ML092150960). The applicant response also indicated that seepage velocities in the Upper Floridan aquifer are greater than those in the surficial aquifer.

The staff issued RAI 2.4.12-08 asking the applicant to clarify the interpretation of vertical groundwater gradients. The applicant responded with a clarification regarding the USGS identification of the LNP area as a recharge/discharge boundary and discussion of the onsite nested-well monitoring results that indicate a generally small but variable downward gradient (ML092150960). The applicant revised the FSAR to include the following text: "Regionally, the USGS has identified the area where the LNP site is located as a recharge/discharge boundary of the Floridan aquifer as shown in Figure 2.4.12-226. Site-specific vertical gradients observed quarterly from early 2007 through early 2008 were all downward and low in magnitude, ranging from 0.0002 to 0.018 (FSAR Table 2.4.12-209)."

The staff reviewed the information provided regarding groundwater levels and the direction and gradient of groundwater movement. The staff determined that the applicant had adequately characterized groundwater movement under pre-construction site conditions through measurements of water levels in both the surficial aquifer and upper Floridan aquifer. Groundwater was found to flow predominately to the southwest with a maximum measured horizontal gradient of 0.0007. The measured vertical component of the pre-construction gradient was consistently downward with a maximum measured gradient of 0.018. The staff agrees that the vertical component of the gradient will continue to be downward during the operational period because pumping of the proposed water supply wells is likely to lower the hydraulic head in the Upper Floridan aquifer. The vertical gradient indicates that any accidentally released contaminants would migrate downward into the bedrock aquifer (Upper Floridan aquifer). However, the staff found that there is uncertainty in the applicant's estimate of future groundwater levels during the period of plant operations because of planned changes to the site, including the placement of fill, changes in surface cover, and installation of stormwater drainage ditches and ponds.

2.4.12.4.4 Site Hydrogeologic Characteristics

Information Submitted by the Applicant

The applicant conducted slug tests in 23 wells to estimate saturated hydraulic conductivity of the surficial and Upper Floridan aquifers. Results ranged from 0.27 m/d (0.9 ft/d) to 8.7 m/d (28.6 ft/d) for the surficial aquifer and from 0.73 m/d (2.4 ft/d) to 16.6 m/d (54.4 ft/d) for the Upper Floridan aquifer.

An aquifer pumping test was also performed at well PW-1. The initial pumping test analysis provided in FSAR Rev 2 resulted in transmissivity values (hydraulic conductivity multiplied by aquifer thickness) ranging from 121 m^2/d (1300 ft^2/d) to 204 m^2/d (2200 ft^2/d) and specific yield estimates from 0.012 to 0.17. The pumping test analysis was later revised and estimates of hydraulic conductivity and groundwater seepage velocity were revised in response to RAIs issued by the NRC staff.

NRC Staff's Technical Evaluation

The staff reviewed information provided in FSAR Rev 2 on site hydraulic characteristics and the related RAI responses. The staff reviewed the multi-layer transient analyses of the applicant's aquifer pumping test provided in response to RAIs 2.4.12-11 (ML092150960) and 2.4.12-22 (ML101740492) and determined that the analysis methods are valid for the test conditions and that these tests provide a reasonable estimate of site-specific hydraulic conductivity of 36.6 to 39.6 m/d (120 to 130 ft/d) for the Upper Floridan aquifer in the vicinity of the test wells. The Multi-Layer Unsteady (MLU) state model used in the analyses tended to over-predict pump-test-induced drawdown at some locations and under-predict drawdown at other locations. However, that is expected because of heterogeneity within the aquifers, and the scatter plots comparing the observed and simulated drawdown response for all monitoring wells indicated a reasonable composite match of the data.

The staff issued RAI 2.4.12-09 asking the applicant to clarify whether any spatial trend or regularities are evident in the hydraulic conductivities measured by the slug tests on the LNP site. The applicant responded by providing maps of the slug test results for both the surficial and bedrock aquifers and stated that values vary across the site by up to an order of magnitude, but do not appear to show any spatial trend (ML092150960). The NRC staff determined that, based on the maps provided, the response was sufficient to meet the requested information need. However, the results of the slug tests were found to not be sufficiently representative of site aquifer conditions. These concerns are addressed in RAI 2.4.12-10, 2.4.12-11 and 2.4.12-12 discussed below.

RAI 2.4.12-10 was issued asking the applicant to clarify the apparent discrepancy in the estimated transmissivity range presented in FSAR Revision 0, Section 2.4.12.1.1 and the average transmissivity values derived from slug tests and to discuss which of these values is most representative of actual site conditions. The applicant responded by explaining that the transmissivity values presented in FSAR Revision 0, Section 2.4.12.1.1 were regional estimates from literature sources and not site-specific.

RAI 2.4.12-11 requested that the applicant justify the approach adopted for analysis of pumping tests in the FSAR. The applicant responded by providing new analyses of the three aquifer pumping tests (ML092150960). The new analyses were based on a transient multi-layer analysis using the MLU model. The applicant used an iterative analysis approach because analysis of the Upper Floridan aquifer data required the properties of the surficial aquifer as input, and analysis of the surficial aquifer data required the properties of the Upper Floridan aquifer as input. The analysis resulted in a single set of hydraulic property values that best matched the observed response at all available monitoring locations, rather than fitting separate sets of hydraulic properties to different locations. The applicant summarized the results of the aquifer pumping tests and determined that transmissivity of the Upper Floridan aquifer at the site ranged from 5760 to 6410 m^2/d (62,000 to 69,000 ft^2/d), with an assumed Upper Floridan aquifer thickness of 158.5 m (520 ft). The applicant calculated an Upper Floridan aquifer hydraulic conductivity from the revised pumping test analyses of 36.6 to 39.6 m/d (120 to 130 ft/d) based on an aquifer thickness of 158.5 m (520 ft). The NRC staff reviewed the calculation package including the pumping test methods and analyses and determined that the analysis methods are valid for the test conditions and that these tests provide a reasonable estimate of site-specific hydraulic conductivity for the Upper Floridan aquifer in the vicinity of the test wells. The hydraulic conductivity may be higher in the upper part of the aquifer and lower in the deeper part based on observations of increasing amounts of evaporate and quartz-filled porosity below depths of 121.9 m (400 ft) noted in the response to RAI 2.4.12-10 (ML092150960).

The staff issued RAI 2.4.12-12 asking the applicant to discuss selection of hydraulic conductivity estimates used in the seepage velocity calculations and whether these result in conservative estimates of groundwater velocity. The applicant responded by describing that the hydraulic conductivity estimates of 8.72 and 16.6 m/d (28.6 and 54.4 ft/d) for the surficial and Upper Floridan aquifers, respectively, were considered conservative when used as a single value to characterize hydrogeological conditions for the entire LNP site because of regional and local variability of this property within the aquifers. As a follow-up to the applicant’s response to RAI 2.4.12-12, the staff issued new RAI 2.4.12-22 asking the applicant to discuss how the seepage velocity reported in the FSAR based on a hydraulic conductivity of 16.6 m/d (54.4 ft/d) was conservative when higher hydraulic conductivity results were indicated by reanalysis of the aquifer pumping tests and the revised groundwater model (ML093620211). The applicant response described conservative assumptions in the FSAR Section 2.4.13 transport calculations including the receptor location on the property boundary and use of a 76-m (250-ft) aquifer thickness when the total Upper Floridan aquifer thickness is estimated at 158.5 m (520 ft). The applicant also referred to the slug test results ranging from 0.73 to 16.6 m/d (2.4 to 54.4 ft/d). The applicant provided a more detailed map of hydraulic conductivity estimated from calibration of the revised groundwater flow model (ML093620211) that showed transmissivity ranging between 736 and 2734 m^2/d (7,920 and 29,429 ft^2/d) between the proposed plants and the property boundary in the direction of groundwater flow. The applicant response continued to support use of a hydraulic conductivity value of 16.6 m/d (54.4 ft/d) in the seepage velocity calculations as being conservative based on regional and local variability within the aquifer. However, the applicant also provided an alternative seepage velocity calculation based on a hydraulic conductivity of 39.6 m/d (130 ft/d) and used this value for a "bounding analysis" of contaminant transport presented in the response to staff's RAI 02.04.13-13 (ML092150960).

The staff found that the hydraulic conductivity range provided by the applicant was not based on all available information. Instead, it was based only on the results of the slug tests and did not consider the new pumping test analyses provided in the response to RAI 2.4.12-10 or the results of the recalibrated version of the District Wide Regulation Model Version 2 (DWRM2) groundwater flow model (ML093620211). The range of hydraulic conductivity calculated by the applicant from the pumping tests was 36.6 to 39.6 m/d (120 to 130 ft/d) for the Upper Floridan aquifer compared to estimates of 8.72 and 16.6 m/d (28.6 and 54.4 ft/d) used in the seepage velocity calculations. The applicant's estimates of hydraulic conductivity were also low compared to the transmissivity (hydraulic conductivity multiplied by aquifer thickness) results of the recalibrated version of the DWRM2 groundwater flow model (ML093620211). The staff reviewed the follow-up RAI 2.4.12-22 requesting more information about the hydraulic conductivity estimates used in the seepage velocity calculations and determined that the hydraulic conductivity range of 36.6 to 39.6 m/d (120 to 130 ft/d) estimated from the aquifer pumping tests (ML092150960) is more representative of site conditions than the slug test results presented in LNP FSAR Revision 2, because the pumping test analysis accounts for vertical flow within and between the aquifers and because the pumping tests are affected by a much larger volume of rock within the aquifer than slug tests. The staff also found that the transmissivity values calculated from the MLU analysis of the aquifer pumping tests (ML092150960) for both the surficial and Upper Floridan aquifers fall within the ranges predicted by the revised groundwater model for the LNP site (ML093620211). The applicant revised the FSAR to include the results of the MLU aquifer test analyses.

The staff agreed with the applicant's assessment that the hydraulic conductivity may be higher in the upper part of the aquifer and lower in the deeper part of the aquifer. The staff agreed because increasing amounts of filled porosity below depths of 122 m (400 ft) were observed in samples from boreholes.

The staff issued RAI 2.4.12-14 asking the applicant to justify the use of the porous media concept for estimating seepage velocity and describe whether preferential flow paths associated with fracturing and solution cavities in carbonate rock aquifers at the LNP site should be considered when developing conservative estimates of groundwater velocity. The applicant responded by providing discussion and references concerning the use of a porous media conceptual model for flow and transport calculations in the Upper Floridan aquifer (ML092150960). The applicant included a reference to the EPA document (EPA 1989), which describes the Upper Floridan aquifer as having flow velocities that are likely to be slower than those found in “conduit-flow” aquifers. The applicant argued that the porous media concept assuming diffuse flow through interconnected pores was appropriate for developing a conservative estimate of groundwater flow velocity.

The staff reviewed the applicant's response to RAI 2.4.12-14 and determined that it would be appropriate to use a porous media conceptual model for the groundwater velocity (see page velocity) calculations if the effective porosity value used in the calculations represents the secondary porosity features (fractures and solution channels) of the groundwater flow system rather than the overall porosity of the system. The staff found that this, usually lower, secondary porosity is likely to control the first arrival of groundwater contaminants at a downgradient location within the Upper Floridan aquifer near the LNP site. However, the applicant's seepage velocity calculations presented in the LNP FSAR were based on an effective porosity estimate of 0.15 that pertains to the overall porosity of the limestone aquifer rather than the secondary fracture porosity. The applicant did not provide any site-specific measurements of effective porosity at the LNP site at the scale of the transport calculation. The staff found that published information indicates there is a possibility of preferential groundwater flow through fractures or solution cavities within the Upper Floridan aquifer in the vicinity of the LNP site (Knochenmus and Robinson 1996; Robinson 1995). According to a USGS report "Karst carbonate aquifers can be characterized by conduit flow along irregularly distributed, solution-enlarged fissures (channel porosity) in combination with diffuse flow through the more uniformly distributed, interconnected pores (rock porosity). The Floridan aquifer system of west-central Florida is in this category" (Knochenmus and Robinson 1996). Additional information from the "shallow" tracer test at the Old Tampa Well Field (Robinson 1996) demonstrates that secondary porosity features control the transport of dissolved contaminants in the Upper Floridan aquifer. The "shallow" tracer test was conducted in the upper 90 ft of the Upper Floridan aquifer over a distance of 61 m (200 ft) and resulted in an estimated effective porosity of 0.003 based on the early arrival of the tracer (Robinson 1996). The short travel time and low effective porosity was attributed to secondary aquifer porosity caused by fractures in the limestone.

Because of the lack of site-specific information about effective porosity at the scale of the contaminant transport scenario considered in Section 2.4.13, the staff issued an additional RAI 2.4.12-23 asking the applicant to provide additional discussion of how a porosity of 0.15 represented a conservative value or to justify the exclusion of in situ tests in the Upper Floridan aquifer that resulted in lower values of estimated effective porosity. The applicant responded by describing how the mean porosity value of 0.19 was calculated from porosity values compiled by the USGS for the Avon Park limestone formation (ML101740492). The applicant considered the lower porosity of 0.15 to be conservative, because it was smaller than the field-derived porosity of 0.19. The applicant also stated that, although lower values of porosity are found at some locations in the Upper Floridan aquifer, tests that produced these lower porosities were performed in the Suwannee and Ocala limestones, and these formations are more likely to have thin layers of higher conductivity rock compared to the Avon Park Formation. The applicant also described how tracer tests conducted over small distances are more likely to be dominated by flow through smaller-scale secondary porosity features but will tend to act more like an equivalent porous media over larger distances, as noted by Knochenmus and Robinson (1996). In addition, the applicant provided an alternative seepage velocity calculation based on an effective porosity of 0.05 and used this value for a "bounding analysis" of contaminant transport presented in the response to RAI 2.4.13-13 (ML101830016).

The staff reviewed the applicant's response to RAI 2.4.12-23 regarding effective porosity of the Upper Floridan aquifer (ML101740492). The staff agrees that the Avon Park limestone formation is more likely to behave as a continuous porous medium than the Suwannee or Ocala limestones. The staff also agrees that the longer travel distance of more than 1.6 km (1 mi) to an offsite groundwater user will increase the likelihood that the aquifer will behave as a continuous porous medium compared to tracer tests conducted over smaller distances. However, because of the lack of site-specific measurements of effective porosity and the difficulty of obtaining such estimates that would apply to the scale of the transport scenario, the staff does not concur that 0.15 is a conservative estimate with regard to the transport analysis. The staff concurs that the effective porosity of 0.05 proposed by the applicant as a more conservative alternative value, and used in an alternative seepage velocity calculation provided in the response to RAI 2.4.12-23, is a reasonably conservative parameter for the analysis of contaminant transport to an offsite groundwater user.

The applicant calculated seepage velocities and Darcy flux values between pairs of upgradient and downgradient monitoring wells. The applicant used the hydraulic gradient based on March 2007 water level measurements, the range of hydraulic conductivity values from the slug tests, and porosity values of 0.2 for the surficial aquifer and 0.15 for the Upper Floridan aquifer to calculate seepage velocity. The applicant determined porosity values based on four literature references. Resulting seepage velocities ranged from 0.0003 to 0.037 m/d (0.001 to 0.12 ft/d) for the surficial aquifer and 0.003 to 0.08 m/d (0.01 to 0.27 ft/d) for the Upper Floridan aquifer. The alternative seepage velocity calculation based on an effective porosity of 0.05 and hydraulic conductivity of 39.6 m/d (130 ft/d) used for the “bounding analysis” provided in RAI responses was 0.56 m/d (1.84 ft/d).

The staff reviewed calculated seepage velocities and Darcy flux values reported in FSAR Revisions 2. The use of measured gradients between pairs of monitoring wells based on March 2007 water level measurements were found to give a reasonable gradient. As discussed above, the staff does not concur that the hydraulic conductivity values from the slug tests or the porosity value of 0.15 for the Upper Floridan aquifer are conservative values in regard to the calculation of seepage velocity. The alternative seepage velocity calculation based on an effective porosity of 0.05 and hydraulic conductivity of 39.6 m/d (130 ft/d) used for the “bounding analysis” provided in RAI responses (ML101740492) was 0.56 m/d (1.84 ft/d) and the staff considers this to be a conservative value.

2.4.12.4.5 Effects of Groundwater Usage

Information Submitted by the Applicant

The applicant provided information about nondomestic groundwater use in the portion of Levy County that falls within the SWFWMD. Permitted nondomestic use in that area was stated to be 83.113 MLd (21.956 Mgd) in 2005. The applicant also described that only 29 MLd (7.677 Mgd) of that permitted amount was actually being used in 2005. Total groundwater demand in that area including non-permitted domestic use was 35.942 MLd (9.495 Mgd).

The average groundwater operational use by LNP was projected to be 4.8 MLd (1.269 Mgd) with a maximum use rate of 22.1 MLd (5.848 Mgd). The applicant stated that groundwater will also be withdrawn during temporary dewatering of site excavations and may be used for other purposes such as concrete mixing and dust control.

The applicant determined that the dewatering withdrawals and operational withdrawals of groundwater will not affect local groundwater users.

The applicant provided information about the plant water supply in an earlier section of LNP FSAR Revision 2.

NRC Staff's Technical Evaluation

The applicant's response to RAI 2.4.12-02 provided additional details of plant water supply wells including the design of the wellfield and the projected impacts of pumping on transport pathways, surrounding surface waters, and adjacent offsite groundwater users. The applicant provided the water supply well locations, number of wells, and peak and average expected flow rates (ML092150960).

The staff issued RAI 2.4.12-15 asking that the applicant "clarify the potential effects of groundwater pumping for plant water supply on groundwater levels, transport pathways, surface water, and other water users in the affected area." The applicant responded (ML092150960) by referring to the PEF source (ML092240668), which discussed MODFLOW modeling of groundwater levels, and responses to RAIs 2.4.12-02 (ML092150960) and 2.4.13-04 (ML092080078). However, the groundwater model described in the PEF source (ML092240668) was subsequently revised by the applicant as documented by PEF (ML093620211). The staff reviewed the results of the revised groundwater model as reported by PEF (ML093620211) and found that the applicant resolved RAI 2.4.12-15 by providing a defensible groundwater model that predicts the effects of pumping the water supply wells on the groundwater potentiometric surface. The staff found that the revised groundwater model achieved the goals of matching groundwater levels measured on the LNP site and in four other wells measured in the area by the USGS.

Results from the revised model simulations showed that annual average LNP groundwater usage is relatively small compared to the overall groundwater model water balance, that is, to the total amount of groundwater simulated to be flowing through the model. LNP average operational usage of 6 MLd (1.58 Mgd) represents only 0.8 percent of the total water flux (787 MLd [208 Mgd]) through the model domain. At the projected groundwater withdrawal rate, the LNP wellfield is predicted by the revised model to decrease the surficial and Upper Floridan aquifer discharge to surface waterbodies within the model domain by approximately 1.5 MLd (0.4 Mgd), or about 2 percent of the total groundwater discharge to rivers and lakes as simulated by the model.

The revised groundwater model showed that pumping of the water supply wells will have little effect on offsite groundwater users or surface waterbodies. The staff reviewed the applicant's response and determined, based on the information provided on the planned water supply wells, expected pumping rates, and the revised model calculation of water level impacts, that the response meets the requirements for this information need.

Although the staff did not independently run the applicant's model, the staff reviewed the model, including parameters used, boundary conditions, discretization, calibration results, and calculation validity, and on this basis determined that the results were adequate to estimate future impacts on groundwater use.

2.4.12.4.6 Subsurface Pathways

Information Submitted by the Applicant

In Section 2.4.12.3 of LNP FSAR Rev 4, the applicant refers to the previous Section 2.4.12.2, titled “Sources,” which discusses the locations of wells, and to Section 2.4.13.2, titled “Groundwater Scenarios,” concerning conservative analysis of critical groundwater pathways for a liquid effluent release at the site and the determination of groundwater and radionuclide travel times to the nearest downgradient groundwater user or surface waterbody.

In LNP FSAR Revision 2 Section 2.4.12.4.2, the applicant used water levels measured at onsite monitoring wells to determine flow directions and gradients. Seepage velocities and Darcy flux were calculated between pairs of upgradient and downgradient monitoring wells. Seepage velocity was calculated from the hydraulic gradient based on March 2007 water level measurements, the range of hydraulic conductivity values from the slug tests, and porosity values of 0.2 for the surficial aquifer and 0.15 for the Upper Floridan aquifer. The porosity values were determined based on four literature references. Resulting seepage velocities ranged from 0.0003 to 0.037 m/d (0.001 to 0.12 ft/d) for the surficial aquifer and 0.003 to 0.08 m/d (0.01 to 0.27 ft/d) for the Upper Floridan aquifer.

NRC Staff’s Technical Evaluation

The staff issued RAI 2.4.12-16 asking the applicant to describe plausible groundwater pathways for use in the analysis of transport of accidental liquid radioactive effluent release in the subsurface. The applicant responded by providing a discussion of the plausible potential groundwater pathways that were considered in the analysis of groundwater transport of radioactive releases to the subsurface (ML092150960). Pathways included in the RAI response considered transport to the surficial aquifer, transport from the surficial aquifer to the underlying Upper Floridan aquifer, transport through the Upper Floridan aquifer to nearby private and public wells, transport into the LNP retention pond and wetlands in the direction of groundwater movement, and transport to the Withlacoochee River. The applicant also considered the potential impact of the proposed LNP water supply wells on groundwater transport. Based on the revised groundwater model results (ML093620211), it was concluded that pumping of the supply wells could have a minor impact on groundwater transport. However, the pumping will not result in faster transport of contaminants to off-site users than under non-pumping conditions.

The staff reviewed the information provided in LNP FSAR Revision 2 and RAI responses concerning subsurface pathways for transport of radionuclides through groundwater and determined that all the plausible pathways had been considered. There are no other shallow aquifers that could provide a pathway for groundwater contaminants to move offsite and no other nearby surface water features that are considered potential receptors of groundwater contaminants.

2.4.12.4.7 Groundwater Monitoring or Safeguard Requirements

Information Submitted by the Applicant

The applicant described the monitoring programs that are planned to protect present and projected future groundwater users near the LNP site. The objectives of the groundwater monitoring programs were stated. Monitoring programs are planned for the pre-application period, construction, the preoperational period, and plant operation.

NRC Staff's Technical Evaluation

The staff issued RAI 2.4.12-17 asking the applicant to update FSAR Section 2.4.12.4 with a summary of the details of groundwater monitoring under the Radiation Protection Program included in FSAR Section 12AA.5.4.14 or describe why it is not necessary to update the FSAR with this information. The applicant stated that it added the information in FSAR Section 12AA.5.4.14 to FSAR Section 2.4.12.4 by reference (ML092150960). The staff reviewed the applicant's response and determined that the content of the referenced information is sufficient to address this information need.

2.4.12.4.8 Site Characteristics for Subsurface Hydrostatic Loading

Information Submitted by the Applicant

The applicant stated that the nominal plant grade elevation for the LNP site as 15.2 m (50 ft) NAVD88 and the nominal plant grade floor elevation for LNP 1 and LNP 2 as 15.5 m (51 ft) NAVD88. The AP1000 DCD indicates that the AP1000 is designed for a groundwater elevation up to 14.6 m (48 ft) NAVD88, which is 0.6 m (2 ft) below the nominal plant grade.

The applicant stated that twice daily groundwater elevation measurements recorded every 12 hours by pressure transducers in monitoring wells MW-13S and MW-15S, both completed in the surficial aquifer, resulted in maximum observed water levels during March 2007 and March 2008 that were more than 2.1 m (7 ft) below nominal plant grade elevation. This maximum observed water level corresponds to a water table elevation of 13.1 m (43 ft) NAVD88. The highest groundwater levels measured during quarterly monitoring events were 12.82 m (42.05 ft). These measurements were also at surficial aquifer wells MW-13S and MW-15S.

The applicant stated that "final grading of the LNP site will result in potential hydrologic alteration, including the permanent change in groundwater levels within the plant site from site grading and a series of stormwater drainage ditches…. Stormwater drainage ditches installed within the LNP site will have bottom elevations ranging from approximately 12.97 m (42.55 ft) NAVD88 or lower to approximately 14.57 m (47.80 ft) NAVD88." The applicant concluded that the LNP site meets the requirements for the AP1000 design and that "no dynamic water forces associated with normal groundwater levels will occur because of a higher finished plant grade."

NRC Staff's Technical Evaluation

The staff issued RAI 2.4.12-18 asking the applicant to provide an analysis and description of predicted post-construction groundwater conditions near the safety-related SSCs with respect to the DCD maximum allowable groundwater elevation. The applicant responded by reiterating the information in LNP FSAR Revision 2 concerning monitored water levels in comparison to the plant grade (ML092150960). The applicant referred to a calculation package concerning the effect of grouting on groundwater flow. The staff reviewed this calculation package and determined that it did not address the issue of expected groundwater level during plant operation. The applicant also referred to the response to RAI 2.4.12-02, which describes the results of a revision to the site groundwater model documented by the applicant (ML092240668). However, this model was revised by the applicant as documented by the applicant (ML093620211). The revised groundwater model shows that pumping of the water supply wells may create a drawdown of about 0.15 m (0.5 ft) at the LNP Unit 1 and Unit 2 plant locations.

As a follow-up to the applicant's response to RAI 2.4.12-18, the staff issued RAI 2.4.12-24 asking the applicant to analyze and describe the effects of alterations to the groundwater flow system, including the effects of stormwater runoff caused by the new structures and facilities and how this will affect groundwater levels near the safety-related SSCs with respect to the DCD maximum allowable groundwater elevation.

The applicant responded to RAI 2.4.12-24 by providing descriptions of alterations to the groundwater flow system and a discussion of the potential effects of each alteration on future groundwater elevations with respect to subsurface hydrostatic loading on LNP Unit 1 and LNP Unit 2 (ML101740492). The applicant will install a drainage system designed to remove runoff from up to a 50-year precipitation event. The applicant described that "the drainage system will capture and redirect rainfall and surface runoff away from safety-related SSCs to onsite ditches and retention ponds where the water will recharge, evaporate, or be pumped offsite if needed (via the cooling water tower basins)." The applicant stated that surficial aquifer groundwater elevations near safety-related SSCs would be reduced as a result of the drainage system. The applicant also stated that "if the onsite drainage system becomes blocked, the LNP site can be drained by overland flow directly to the Lower Withlacoochee River or the Gulf of Mexico." The applicant also described changes to the groundwater flow system resulting from the installation of impervious surfaces such as buildings and parking lots. The applicant stated that these impervious surfaces would result in less infiltration and reduce the potential for groundwater mounding around the safety-related SSCs during rainfall events. The applicant described planned grading of the site to drain surface flow away from the safety-related SSCs. The applicant described the planned dewatering system that will be used to lower groundwater levels around the nuclear islands during foundation emplacement and referred to a calculation package that was reviewed by the staff.

The staff issued RAI 2.4.12-25 asking the applicant to provide an estimate of the maximum post-construction groundwater level that is based on anticipated post-construction surface conditions, the anticipated properties of the fill material, the conceptual model of the subsurface, and expected maximum recharge rates. The applicant was also requested to provide proposed updates to the FSAR that would include the results of this analysis and supporting information used in the analysis.

The applicant responded by: (1) describing the planned installation of diaphragm walls at the excavation limits of the nuclear islands and grouting at the base of the excavations; (2) describing the surface grading and storm drainage system that is designed to direct stormwater and groundwater away from LNP Unit 1 and LNP Unit 2; and (3) providing the results of MODFLOW groundwater modeling performed to evaluate the maximum water table elevation (ML110800090). This modeling is distinct from the original and revised models used to investigate potential effects of groundwater usage, as described in Section 2.4.12.4.5 of this SER.

The staff reviewed the local groundwater model provided by the applicant and made independent model runs to confirm the applicant's conclusions and, in addition, to investigate the sensitivity of the model to certain parameters. Model input files were obtained from the applicant and the model parameters, boundary conditions, and results were verified. The groundwater model simulated the water table response under conditions of a 72-hr duration PMP design storm. The model divided the LNP site into specified areas of impervious surface material with no recharge of precipitation to the aquifer and areas of pervious materials that would experience a varying recharge rate calculated based on the hourly PMP precipitation rate. Three layers were implemented in the model. The top layer representing the surficial aquifer was assigned a uniform horizontal hydraulic conductivity of 2.8 m/d (9.2 ft/d) and a vertical hydraulic conductivity of 0.28 m/d (0.92 ft/d). Layers 2, 3, and 4 represented the Upper Floridan aquifer and were assigned a horizontal hydraulic conductivity of 4.2 m/d (13.9 ft/d) and vertical hydraulic conductivity of 0.4 m/d (1.39 ft/d). The horizontal hydraulic conductivity values applied to the Upper Floridan aquifer are significantly lower than the range of 36.6 to 39.6 m/d (120 to 130 ft/d) for the hydraulic conductivity determined from the MLU analyses of the applicant's pumping test. The value applied to the surficial aquifer is within the range of 0.27 to 8.72 m/d (0.9 to 28.6 ft/d) from the applicant's analysis of slug tests in the surficial aquifer. The staff determined that applying a relatively low hydraulic conductivity to the Upper Floridan aquifer model layer was conservative with regard to maximum water table elevation because a higher hydraulic conductivity would result in less mounding of the water table in response to infiltration of precipitation.

Recharge rates applied to the pervious areas of the model were calculated based on the average PMP precipitation rate during each model time step. The staff review of the model files showed that of a total of 90.7 cm (35.7 in.) of water recharged the upper layer of the model in pervious surface areas during the simulated PMP storm compared to a total PMP precipitation of 90.9 cm (35.8 in). This high rate of infiltration is a conservative factor in the analysis.

The applicant's model showed that during a PMP event, the water table elevations at the SSCs are predicted to be less than 13.7 m (45 ft) NAVD88, which is well below the 14.6 m (48 ft) NAVD88 limit defined by the DCD. The SSCs are surrounded by areas of impervious surface materials. Runoff will be routed to the stormwater drainage ditches that have bottom elevations from 13 to 14.6 m (42.5 ft to 47.8 ft) NAVD88. Based on the model results, the staff concludes that the maximum groundwater level will likely not exceed the DCD-specified maximum of

14.6 m (48 ft) NAVD88 at the safety-related structures. The water table was predicted by the model to reach the ground surface elevation of 15.2 m (50 ft) NAVD88 in some areas covered with pervious materials during a PMP design storm. However, the staff concludes that excess precipitation will runoff to the stormwater ditches and ponds and will not create a potential for groundwater levels exceeding the DCD limit.

Planned installation of diaphragm walls at the excavation limits of the nuclear islands and grouting at the base of the excavations will also reduce the potential for the water table to exceed the DCD design limit within the excavation areas. The staff determined that the planned diaphragm walls will not retain groundwater after plant construction in a way that would cause groundwater levels around the plant foundations to exceed the DCD design limit.

The applicant committed to revising the FSAR to include a description of the local-scale groundwater model and results related to estimating the expected maximum water table at safety-related structures. The staff is tracking this issue as **Confirmatory Item 2.4.12-1**.

Resolution of Confirmatory Item 2.4.12-1

Confirmatory Item 2.4.12-1 is an applicant commitment to update Section 2.4.12 of its FSAR. The staff verified that LNP COL FSAR Section 2.4.12 was appropriately updated. As a result, Confirmatory Item 2.4.12-1 is now closed.

2.4.12.5 *Post Combined License Activities*

There are no post COL activities related to this section.

2.4.12.6 *Conclusion*

The staff has reviewed the application and has confirmed that the applicant addressed the information relevant to groundwater, and that there is no outstanding information required to be addressed in the COL FSAR related to this section.

As set forth above, the applicant presented and substantiated information to establish the site description. The staff has reviewed the information provided and, for the reasons given above, concludes that the applicant has provided sufficient details about the site description for the staff to determine, as documented in Section 2.4.12 of this SER, that the applicant has met the relevant requirements of 10 CFR 52.79(a)(1)(iii) and 10 CFR Part 100 with respect to determining the acceptability of the site. This addresses COL information item 2.4-4.

2.4.13 Accidental Release Of Radioactive Liquid Effluent In Ground And Surface Waters

2.4.13.1 *Introduction*

FSAR Section 2.4.13 provides a characterization of the attenuation, retardation, dilution, and concentrating properties governing transport processes in the surface water and groundwater environment at the site. This section's goal is not to assess the impacts of all possible specific release scenarios, but to provide a suitable conceptual model of the transport through the hydrological environment for possible later use in other assessments. Because it would be impractical to characterize all the physical and chemical properties (e.g., hydraulic conductivities, porosity, mineralogy) of a time-varying and heterogeneous environment, FSAR Section 2.4.13 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 on 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 (NRC 2007a) Section 11.2 following the guidance in Branch Technical Position (BTP) 11-6, "Postulated Radioactive Releases Due to Liquid-containing Tank Failures" (NRC 2007d). The source term is determined from a postulated release from a single tank outside of the containment. The tank having the greatest potential inventory of radioactive materials is assumed as the source of the release.

Section 2.4.13 of this SER presents an evaluation of the following specific areas: (1) alternative conceptual models of the hydrology at the site that reasonably bound hydrogeological conditions at the site inasmuch as 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 impact on existing uses and known and likely future uses of groundwater and surface water resources in the vicinity of the site; (3) ability of the groundwater and surface water environments to delay, disperse, dilute, or concentrate accidentally released radioactive liquid effluents during transport; and (4) assessment of scenarios wherein an accidental release of radioactive effluents is combined with potential effects of seismic and non-seismic events (e.g., assessing effects of hydraulic structures located upstream and downstream of the plant in the event of structural or operational failures and the ensuing sudden changes in the regime of flow); and (5) any additional information requirements prescribed in the "Contents of Application" sections of the applicable subparts to 10 CFR Part 52.

2.4.13.2 *Summary of Application*

This section of the COL FSAR addresses the accidental release of radioactive liquid effluents in groundwater and surface waters. The applicant addressed these issues as follows:

<u>*AP1000 COL Information Item*</u>

- LNP COL 2.4-5

This COL item is addressed by FSAR Section 2.4.13. In particular, this section addresses the following COL-specific information that is defined and discussed in Section 2.4.1.5 of Revision 19 of the AP1000 DCD.

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 the Commission regulations for the pathways of liquid effluents in groundwater and surface water, and the associated acceptance criteria, are described in Section 2.4.13 of NUREG-0800 (NRC 2007a).

The applicable regulatory requirements for liquid effluent pathways for groundwater and surface water are as follows:

- 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), as it 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.

- 10 CFR 20, as it relates to effluent concentration limits.

- 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.

Appropriate sections of the following documents are used for the related acceptance criteria:

- BTP 11-6 (NRC 2007d) provides guidance in assessing 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.

- Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I" (NRC 1977b)

2.4.13.4 *Technical Evaluation*

The NRC staff reviewed Section 2.4.13 of the LNP 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 accidental releases of radioactive liquid effluents in ground and surface waters. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

2.4.13.4.1 Radioactive Tank Rupture

Information Supplied by the Applicant

The applicant selected the accidental release to groundwater scenario based on information provided by the AP1000 reactor vendor. According to the applicant, the scenario is an instantaneous release from one of the two effluent holdup tanks located in the lowest level of the AP1000 auxiliary building. Each effluent holdup tank holds 105,992 L (28,000 gallons). The failed tank was assumed to have maximum radionuclide concentrations corresponding to101 percent of the reactor coolant source term. It was assumed that 80 percent of the tank's volume, or 84,793 L (22,400 gal) is released. The applicant provided the expected tank inventory in LNP FSAR Revision 2 Table 2.4.13-202. The applicant described the effluent holdup tanks as having the highest potential radionuclide concentration and the largest volume and, therefore, release from one of those tanks was considered a conservative selection for the purpose of calculating the potential for contamination of groundwater.

The applicant assumed that the effluent release occurs at the bottom floor of the auxiliary building and directly to the Floridan aquifer. No credit was taken for transit time through the walls of the auxiliary building, or through the surficial aquifer that overlies the Floridan aquifer. The bottom floor of the auxiliary building was described as 10.4 m (34 ft) below the design plant grade of 15.2 m (50 ft) elevation (NAVD88). The applicant considered a release directly to the Floridan aquifer to be conservative because the analysis does not take credit for transit time through the surficial aquifer and because the Floridan aquifer has higher seepage velocities than the surficial aquifer.

The applicant considered two transport cases. The first case was transport to a well completed in the Upper Floridan aquifer located on the LNP site boundary in the direction of groundwater flow at a distance of 2 km (1.2 mi). The second case considered groundwater transport to the Lower Withlacoochee River downgradient from LNP Units1 and 2 at a distance of approximately 6.9 km (4.3 mi).

The applicant determined the direction of groundwater flow to the southwest by examining observed groundwater head contour maps based on water levels measured in the onsite monitoring wells.

NRC Staff's Technical Evaluation

The staff reviewed the accidental release scenario and conceptual model. The tank rupture scenario was determined to be conservative because it assumes that 80 percent of the tank volume is instantaneously transmitted into the aquifer and this volume contains 101 percent of the coolant source term. The two transport cases are evaluated in the following section.

2.4.13.4.2 Groundwater Scenarios

Information Supplied by the Applicant

LNP FSAR Rev 2 stated that "The surficial aquifer is not a well-developed aquifer system near the LNP site and no users of surface water have been identified near the LNP site. … The Floridan aquifer is the principal source of potable water near the LNP site." Therefore, the transport analysis was based on immediate release to the Floridan aquifer with no credit for transport time through the containment building or through the surficial aquifer.

The applicant calculated transport of radionuclides in groundwater using the analytical equation for three-dimensional, transient transport in a saturated porous medium with one-dimensional, steady advection in the x-direction, three-dimensional dispersion, linear equilibrium adsorption, and first-order decay. However, LNP FSAR Revision 2 states "The maximum concentration at a well in the Floridan aquifer is taken as the aquifer's concentration at the distance downgradient from the point of release with vertical mixing assumed in the aquifer." Therefore, the analysis assumes that the radionuclides are completely mixed over the assumed 76.2-m (250-ft) thickness of the aquifer.

The applicant identified key parameters used in radionuclide transport calculations. Seepage velocities used in the calculation were presented in Section 2.4.12 of LNP FSAR Rev 4. Distribution coefficients (K_d) for cesium and strontium were selected using EPA (1999) guidance for conservative selection of distribution coefficients. Other radionuclides were given K_d of zero, indicating no sorption. FSAR Rev. 4 references NUREG/CR-3332 (EPA 1983) to show that longitudinal dispersivity of αL = 10 to 15 m (32.8 to 49.2 ft) for limestone and carbonate aquifers are reasonable. However, the evaluation presented in FSAR Rev. 4 conservatively assumed longitudinal and transverse dispersivities of αL = 1 m and αL*αT = 1 m^2, respectively. Lower dispersivity values used in the analysis will result in higher concentrations of radionuclides at the receptor locations.

The LNP FSAR Revision 4 calculations of maximum activity concentrations in well water from a release to the Floridan aquifer resulted in an effective dose equivalent of less than 0.7 percent of the regulatory allowable activity. Tritium was found to be responsible for essentially the entire dose for water use derived from the well. The applicant also calculated radionuclide concentrations and resulting dose equivalents in the Lower Withlacoochee River. The calculated effective dose equivalent for the river water was negligible when compared to allowable limits.

NRC Staff's Technical Evaluation

The staff issued RAI 2.4.13-02 asking the applicant to describe the process followed to ensure that the most conservative of plausible conceptual models were identified. The applicant responded with additional details concerning the identification of groundwater and surface water users, general site characteristics, and plausible surface and subsurface pathways (ML092080078). The most conservative conceptual models identified were (1) transport to a groundwater user located 2 km from the spill through the Upper Floridan aquifer with no credit for transport time through the containment building or through the surficial aquifer, and (2) contaminated groundwater entering the Withlacoochee River 7 km (4.3 mi) away from the spill also with no credit for transport time through the containment building or through the surficial aquifer.

The staff issued RAI 2.4.13-03 asking the applicant to clarify the total thickness of the Upper Floridan aquifer at the LNP site. The applicant responded by providing additional information about the thickness of the Upper Floridan aquifer above the MCU (ML092080078) and revised the FSAR discussion in Section 2.4.13.2. The applicant RAI response stated “Based on limited downhole geophysical testing and monitoring of drilling fluid losses at the LNP site, the most productive interval of the Upper Floridan aquifer appears to be at depths of approximately 30 to 60 m (100 to 300 ft) bgs.” However, 60 m would be equivalent to about 200 ft. The applicant used an aquifer thickness of 76.2 m (250 ft) in the assessment of an accidental release of radioactive effluents in groundwater. As a follow-up to the applicant’s response to RAI 2.4.13-03, the staff issued a new RAI 2.4.13-12 asking the applicant to clarify the apparent discrepancy regarding the depth of the most productive interval of the Upper Floridan aquifer. The applicant responded that the depth of 60 m is incorrect and the correct depth is 91 m, which corresponds to the 91.4-m (300-ft) value in FSAR Revision 2.

As a follow-up to RAI 2.4.13-02, the staff issued RAI 2.4.13-13 requesting that the applicant provide a discussion of the degree of conservatism in the transport analysis regarding (1) parameters used in seepage velocity calculations, (2) the assumption that the released contamination is evenly distributed over an aquifer thickness of 76.2 m (250 ft), and (3) the use of a groundwater head gradient in the transport analysis that is smaller than the gradient calculated from the potentiometric map for the Upper Floridan aquifer presented in the recalibrated version of the groundwater flow model (ML093620211), which is based on a more extensive well network. The applicant responded by describing a number of conservative assumptions in the analysis, including the receptor location on the site boundary and the direct release of effluent to the Upper Floridan aquifer (ML101830016). The applicant's response also discussed the hydraulic conductivity and effective porosity values, the aquifer thickness used in the analysis, and hydraulic gradients. Although the applicant defended the parameters and assumptions used in the FSAR analysis, the applicant also provided an “alternate evaluation” of groundwater transport through the Upper Floridan aquifer based on more conservative assumptions concerning aquifer hydraulic conductivity and effective porosity that reflect the potential for preferential flow paths within the fractured limestone aquifer. The parameters used in the alternate evaluation and the alternate transport analysis results, including the sum of fractions of the predicted concentration/Effluent Concentration Limits (ECL) reported in the RAI response, are listed below:

Alternate Analysis Parameters (different from original analysis):

- Hydraulic conductivity = 39.6 m/d (130 ft/d)
- Effective porosity = 0.05

Alternate Analysis Results:

- Linear velocity = 0.56 m/d (1.8 ft/d)
- Concentration/ECL – all nuclides = 54 percent (at offsite groundwater well)
- Peak time – tritium = 9.8 yr (at offsite groundwater well)
- Peak concentration – tritium = 5.2E-04 μCi/cm^3 (at offsite groundwater well)
- Concentration/ECL – tritium only = 52 percent (at offsite groundwater well)

The alternate transport analysis used the same aquifer thickness (76.2 m [250 ft]) and gradient as were used in the FSAR Revision 2 analysis.

The applicant also provided an analysis of vertical dispersion for comparison with the assumption of complete vertical mixing over the assumed 76.2 m (250 ft) aquifer thickness to address the staff concern. The analysis showed that for a contaminant not affected by decay or retardation, the vertical distribution of contaminant concentrations at the top and bottom of the 76.2-m (250-ft) aquifer are within 7 percent of “fully mixed” when the center of the plume has moved 2 km (1.24 mi) from the release point. The analysis was based on the parameters applied in the LNP FSAR Revision 2 transport calculations.

In the response to RAI 2.4.13-13 (ML101830016), the applicant compared groundwater gradients from onsite measurements to the potentiometric map for the Upper Floridan aquifer presented in the recalibrated version of the groundwater flow model (ML093620211). The potentiometric map was based on some wells located in an area of higher groundwater levels more than 4 mi northeast of the LNP site and on synthetic wells based on modeled USGS water level contours. The applicant presented the data to show that the gradient of 0.0007 used in transport modeling is at the upper range calculated from onsite well measurements for the direction of groundwater flow from the reactor locations toward the receptor well.

The staff reviewed the applicant's responses to RAI 2.4.13-02 (ML092080078) and RAI 2.4.13-13 (ML101830016) and determined that the release to groundwater scenarios for contaminant transport presented in the FSAR are conservative except with regard to values of saturated hydraulic conductivity (16.6 m/d [54.4 ft/d]) and effective porosity (0.15) used in the seepage velocity calculations. The staff determined that the applicant's "alternate evaluation" of groundwater transport through the Upper Floridan aquifer provides a conservative analysis of the pathway associated with an accidental spill to groundwater. The alternate analysis was based on a higher (more conservative) saturated hydraulic conductivity (39.6 m/d [130 ft/d]) from MLU analysis of the aquifer pumping test and a lower (more conservative) effective porosity (0.05) that reflects the possibility of preferential flow paths within the fractured limestone aquifer. Other parameters used in the alternate evaluation matched those used in the FSAR analysis.

The staff also reviewed the discussion and analysis of vertical dispersion provided in response to RAI 2.4.13-13 (ML101830016). The analysis showed that for a contaminant not affected by decay or retardation, the vertical distribution of contaminant concentrations at the top and bottom of the assumed 250-ft aquifer are within 7 percent of “fully mixed” when the center of the plume has moved 2 km (1.24 mi.) from the release point. The analysis was based on the parameters applied in the LNP FSAR Revision 2 transport calculations. The staff considers the analysis based on a contaminant not affected by decay or retardation to be appropriate because tritium is the primary dose contributor.

The staff issued RAI 2.4.13-04 asking the applicant to “discuss LNP groundwater usage from the Upper Floridan aquifer in relation to the projected impacts of pumping on subsurface radionuclide transport pathways at the LNP site.” Related RAIs, 2.4.12-02 and 2.4.12-24, asked the applicant to discuss the effects of alterations to the groundwater flow system, including details of plant water supply wells and the projected impacts of pumping on transport pathways, surrounding surface waters, and adjacent offsite groundwater users. The applicant responded (ML092080078) with additional information about the planned water supply wells and discussed the results of a site groundwater model (ML092240668). However, this model was subsequently revised by the applicant based on an RAI related to the LNP EIS. The new revision of the groundwater model was documented by the applicant (ML093620211). The applicant's revised groundwater flow model (ML093620211) predicts drawdown of 0.46 to 0.61 m (1.5 to 2 ft) in the southern portion of the LNP site after 1 year caused by operation of the water supply wells. This would result in a larger gradient to the south. A 0.6-m (2-ft) decrease in head near the water supply wells, about 2.4 km (1.5 mi) from the release point, would result in a gradient increase from 0.0007 to 0.00095 based on the revised model results. However, pumping at the supply wells would also result in a longer south-southwest flow path to the site boundary of about 3.2 km (2 mi), which would result in a slightly longer travel time than that calculated based on the gradient and flow path used in the LNP FSAR Revision 2 analysis.

The staff reviewed the applicant response to RAI 2.4.13-04 regarding the impact of groundwater usage from the Upper Floridan aquifer, including pumping of the proposed plant water supply wells on subsurface radionuclide transport pathways. The staff concurs that the water table may experience drawdown of 0.5 to 0.6 m (1.5 to 2 ft) in the southern portion of the LNP site after 1 year because of the water supply wells and this would result in a larger gradient to the south. However, the change in water table configuration would result in a longer south-southwest flow path to the site boundary of about 3.2 km (2 mi), which would result in a slightly longer travel time than that calculated based on the gradient and flow path used in the LNP FSAR Revision 2 analysis. The staff also agrees that the onsite measurements used by the applicant in gradient calculations are more representative of groundwater flow conditions along the hypothetical transport path than the potentiometric map for the Upper Floridan aquifer presented in the recalibrated version of the groundwater flow model (ML093620211), because the potentiometric map was based on some wells located in an area of higher groundwater levels more than 6.4 km (4 mi) northeast of the LNP site and on synthetic wells based on modeled USGS water level contours.

RAI 2.4.13-05 asked the applicant to discuss why assuming a release at the top of the Floridan aquifer is conservative and whether a release to the surficial aquifer could result in a pathway to surface water, such as the Withlacoochee River, and including marshes or ditches at the LNP site that are closer than the nearest offsite well. The applicant responded (ML092080078) by explaining that the release would occur about 7.6 m (25 ft) below the top of the surficial aquifer, and about 7.6 m (25 ft) above the top of the Floridan aquifer. Downward head gradients within the surficial aquifer would make radionuclides migrate downward to the Floridan aquifer. The applicant also provided additional information about the site topography and surface features and the planned surface water drainage system.

The staff concurs with the applicant's response to RAI 2.4.13-05 that a release to surface water is not likely because of the location of the release 10.4 m (34 ft) below the nominal plant grade elevation. The measured downward vertical hydraulic gradient would also make it unlikely that contaminants would migrate upward through the surficial aquifer. It is unlikely that contaminants would migrate from this depth to marshes or ditches at the LNP site that are closer than the nearest offsite well. RAI 2.4.13-06 stated that “PEF needs to clarify why use of the one-dimensional advection-dispersion equation for solute transport in porous media is appropriate at the LNP site.” The applicant responded (ML092080078) with additional information and references describing groundwater flow and transport characteristics expected for the Upper Floridan aquifer. The applicant presented evidence that groundwater flow between the LNP plant locations and an offsite receptor well is expected to be laminar and dispersive and follow Darcy's law. The applicant response also provided sensitivity calculations showing the effects of higher pore velocities (compared with those in Section 2.4.12 of FSAR Revision 1) on the total dose calculated at the hypothetical downgradient well.

The staff reviewed the applicant's response to RAI 2.4.13-06 regarding use of the one-dimensional advection-dispersion equation for solute transport in porous media. The staff agrees that groundwater flow between the LNP plant locations and an offsite receptor well is expected to be laminar and follow Darcy's law.

The staff issued RAI 2.4.13-07 asking the applicant to describe the computer software used to implement the mathematical model described in FSAR Section 2.4.13.2.1. Verification and validation procedures used to verify the accuracy of the model, as implemented in the software, were also requested. The applicant responded (ML092080078) by providing additional information about the calculation method, the Project Quality Plan and verification review procedures.

RAI 2.4.13-08 asked the applicant to list the sources of the model parameters listed in FSAR Table 2.4.13-203. The applicant response (ML092080078) provided a table listing the requested model parameters and notes with information about the sources. The applicant revised the FSAR by substituting the new Table 2.4.13-203.

The staff issued RAI 2.4.13-09 asking the applicant to provide the tritium concentration as a function of time in the FSAR, or justify why this information is not necessary. The applicant responded (ML092080078) by stating that “Because the evaluation for meeting 10 CFR 20 criteria is made using the maximum nuclide concentrations, the criteria is satisfied for all other times.” These maximum calculated nuclide concentrations are shown in the FSAR. The applicant's response also included plots of tritium concentration over time from the transport calculations and noted that almost the entire dose at the receptor locations is caused by tritium. The applicant also noted that the sum of all of the ratios of radionuclide concentrations to concentration limits are also provided in the FSAR to demonstrate that the criteria for mixtures are satisfied. The applicant made minor wording changes to the FSAR discussion in Section 2.4.13.2. The staff agrees that the radionuclide concentrations over time do not need to be shown in the FSAR as long as the maximum concentration over time is stated and is used in the evaluation for meeting the 10 CFR 20 criteria.

In RAI 2.4.13-10, the staff requested that the applicant provide site-specific measurements of K_d as required by 10 CFR 100.20(c)(3). The applicant had used literature-based values of K_d for the transport analysis described in FSAR Revision 2. In a letter dated July 22, 2009, the applicant provided laboratory measurements of K_d values on 16 soil and rock samples from the site. The applicant showed that using the site-specific K_d values in the transport analysis did not significantly change the results of the transport calculations. The applicant revised the FSAR by adding information about the site-specific K_d measurements.

The staff issued RAI 2.4.13-11 asking the applicant to discuss the potential impacts of chelating agents on K_d values and on radionuclide transport in the FSAR. In response to RAI 2.4.13-11, the applicant stated that only cesium and strontium were given non-zero K_d in the transport calculation. The applicant provided evidence from the literature that the transport behavior of cesium is not likely to be strongly influenced by chelating agents. The applicant also stated that cesium and strontium are unlikely to form complexes with chelating agents in groundwater because of the abundance of competing calcium and magnesium ions (ML092080078). The staff reviewed this information and determined that, based on the evidence for minor influence of chelating agents on cesium and strontium behavior in the groundwater and minor impact on the calculated sum of radionuclides at the receptor locations, the applicant's response meets this information need.

The staff evaluation confirmed that assuming immediate release to the Upper Floridan aquifer with no credit for transport time through the containment building or through the surficial aquifer was a conservative assumption. This pathway is the most conservative of the plausible pathways discussed in Section 2.4.12. The hypothetical release occurs about 7.6 m (25 ft) below the top of the surficial aquifer and 7.6 m (25 ft) above the top of the Upper Floridan aquifer. The measured downward vertical flow gradient makes it unlikely that contaminants will migrate upward to wetlands or other receptors at the ground surface. The applicant did not take credit for time required for released contaminants to migrate from inside the auxiliary building through the surficial aquifer sediments or through the diaphragm wall that will extend about 30 ft into the pressure grouted limestone at the top of the Upper Floridan aquifer (LNP FSAR Revision 4 Section 2.5.4.6. The diaphragm walls are specified to be a minimum of 1.1 m (3.5 ft) thick. The staff checked site borehole logs to verify that there is approximately 7.6 m (25 ft) of surficial aquifer sediment below the release elevation and above the top of the Upper Floridan aquifer.

To summarize, the staff reviewed the transport calculation equations provided in LNP FSAR Rev 2 and determined that they are consistent with the solutions given in NUREG/CR-3332 Section 4.5.3 (EPA 1983). The values used by the applicant for K_d and dispersivity parameters were found to be conservative estimates for the Upper Floridan aquifer. However, the seepage velocity values used in the transport calculations were found to not be conservative in the analysis presented in LNP FSAR Revision 2. These issues were addressed in RAIs issued to the applicant and ultimately resulted in the applicant providing an “alternate analysis” of groundwater transport through the Upper Floridan aquifer based on more conservative assumptions concerning aquifer hydraulic properties.

The staff determined that the applicant's “alternate analysis” of groundwater transport provided in response to RAI 2.4.13-13 (ML101830016) presents a conservative calculation of the potential dose impacts from a release of radioactive liquid effluent to groundwater. The hydraulic conductivity and effective porosity values used in the alternative analysis are conservative yet conceivable estimates of the conditions found in this portion of the Upper Floridan aquifer. The selected pathway through the Upper Floridan aquifer to a groundwater user is the most conservative of the reasonably foreseeable pathways based on the available site data. Although there is uncertainty in some of the parameters used in the analysis and more conservative parameter values are possible, the very conservative assumption of not accounting for migration time through the containment building, the diaphragm walls and grouted limestone, or the 7.6-m (25-ft) thickness of surficial aquifer, through which radionuclides would migrate downward, results in calculated travel times that are bounding. Including transport through the dewatering structure would result in travel times more than double those calculated in the alternative analysis. The assumption of complete mixing of contaminants over the aquifer thickness is not conservative, but the applicant has demonstrated that the predicted radionuclide concentrations at the offsite receptor location will be less than 10 percent lower than the values calculated using a vertical dispersion model. This is compensated by use of a 76.2-m (250-ft) rather than a 91.4-m (300-ft) aquifer thickness.

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 has reviewed the application and has confirmed that the applicant addressed the relevant information and there is no outstanding information required to be addressed in the 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 has 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, for the staff to determine, as documented in Section 2.4.13 of this SER, that the applicant has met the relevant requirements of 10 CFR 52.79(a)(1)(iii) 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 effluent concentration limits. This addresses COL information item 2.4-5.

2.4.14 Technical Specifications and Emergency Operation Requirements

2.4.14.1 *Introduction*

FSAR Section 2.4.14 of the LNP COL application describes the 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 SER presents an evaluation of the following specific areas: (1) control of hydrological events, as determined in previous hydrology sections of the FSAR, to identify the bases for emergency actions required during these events; (2) the amount of time available to initiate and complete emergency procedures before the onset of conditions while controlling hydrological events that may prevent such action; (3) review of 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; (4) potential effects of seismic and non-seismic information about the postulated technical specifications and emergency operations for the proposed plant site; and (5) any additional information requirements prescribed in the "Contents of Application" sections of the applicable subparts to 10 CFR Part 52.

2.4.14.2 *Summary of Application*

This subsection of the COL FSAR addresses technical specifications and emergency operation requirements. The applicant addressed the information as follows:

AP1000 COL Information Item

- LNP COL 2.4-6

In addition, this section addresses the following COL-specific information identified in Section 2.4.1.6 of Revision 19 of the AP1000 DCD.

> 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 the Commission regulations for consideration of emergency protective measures, and the associated acceptance criteria, are described in Section 2.4.14 of NUREG-0800 (NRC 2007a).

The applicable regulatory requirements are as follows:

- 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), as it 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.
- 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 50.36, 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.

2.4.14.4 *Technical Evaluation*

The NRC staff reviewed Section 2.4.14 of the LNP 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 technical specifications and emergency operation requirements. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

Information Submitted by the Applicant

The applicant stated that the AP1000 design does not have a safety-related cooling-water system. The applicant also stated that flooding of the safety-related facilities is not a concern at the LNP site. The applicant concluded that no emergency protective measures are needed at the LNP site for hydrology-related adverse events.

NRC Staff's Technical Evaluation

The NRC staff has concluded in previous sections of this SER that floods caused by natural phenomena at and near the LNP site would not result in inundation of the plant grade. The AP1000 design does not use a safety-related cooling-water system. Therefore, the staff concluded that no technical specification or emergency procedures related to hydrologic events are required at the LNP 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 COL FSAR related to this section.

As set forth above, the applicant has presented and substantiated site-specific information related to technical specifications and emergency operations. The staff has reviewed the information provided and, for the reasons given above, concludes that the applicant has provided sufficient details about the site description for the staff to determine, as documented in Section 2.4.14 of this SER, that the applicant has met the relevant requirements of 10 CFR 52.79(a)(1)(iii) and 10 CFR Part 100 with respect to determining the acceptability of the site. This addresses COL Information Item 2.4-6.

2.5 Geology, Seismology, and Geotechnical Engineering

In Section 2.5, “Geology, Seismology, and Geotechnical Engineering,” of the Levy Nuclear Plant (LNP) Units 1 and 2 Final Safety Analysis Report (FSAR), the applicant described geologic, seismic, and geotechnical engineering characteristics of the proposed combined license (COL) site. Following the U.S. Nuclear Regulatory Commission (NRC) guidance in Regulatory Guide (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,” the applicant defined the following four zones around the LNP COL site and conducted technical investigations in these zones:

Site region – Area within a 320-kilometer (km) (200-mile (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 LNP Units 1 and 2.

The applicant referred to the FSAR prepared by Florida Power Corporation (Florida Power Corporation, 1976) for the Crystal River Unit 3 Nuclear Generating Plant (CR3), located about 18 km (11 mi) southwest of the LNP COL site, to provide limited information deemed pertinent for understanding the geologic setting of the LNP site, particularly in regard to karst development. However, most material in Section 2.5 of the LNP COL FSAR draws on information developed from sources published since the CR3 site’s FSAR, as well as data derived from geologic, seismic, and geotechnical engineering investigations performed specifically for characterization of the LNP site.

The applicant used seismic source models previously published by the Electric Power Research Institute (EPRI, 1986 and 1989) as the starting point for characterizing potential regional seismic sources and vibratory ground motion resulting from those sources. The applicant then updated these EPRI seismic source models in light of more recent data and evolving knowledge. The applicant also replaced the original EPRI ground motion models (EPRI, 1989) with more recent EPRI models (EPRI, 2004), and then applied the performance-based approach described in RG 1.208 to develop the ground motion response spectra (GMRS) for the LNP site. The applicant revised its original GMRS calculations presented in LNP COL Revisions 1 through 4 by scaling up the original GMRS by a factor of 1.212. This scaling factor is the same factor applied to the foundation input response spectra (FIRS) in compliance with the requirement in 10 CFR Part 50, Appendix S, that the horizontal component of the FIRS in the free-field at the foundation level of the structure be a response spectrum with a minimum PGA of 0.1g.

In addition, to address recommendations of the Fukushima Near-Term Task Force described in SECY-12-0025 and evaluate potential seismic hazards at the LNP site in light of these recommendations, the applicant performed sensitivity studies using the central and eastern United States seismic source characterization (CEUS SSC) model presented in NUREG-2115.

The GMRS calculated using the CEUS SSC model combined with the updated cumulative absolute velocity (CAV) filter methodology, as described in SECY-12-0025, is enveloped by the scaled GMRS based on the updated EPRI-SOG model with full CAV, except the maximum exceedance of 4 percent near 1 Hz.

As discussed further in SER Section 20.1, based on its review of the applicant’s two seismic hazard evaluations using the EPRI-SOG model and CEUS SSC model using the updated CAV filter, the staff concludes that the LNP GMRS, FIRS, and performance based soil response spectra (PBSRS) calculated by the applicant using the CEUS SSC model are either bounded by the respective spectra calculated by the applicant using the updated EPRI-SOG model, or are within a range of percentage error expected for those calculations. Therefore, it is not necessary for the applicant to update the UHRS, GMRS, FIRS, and PBSRS calculated using the updated EPRI-SOG model.

This safety evaluation report (SER) for Section 2.5 is divided into five main parts, SER Sections 2.5.1 through 2.5.5, which parallel the five FSAR sections prepared by the applicant for the LNP COL application. The five SER 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” (including information regarding embankments and dams). These SER sections present the staff’s evaluations and conclusions in regard to the geologic, seismic, and geotechnical engineering characteristics for proposed LNP Units 1 and 2.

2.5.1 Basic Geologic and Seismic Information

2.5.1.1 *Introduction*

LNP COL FSAR Section 2.5.1 describes the basic geologic and seismic information collected by the applicant during site characterization investigations. This information addresses both regional and site-specific geologic and seismic characteristics. The investigations included surface and subsurface field studies, performed by the applicant at progressively greater levels of detail closer to the site within each of four circumscribed areas, which correspond to site region, site vicinity, site area, and site location, as previously defined. The applicant conducted these investigations to assess geologic and seismic suitability of the site; determine whether new geologic or seismic data exist that could significantly impact seismic design based on the results of probabilistic seismic hazard analysis (PSHA); and to provide the geologic and seismic data appropriate for plant design.

2.5.1.2 *Summary of Application*

Section 2.5 of the LNP COL FSAR, Revision 9, incorporates by reference Section 2.5.1 of the AP1000 Design Control Document (DCD), Revision 19.

In addition, in LNP COL FSAR Section 2.5.1, the applicant provided site-specific supplemental information to address the following:

AP1000 COL Information Item

- LNP COL 2.5-1

The applicant provided additional information in LNP COL 2.5-1 to address COL Information Item 2.5-1 (COL Action Item 2.5.1-1). LNP 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 area.

LNP COL FSAR Section 2.5.1 is divided into two main sections. FSAR Section 2.5.1.1, “Regional Geology,” describes physiography and topography; geologic history; stratigraphy, including general characteristics of carbonate terrain; and tectonic setting, including seismicity, within the LNP site region. FSAR Section 2.5.1.1 also discusses significant seismic sources outside the site region. FSAR Section 2.5.1.2, “Site Geology,” addresses physiography and topography, including characteristics of marine terraces and karst terrain; geologic history; stratigraphy, including carbonate units and karst phenomena; and structural geology within the LNP site vicinity and site area. FSAR Section 2.5.1.2 also discusses geomorphology and stratigraphy, including karst development, at the site location, and evaluates geologic hazard and engineering geology of the site area and site location, respectively.

The applicant developed LNP COL FSAR Section 2.5.1 based on information derived from maps and reports published by state and federal agencies and research workers; remote sensing imagery and aerial photographs; digital elevation models (DEMs); oil and gas exploration programs; communications with researchers familiar with previous investigations in the site region, site vicinity, and site area; and geologic and geotechnical field studies performed specifically for characterization of the LNP site location, site area, and site vicinity. The applicant also provided limited information deemed pertinent for understanding the geologic setting of the LNP site, particularly in regard to karst development, as derived from the CR3 FSAR (Florida Power Corporation, 1976).

Based on the geologic and seismic investigations performed for LNP Units 1 and 2, the applicant concluded in FSAR Section 2.5.1 that no geologic or seismic conditions exist at the site, which would negatively impact the construction or operation of safety-related structures. The applicant further concluded that possible non-tectonic surface deformation related to dissolution of carbonate and resultant collapse or subsidence is the only potential geologic hazard in the site area, and that this hazard will be mitigated either during construction or by appropriate design. A summary of the basic geologic and seismic information the applicant provided in LNP COL FSAR Section 2.5.1 is presented below.

2.5.1.2.1 Regional Geology

FSAR Section 2.5.1.1 discusses the physiography and topography, geologic history, stratigraphy, and tectonic setting of the LNP site region, defined as that area which lies within a 320-km (200-mi) radius of the site. In the discussion of regional tectonic setting, the applicant also addressed regional seismicity and significant seismic sources at a distance greater than 320 km (200 mi) from the site. The following sections summarize the information the applicant provided in FSAR Section 2.5.1.1.

2.5.1.2.1.1 Regional Physiography and Topography

FSAR Section 2.5.1.1.1 describes physiography and topography of the Coastal Plain physiographic province in the site region, including the Sea Island, East Gulf, and Floridian sections of that physiographic province. SER Figure 2.5.1-1 (reproduced from FSAR Figure 2.5.1-201) shows the location of the LNP site in relation to these three sections of the Coastal Plain physiographic province, the Florida peninsula, and the Floridian plateau. The region containing the Floridian plateau and the Florida peninsula separates the Gulf of Mexico from the Atlantic Ocean and makes up the Florida platform. The LNP site lies on the Gulf side of the Florida peninsula, atop the Florida platform, in the Floridian section of the Coastal Plain physiographic province.

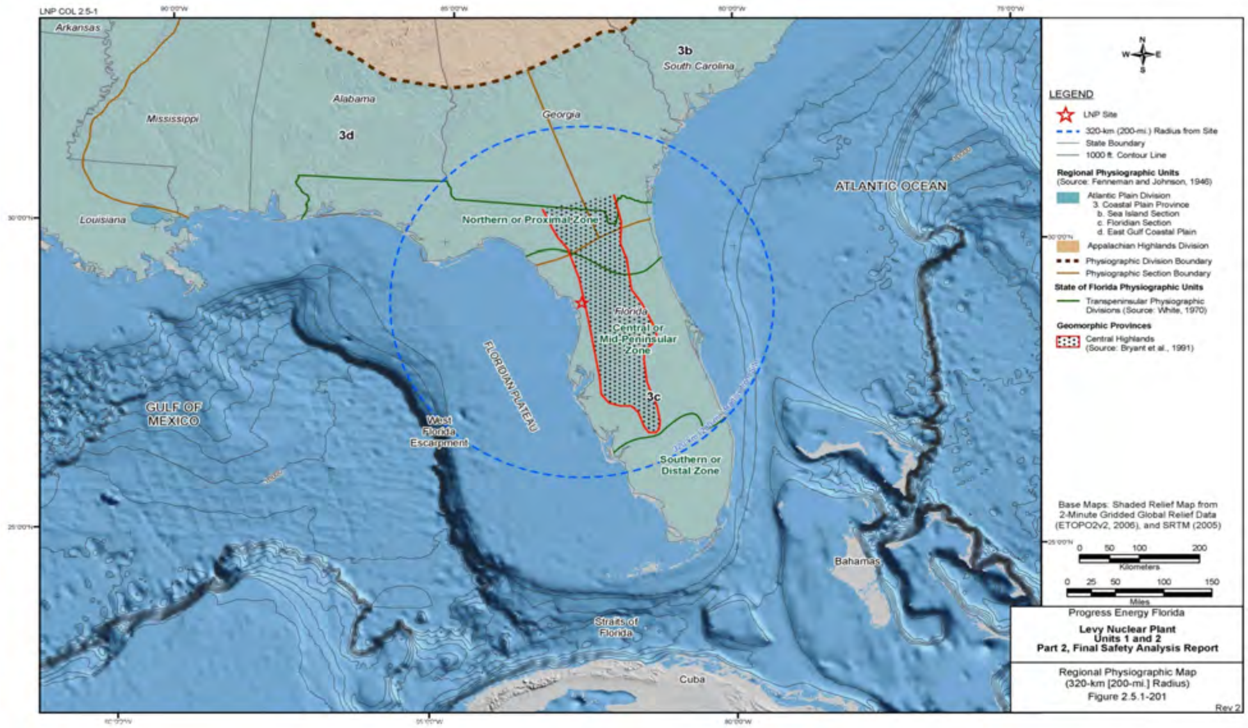


Figure 2.5.1-1. Regional Physiographic Map Showing Location of the LNP Site (FSAR Figure 2.5.1-201)

In FSAR Section 2.5.1.1.1.1.1, the applicant stated that the Sea Island section of the Coastal Plain province (3b in SER Figure 2.5.1-1) is a youthful to mature terraced surface with a slightly submerged margin. In FSAR Section 2.5.1.1.1.1.2, the applicant described the East Gulf section of the Coastal Plain province (3d in SER Figure 2.5.1-1) as a youthful to maturely dissected region, consisting of alternating asymmetric ridges and lowlands with terraces along its outer margin.

In FSAR Section 2.5.1.1.1.1.3, the applicant noted that the Floridian section of the Coastal Plain physiographic province in which the LNP site is located encompasses the entire Florida peninsula (3c in SER Figure 2.5.1-1). The applicant reported that the Floridian section is a recent emergent platform characterized by widespread carbonate rocks with associated karst features. The Floridian section contains the Florida Keys along the southern tip of the Florida peninsula. Three physiographic zones comprise the Florida peninsula, namely the northern (proximal), central (midpeninsular), and southern (distal) zones. The LNP site lies in the midpeninsular zone as shown in SER Figure 2.5.1-1. Discontinuous subparallel ridges, oriented parallel to the length of the Florida peninsula and rising to about 61 meters (m) (200 feet (ft)) above mean sea level (amsl) and separated by broad valleys that may contain shallow lakes, comprise the midpeninsular zone.

2.5.1.2.1.2 Regional Geologic History

FSAR Section 2.5.1.1.2 discusses Late Proterozoic (> 542 million years in age, or Ma), Paleozoic (542 to 251 Ma), Mesozoic (251 to 65.5 Ma), and Cenozoic (65.5 Ma to present) geologic history of the LNP site region.

Late Proterozoic, Paleozoic, and Mesozoic Geologic History

The applicant summarized Late Proterozoic and Paleozoic geologic and tectonic history of the broad region containing the LNP site in FSAR Section 2.5.1.1.2.1. The applicant indicated that breakup of a supercontinental land mass by extensional rifting occurred around Late Proterozoic-Cambrian time (> 488 Ma), and that stratigraphic evidence shows several later compressional events, which culminated in formation of the Appalachian orogen at the end of the Paleozoic (251 Ma).

Regarding Mesozoic geologic and tectonic history, the applicant stated in FSAR Section 2.5.1.1.2.2 that rifting initiated during Triassic and Jurassic time (251 to 145.5 Ma) created the present-day Atlantic Ocean, and that the Gulf of Mexico formed completely by the end of the Jurassic (145.5 Ma). The applicant indicated that, since the end of extensive Triassic and Jurassic rifting, the entire Florida platform has been tectonically quiet based on the occurrence of undisturbed Upper Cretaceous (145.5 to 65.5 Ma) and Tertiary (65.5 to 2.6 Ma) strata on the platform.

Cenozoic Geologic History

In FSAR Section 2.5.1.1.2.3, the applicant stated that, during the first 35 million years of Cenozoic (65.5 Ma to present) time, sea levels were high and carbonate sedimentation dominated deposition on the Florida platform. The applicant noted that encroachment of clastic sediments onto the platform occurred slowly, with these sediments dominating deposition patterns on the platform during late Miocene to Pliocene (11.6 to 5.3 Ma) time. The applicant indicated that periodic regressions of the sea during the Miocene (23 to 5.3 Ma), Pliocene (5.3 to 2.6 Ma), and Quaternary (2.6 Ma to present) exposed vast areas of the carbonate platform, allowing karst features to develop. The applicant also stated that high sea-level stands occurred during the Pleistocene (2.6 Ma to 10,000 years) in southern Florida, and that no evidence exists in the Florida Keys to suggest any significant subsidence, uplift, or tectonic deformation of late Quaternary deposits.

2.5.1.2.1.3 Regional Stratigraphy

FSAR Section 2.5.1.1.3 describes stratigraphic relationships for pre-Cretaceous (> 145.5 Ma), Cretaceous (145.5 to 65.5 Ma), and post-Cretaceous (< 65.5 Ma) rock units, which occur in the LNP site region. The applicant stated that the low relief of the Florida peninsula reflects the nearly horizontal attitude of the predominately Cretaceous and Cenozoic (65.5 Ma to present) carbonate section, which underlies the peninsula and overlies pre-Cretaceous basement rocks of variable age and composition.

2.5.1.2.1.3.1 Pre-Cretaceous Stratigraphy

In FSAR Section 2.5.1.1.3.1, the applicant described the basement rocks which pre-date and underlie the Cretaceous (145.5 to 65.5 Ma) stratigraphic section at depth in the site region. These rocks are primarily Jurassic (201.6 to 145.5 Ma) igneous and volcaniclastic rocks in south Florida; Paleozoic (542 to 251 Ma) igneous and metamorphic rocks in central Florida; relatively undeformed Paleozoic sedimentary rocks in northern Florida; and faulted Paleozoic sedimentary units, which are covered by Triassic (251 to 201.6 Ma) sedimentary rocks, in the Florida panhandle.

2.5.1.2.1.3.2 Cretaceous and Post-Cretaceous Stratigraphy

In FSAR Section 2.5.1.1.3.2, the applicant indicated that Cretaceous (145.5 to 65.5 Ma) and post-Cretaceous (i.e., Cenozoic, 65.5 Ma to present) sedimentary strata of the Coastal Plain unconformably (i.e., representing a gap in the geologic record rather than continuous deposition) overlie pre-Cretaceous (> 145.5 Ma) basement rocks in Florida and adjacent areas of Alabama and Georgia. These strata, deposited in a relatively stable tectonic environment, consist of nearly flat-lying marine units approximately 7 km (4 mi) thick that terminate at the escarpments bounding the Florida platform. This stratigraphic section generally exhibits a west-to-east and north-to-south gradation from clastic to carbonate units.

The applicant reported a striking lithologic contrast between strata of peninsular Florida, which are primarily carbonates, and the predominantly clastic rocks of the Florida panhandle. The middle Eocene (48.6 to 40.4 Ma) Avon Park Formation, the oldest exposed rock unit in Florida, is a carbonate sequence that underlies all of peninsular Florida and forms the foundation unit for the LNP site. The formation exhibits pervasive dolomitization of some stratigraphic horizons (i.e., pure limestone of the Avon Park, made up of calcium carbonate, has been altered to dolomite, calcium magnesium carbonate, by magnesium-bearing waters), and it contains interbedded evaporite deposits (i.e., sedimentary rock units composed mainly of minerals produced from saline solutions as a result of extensive evaporation of the solvent fluid) in its lower part.

2.5.1.2.1.4 Regional Tectonic Setting

FSAR Section 2.5.1.1.4 discusses tectonic setting of the site region. The applicant addressed contemporary tectonic stress; structural setting and geophysical framework as defined by gravity and magnetic data; regional tectonic structures; significant seismic sources at a distance greater than 320 km (200 mi) from the LNP site; and regional seismicity. The applicant specifically assessed major Paleozoic, Mesozoic, and Cenozoic tectonic structures and concluded that none of these regional features are capable tectonic structures.

2.5.1.2.1.4.1 Contemporary Tectonic Stress

In FSAR Section 2.5.1.1.4.1, based on Zoback and Zoback (1989), the applicant indicated that a relatively uniform east-northeast compressive stress field extends regionally from the midcontinent eastward toward the Atlantic continental margin, and that no available data support a distinct Atlantic Coastal Plain stress province. The applicant cited Zoback and Zoback (1980) to suggest that southward-oriented extension along the northern Gulf of Mexico region reflects crustal loading and deformation within the Mississippi River delta complex, rather than effects of the regional east-northeast compressive stress field. The applicant cited Crone and others (1997) to classify the site region as a stable continental region (SCR), and characterized the region as exhibiting low earthquake activity and low stress based on Johnston and others (1994).

2.5.1.2.1.4.2 Regional Structural Setting and Geophysical Framework

In FSAR Section 2.5.1.1.4.2, the applicant stated that continental crust modified by Middle Jurassic (176 to 161 Ma) or later extensional rifting underlies the LNP site at depth. The site lies on the Florida platform near the northeastern margin of the Gulf Coast basin, and the applicant noted that this basin contains sedimentary strata up to 15 km (9 mi) thick, which overlie basement and range in age from Late Triassic (235 to 201.6 Ma) to Holocene (10,000 years to present). Based on Smith and Lord (1997), the applicant indicated that these strata contribute little to regional gravity and magnetic anomalies. The applicant attributed the marked contrast in gravity and magnetic anomalies between southern and northern Florida to a major change in composition of crustal basement from oceanic crust beneath southern Florida to continental crust beneath northern Florida. The applicant commented that this disparity in gravity and magnetic anomalies between northern and southern Florida has been postulated as evidence for a regional basement fault beneath peninsular Florida, which developed during Jurassic (201.6 to 145.5 Ma) time. The applicant noted that Smith and Lord (1997) referred to this basement feature as the Jay fault, or Florida lineament, and interpreted it to represent the northwestern extension of the Bahamas fracture zone across southern Florida.

2.5.1.2.1.4.3 Regional Tectonic Structures

In FSAR Section 2.5.1.1.4.3, the applicant discussed regional tectonic structures within a 320-km (200-mi) radius of the LNP site, including Paleozoic (542 to 251 Ma), Mesozoic (251 to 65.5 Ma), and Cenozoic (65.5 Ma to present) tectonic structures. The following SER sections address these regional tectonic features.

Postulated Basement Faults

In FSAR Section 2.5.1.1.4.3.1, the applicant described two postulated basement structures in the site region. These structures include the faults postulated by Applin and Applin (1965) and Barnett (1975). Based on available data, the applicant concluded that these postulated structures are pre-Mesozoic (> 251 Ma) in age and are not capable tectonic features.

Paleozoic Tectonic Structures

In FSAR Section 2.5.1.1.4.3.2, the applicant described four basement structures postulated in the site region, inferred to be Paleozoic in age (> 251 Ma). These structures include the Peninsular arch, the Suwannee-Wiggins suture, the East Suwannee Basin (North Florida Basin), and the Jay fault. The applicant presented information suggesting that the Peninsular arch, a basement high, is spatially associated with a subparallel high in Upper Cretaceous strata that resulted from upwarping produced by compressional tectonics, possibly intermittently during Cenozoic (65.5 Ma to present) time (Miller, 1986). Based on available data, the applicant concluded that these postulated basement structures are not capable tectonic features.

Mesozoic Tectonic Structures

In FSAR Section 2.5.1.1.4.3.3, the applicant described nine basement structures in the site region, inferred from existing published data to be Mesozoic (251 to 65.5 Ma) in age. These structures include the Bahamas and Sunniland fracture zones, Florida Elbow fault, Apalachicola basin, Middle Ground arch, Sarasota arch, South Florida basin, South Georgia rift, and Tampa basin. The applicant documented a Mesozoic age for these structures, and concluded that they are not capable tectonic features.

Cenozoic Tectonic Structures

In FSAR Section 2.5.1.1.4.3.4, the applicant described Cenozoic (65.5 Ma to present) tectonic structures in the site region. These structures include the Brevard, Ocala, and St. Johns platforms; Gulf trough; Jacksonville and Okeechobee basins; Nassau nose; Osceola low; Sanford high; Sarasota arch; Suwannee strait; and faults postulated by Vernon (1951), Carr and Alverson (1959), Pride and others (1966), Sproul and others (1972), Miller (1986), Hutchinson (1992), and Winston (1996). SER Figure 2.5.1-2, reproduced from FSAR Figure 2.5.1-223, shows the locations of the faults, postulated by numerous authors based on apparent displacements inferred from limited outcrops and widely-spaced subsurface borehole data. The applicant stated that the actual existence of many of these faults is controversial and not well-supported by available data, and concluded that neither the faults nor the other structural features are capable tectonic structures.

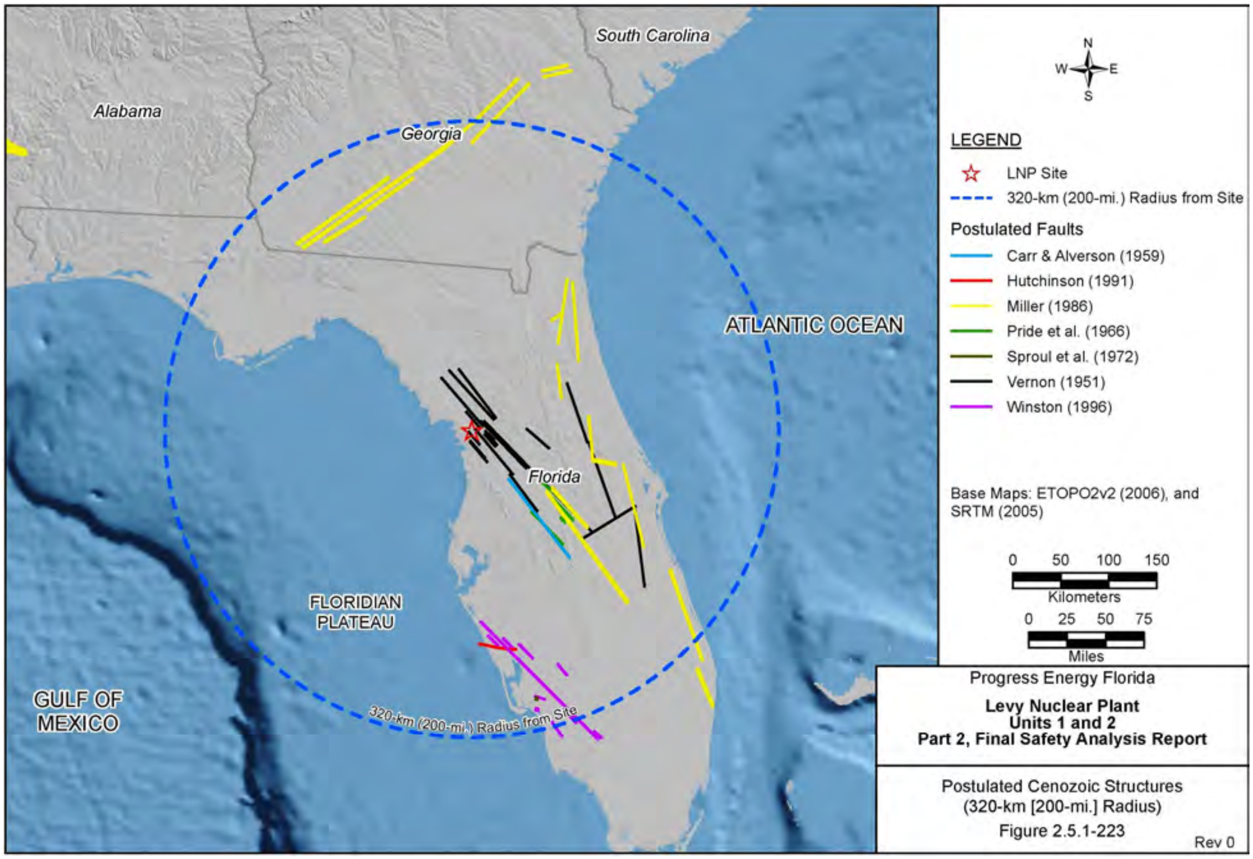


Figure 2.5.1-2. Postulated Cenozoic Tectonic Structures in the LNP Site Region (FSAR Figure 2.5.1-223)

Quaternary Tectonic Structures

In FSAR Section 2.5.1.1.4.3.5, the applicant indicated that there is no geologic or geomorphic evidence of Quaternary faulting in the site region, including the faults postulated by Vernon (1951) to occur within the site area and site vicinity.

2.5.1.2.1.4.4 Significant Seismic Sources at a Distance Greater than 320 km (200 mi)

In FSAR Section 2.5.1.1.4.4, the applicant emphasized the Charleston seismic source zone because, in August 1886, a currently unknown tectonic source in that zone produced one of the largest historical earthquakes in the CEUS in the Charleston, South Carolina area. The applicant incorporated significant new information on source geometry and earthquake recurrence interval for the Charleston earthquake, developed after the initial EPRI studies (EPRI, 1986 and 1989), into an updated Charleston seismic source (UCSS) model that is discussed in detail in FSAR Section 2.5.2. The applicant acknowledged that this model is the same as that used for the Vogtle Electric Generating Plant (VEGP) Early Site Permit (ESP) application (Southern Nuclear Company, 2007), which has been reviewed and approved by NRC staff in NUREG-1923, "Safety Evaluation Report for an Early Site Permit (ESP) at the Vogtle Electric Generating Plant (VEGP) ESP Site." SER Figure 2.5.1-3, reproduced from FSAR Figure 2.5.1-232, illustrates seismicity inside and outside the site region for the time period of 1758 to 2007, including the Charleston region. In addition, the applicant performed sensitivity studies using the CEUS SSC model (NUREG-2115) to address recommendations of the Fukushima Near-Term Task Force described in SECY-12-0025 and evaluate potential seismic hazards at the LNP site in light of these recommendations. SER Section 20.1 presents the staff's evaluation of the sensitivity studies.

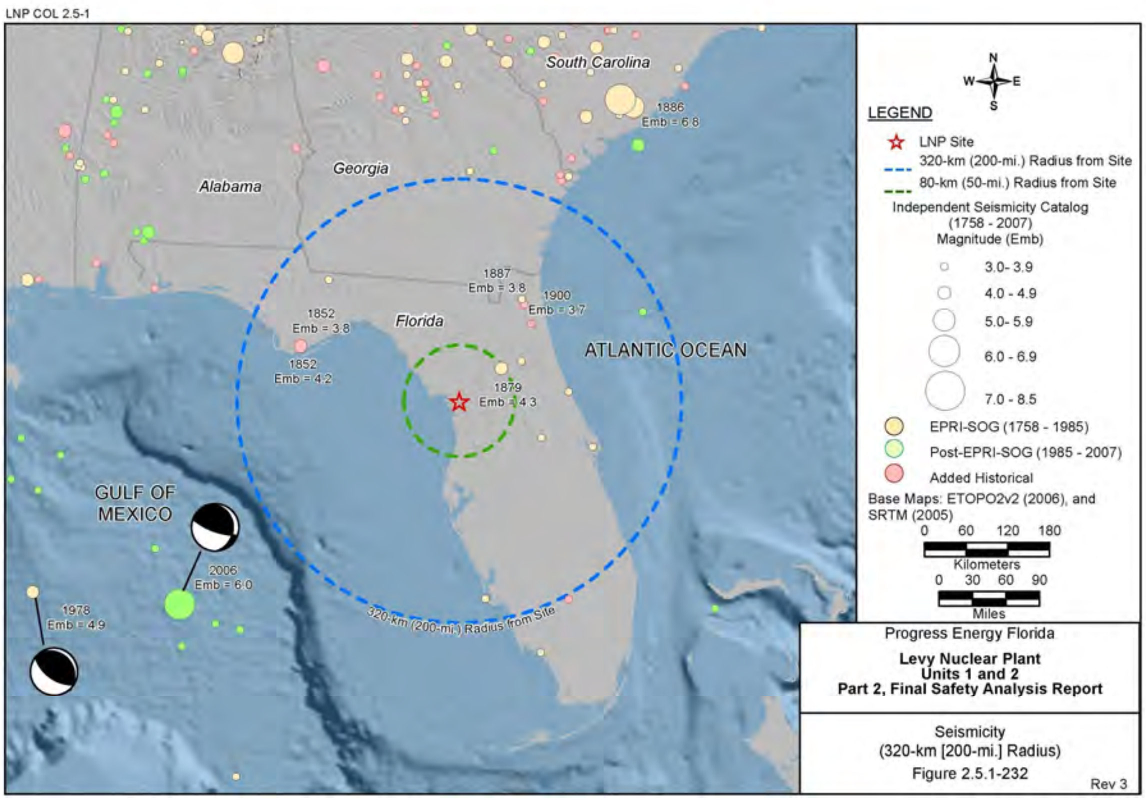


Figure 2.5.1-3. Seismicity in the LNP Site Region and Site Area Between 1758 and 2007. (FSAR Figure 2.5.1–232)

Postulated Associated Tectonic Structures in the Charleston Area

The applicant described five faults postulated to occur in the Charleston area, including the East Coast fault system (ECFS); the Helena Banks fault zone; and the Adams Run, Sawmill Branch, and Summerville faults. The applicant indicated that none of these postulated structures, or any others suggested as occurring in the Charleston area, can be definitively interpreted as a tectonic feature to which the 1886 Charleston earthquake can be related.

Indirect Evidence Related to the Charleston Seismic Source

The applicant discussed the relationship between large global intraplate earthquakes and tectonic environments; liquefaction features produced by the 1886 event and prehistoric earthquakes in the Charleston region; intensity data from the 1886 Charleston earthquake; and instrumental seismicity.

Based on Johnston and others (1994), the applicant documented that the Charleston meizoseismal area (i.e., the area in which an earthquake is most strongly felt) occurs within the region of Mesozoic (251 to 65.5 Ma) or younger extended crust along the southeastern margin of the North American craton, a tectonic environment characterized by large-magnitude earthquakes on a global scale. The applicant also documented that the distribution of liquefaction features produced both by the 1886 Charleston earthquake and pre-1886 events suggest that the Charleston meizoseismal area may encompass the seismic source for 1886 and the pre-1886 events. Intensity data for the 1886 Charleston earthquake also indicate a meizoseismal area centered on Charleston. The applicant further indicated that elevated instrumental seismicity occurs in the Middleton Place-Summer seismic zone, which is located about 20 km (13 mi) northwest of Charleston in the Charleston meizoseismal area. Based on these lines of evidence, the applicant stated that information published since the results of the original EPRI study (EPRI, 1986) strongly indicate that the Charleston seismic source is localized in the meizoseismal area of the 1886 Charleston earthquake, or in the region of coastal South Carolina as constrained by paleoliquefaction data.

M_{max} and Recurrence Interval for the Charleston Seismic Source

In regard to maximum moment magnitude (M_{max}) for the Charleston seismic source, the applicant stated that, given the large uncertainties in working with paleoliquefaction data and the methods for estimating magnitudes from these data, the best representation of M_{max} for the Charleston seismic source should be based on the maximum magnitude of the 1886 earthquake. The applicant reviewed data generated since the original EPRI study (EPRI, 1986), and concluded that M_{max} for the 1886 Charleston earthquake ranges between 6.75 and 7.5.

Concerning recurrence interval for the Charleston seismic source, based on Talwani and Schaeffer (2001), the applicant noted that studies of paleoliquefaction features conducted since the original EPRI study (EPRI, 1986) suggest a recurrence interval for large earthquakes generated by that source of 500-600 years. The applicant incorporated this updated information into the UCSS model as discussed in detail in FSAR Section 2.5.2.

2.5.1.2.1.4.5 Regional Seismicity

In FSAR Section 2.5.1.1.4.5, the applicant indicated that infrequent and low seismicity characterize the U.S. Gulf Coast region in which the LNP site lies (see SER Figure 2.5.1-3). The applicant stated that only 15 earthquakes larger than a body-wave magnitude (m_b) 3.0 have occurred within the LNP site region. The largest event, an 1879 m_b 4.3 earthquake located about 77 km (48 mi) northeast of the LNP site, is the only event within 80 km (50 mi) of the site.

The applicant acknowledged an m_b 6.0 earthquake outside the site region in the Gulf of Mexico, which occurred on 10 September 2006. The focal plane mechanism for that earthquake indicated a compressive stress regime of tectonic origin. On 10 February 2006, an m_b 4.9 event, interpreted to be related to gravity-driven displacement along a growth fault, also occurred outside the site region along the Gulf Coast. The applicant recognized that these two earthquakes may have implications for evaluation of seismicity at the LNP site, and discussed the events in detail in FSAR Section 2.5.2.

2.5.1.2.2 Site Geology

FSAR Section 2.5.1.2 discusses physiography and topography, geomorphology, geologic history, stratigraphy, structural geology, geology, geologic hazard, and engineering geology within the 40 and 8 km (25 and 5 mi) site vicinity and area, respectively. In some of these discussions, the applicant also evaluated the area within the 1 km (0.6 mi) site location. The applicant specifically addressed features commonly developed in karst terrains (e.g., sinkholes) because the LNP site lies within the Limestone Shelf and Hammocks subzone of the Gulf Coastal Lowlands, a geomorphic province underlain by limestones of Eocene age (55.8 to 33.9 Ma), including the Avon Park Formation, which have been subjected to dissolution. The following sections summarize the information the applicant provided in FSAR Section 2.5.1.2.

2.5.1.2.2.1 Site Physiography, Topography, and Geomorphology

FSAR Section 2.5.1.2.1 discusses physiography, topography, and geomorphic provinces within the site vicinity and site area in relation to development of marine terraces and karst terrain, both of which characteristically occur in the site region. The applicant stated that the LNP site lies within the Gulf Coastal Lowlands geomorphic province of the midpeninsular physiographic zone, and that this geomorphic province represents a mature karst terrain overlain by a thin veneer of marine terrace deposits. The other geomorphic province comprising the midpeninsular physiographic zone, the Central Highlands, occurs within the site vicinity as illustrated in SER Figure 2.5.1-4, reproduced from FSAR Figure 2.5.1-234.

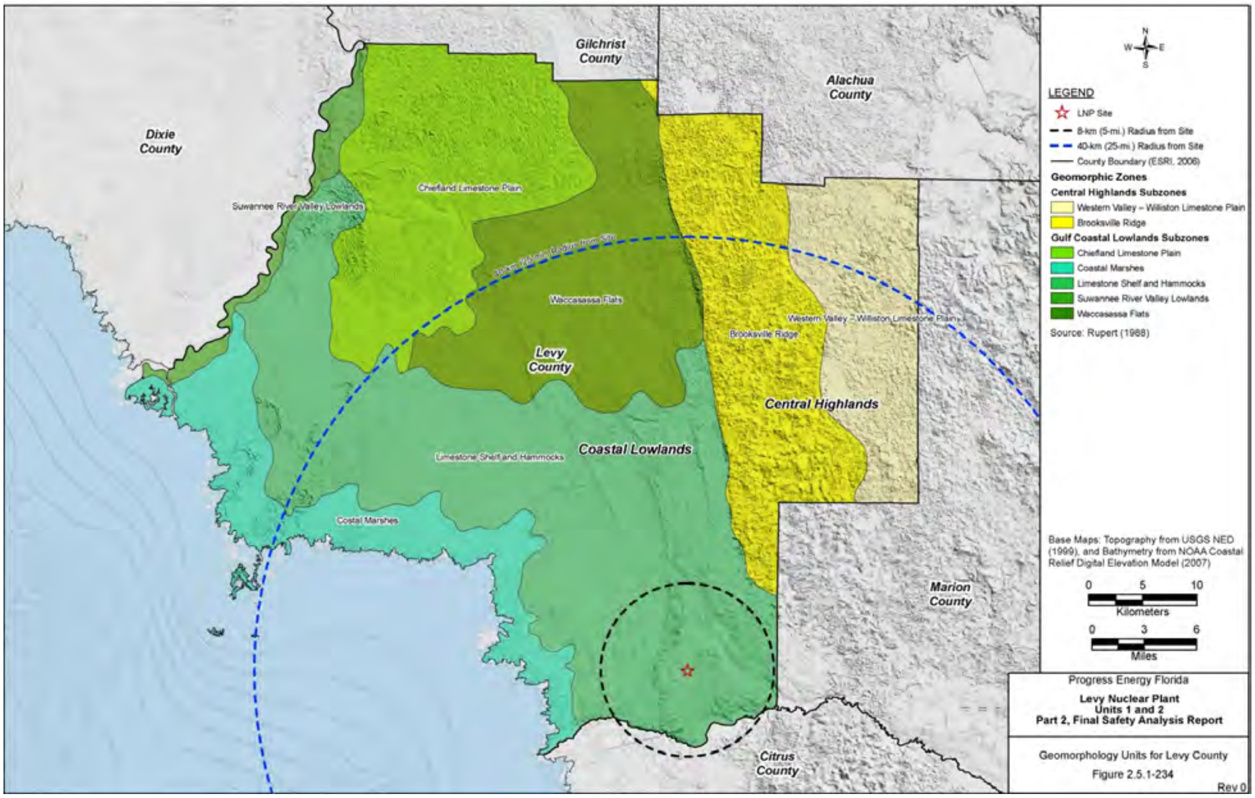


Figure 2.5.1-4. Geomorphic Divisions of Levy County
(FSAR Figure 2.5.1-234)

The applicant noted that the Central Highlands geomorphic province includes a series of highlands and ridges separated by valleys, all of which generally parallel the coastline of the central Florida peninsula. The highlands and ridges, interpreted to be relict coastal features, range in elevation from about 23 to 64 m (75 to 210 ft) amsl. The applicant indicated that the LNP site lies specifically in the Limestone Shelf and Hammocks subzone of the Gulf Coastal Lowlands province (see SER Figure 2.5.1-4), and that this subzone exhibits a highly karstic, irregular, dissolutioned erosional surface composed of Eocene (54.8 to 33.7 Ma) limestones. The karstic limestone units are overlain by sand dunes, ridges, and belts of coastline-parallel paleoshoreline sands associated with the Pamlico marine terrace of Pleistocene (2.6 Ma to 10,000 years) age. The applicant stated that the five marine terraces present in the site vicinity record the long-term effects of late Tertiary (5.3 to 2.6 Ma) to Quaternary (2.6 Ma to present) sea level changes on the stable Florida platform.

2.5.1.2.2.2 Site Vicinity Geologic History

FSAR Section 2.5.1.2.2 summarizes the geologic history of the site vicinity. The applicant indicated that the Florida platform has been tectonically quiescent since Cretaceous (145.5-65.5 Ma) time, allowing a thick sequence of shallow-water marine carbonate rocks to be deposited in the site vicinity, with periodic pulses of clastic sediments interrupting the carbonate deposition. The applicant stated that carbonate deposition ceased on the platform by Middle to Late Pliocene (i.e., between about 3.6-2.6 Ma), due to an influx of clastic sediments, and that total accumulated thickness of sedimentary units in the site vicinity is approximately 1,320 m (4331 ft) based on borehole data. The applicant noted that sea level fluctuations from Miocene (23-5.3 Ma) into Quaternary (2.6 Ma to present) influenced deposition and distribution of sediments on the Florida platform in the site vicinity, and sea level rose to its present-day level following the latest sea level regression during the Pleistocene (2.6 Ma to 10,000 years).

2.5.1.2.2.3 Site Vicinity and Site Area Stratigraphy

FSAR Section 2.5.1.2.3 addresses stratigraphy of the site vicinity and site area. The applicant stated that, within the site vicinity and site area, undifferentiated sediments consisting of surficial sands, clayey sands, and alluvium of Pleistocene (2.6 Ma to 10,000 years) to Holocene (10,000 years to present) age overlie a thick section of Cretaceous (144.5 to 65.5 Ma) and Cenozoic (65.5 Ma to present) carbonates (i.e., limestone and dolomite). The applicant indicated that the Cenozoic carbonate section lies atop basement rocks of Triassic (251 to 201.6 Ma) and Paleozoic (> 251 Ma) age.

The applicant noted that the undifferentiated surficial sediments of Pleistocene to Holocene age are commonly thickest in areas where they accumulated as infilling of karst features. The applicant stated that the surficial sediments mapped at the LNP site generally have a thickness of about 1 to 2 m (3.2 to 6.5 ft). The applicant also noted that sinkholes and related karst features associated with dissolution of the underlying limestone bedrock are common in the site vicinity.

The applicant further indicated that the Avon Park Formation, the foundation unit at the LNP site and the oldest exposed rock unit in Florida, is part of the Cenozoic carbonate section and Middle Eocene (48.6 to 40.4 Ma) in age. The applicant stated that the Avon Park Formation is approximately 243 to 304 m (800 to 1,000 ft) thick in Levy County.

2.5.1.2.2.4 Site Vicinity and Site Area Structural Geology

FSAR Section 2.5.1.2.4 discusses structural geology of the site vicinity and site area. The applicant stated that recent geologic maps encompassing the site vicinity show only a single potential structural feature, the Ocala platform, and no faults. The long axis of the Ocala platform, located about 14 km (8.7 mi) northeast of the LNP site at its nearest point, trends northwest-southeast across midpeninsular Florida. Based on personal communications with regional experts (T. Scott, 2009, and S. Upchurch, 2009), the applicant indicated that the Ocala platform likely resulted from sedimentary, rather than tectonic, processes. The applicant noted that a primary northwest-southeast fracture set parallels the axis of the Ocala platform, while a secondary northeast-southwest fracture set exhibits a strike, which parallels the approximate down dip direction of the flanks of the platform. The applicant recognized that regional fracture systems control stream drainage patterns and sinkhole alignments.

The applicant stated that no known faults occur at the site location based on current field evidence. However, the applicant noted that Vernon (1951) postulated seven northwest-trending faults along the Levy-Citrus County boundary, five of which lie within the LNP site vicinity. The five faults postulated by Vernon (1951) to occur in the site vicinity are as follows:

- Bronson graben – located 24 km (15 mi) northeast of the site.
- Inverness fault – located east of the site within the site area.
- Long Pond fault – located 10 km (6 mi) northeast of the site.
- Unnamed faults “A” and “B” – located 4 km (2.5 mi) southwest and 7 km (4 mi) northeast of the site, respectively.

The applicant documented that subsequent geologic investigations provided no evidence to support the existence of any of the faults proposed by Vernon (1951), and concluded that none of these postulated structures are capable tectonic sources. The applicant also reported two small domal structures, the Homosassa Springs dome located 25 km (15.5 mi) south of the site and the West Levy dome located 45 km (28 mi) northwest of the site. The applicant concluded that these two domal structures pose no geologic hazard for the LNP site because no field evidence exists to indicate that they are tectonically active features.

2.5.1.2.2.5 Site Location Geology

FSAR Section 2.5.1.2.5 discusses geology of the site location, including location-specific geomorphology, stratigraphy, and karst development, based on information derived from field reconnaissance and subsurface exploration. In FSAR Section 2.5.1.2.5.1, the applicant stated that surface morphology is characterized by shallow depressions less than 1 to 2 m (2 to 6 ft) deep above sinkholes or paleosinks, which vary from well-defined, small circular depressions less than 50 m (164 ft) in diameter in the eastern half of the site location to large, irregular depressions up to 600 m (2000 ft) wide in the western half. By analogy with similar morphology of the present-day coastline south of the site in Citrus County, the applicant concluded that this surface morphology indicates older marine terrace surfaces, which have been karstified due to dissolution of carbonate rocks, underlie the site. A thin veneer of Quaternary (2.6 Ma to present) sediments mantle the terrace surfaces.

In FSAR Section 2.5.1.2.5.2, based on results of the geotechnical drilling program conducted at the LNP site to investigate subsurface stratigraphy, the applicant indicated that the Middle Eocene (48.6 to 40.4 Ma) Avon Park Formation is the marine carbonate unit encountered immediately below surficial sedimentary aquifer deposits. The applicant noted that the thickness of Quaternary sediments varied across the site, generally from less than 3 m (10 ft) to about 30 m (100 ft), with an approximate thickness of 2 m (6 ft) beneath the proposed location of the nuclear island and a maximum measured thickness of 73.5 m (241 ft) at one borehole located just beyond the perimeter of the LNP Unit 2 site. The applicant stated that the Avon Park Formation occurs as a soft fossiliferous limestone near the top of the sequence, with increasing dolomitization at depth, particularly in a zone of denser rock at depths around 40 to 60 m (140 to 190 ft). The applicant noted that the Avon Park Formation was softer, and consequently exhibited poorer core recovery, at depths below about 61 m (200 ft).

In FSAR Section 2.5.1.2.5.3, the applicant evaluated the potential for karst development at the site location. The applicant stated that the rectilinear margins of topographic lows, the orientations of depression axes, and the spatial distribution of deeper circular surficial dissolution features suggest control by joint systems in the underlying rock units, including the Avon Park Formation. However, the applicant indicated that the carbonate units in the Avon Park Formation typically exhibit greater degrees of dolomitization than younger limestone units in the site vicinity, and would, therefore, be less susceptible to dissolution and development of karst. The applicant concluded that surface morphology and stratigraphy at the site location are consistent with the anticipated characteristics of a paleokarst landscape mantled by a veneer of Quaternary (2.6 Ma to present) sands. The applicant cross-referenced FSAR Section 2.5.4.1.2.1, and stated that subsurface karst features identified in borings under proposed safety-related structures at the LNP site varied in lateral extent from a few centimeters to about 1.5 m (5 ft) when associated with dissolution controlled by vertical fractures, and from a few centimeters to approximately 3 m (10 ft) in lateral extent when associated with dissolution controlled by horizontal bedding planes.

2.5.1.2.2.6 Site Area Geologic Hazard Evaluation

FSAR Section 2.5.1.2.6 presents an evaluation of potential geologic hazards at the LNP site based on the applicant’s review of published information, reconnaissance investigations performed in the site area, discussion with karst experts, and site characterization results. The applicant concluded the following in regard to potential geologic hazards in the site area:

- The site lies in an area of low seismicity and there are no capable tectonic sources in the site area. Therefore, the potential for surface tectonic deformation at the site is minor.
- No natural processes that could cause tectonic uplift are active at the site.
- Unrelieved residual stresses do not pose a hazard to the site.
- Ground failure and differential settlement due to liquefaction do not pose hazards to the site. (The applicant discussed this potential hazard in detail in FSAR Section 2.5.4.)
- Potential surface deformation due to carbonate dissolution and collapse or subsidence related to karst development is the only geologic hazard identified in the LNP site area.

2.5.1.2.2.7 Site Engineering Geology Evaluation

FSAR Section 2.5.1.2.7 addresses the potential engineering significance of geologic and geotechnical features and materials at the site, including zones of alteration, weathering, weakness due to the presence of faults or fault zones, karst, and deformation. In FSAR Section 2.5.1.2.7.1, the applicant cross-referenced FSAR Section 2.5.4 and stated that it addressed engineering behavior of soil and rock materials. In FSAR Section 2.5.1.2.7.2, the applicant indicated that the Avon Park Formation, the bedrock unit underlying the LNP site, has been altered by weathering and dissolution, but no zones of weakness related to faults or fault zones have been identified at the site. Furthermore, the applicant stated that recent studies do not provide evidence of faults postulated by Vernon (1951) to occur in the site vicinity. The applicant acknowledged that smaller-scale fractures and joints parallel to regional fracture trends occur in bedrock outcrops in the site area and in boreholes at the LNP site, and that these discontinuities, particularly in combination with bedding planes along which dissolution may also occur, are key elements controlling the development of karst.

In FSAR Section 2.5.1.2.7.3, the applicant explained that karst features, which occur within the LNP site location, are expected to be associated with vertical fractures and horizontal bedding planes, and that karst-related dissolution and infilled zones, which may exist in the subsurface beneath the LNP foundation, would be addressed through appropriate design considerations as discussed in FSAR Section 2.5.4. In FSAR Section 2.5.1.2.7.4, the applicant stated that, with the exception of possible paleosinkholes, no deformation zones were encountered during site exploration studies for LNP Units 1 and 2, and that excavation mapping would be done during construction to further evaluate the possible existence of deformation zones at the site. Groundwater conditions at the site are discussed in FSAR Sections 2.4 and 2.5.4.6.

2.5.1.3 *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," and its supplements.

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 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, "Geologic and seismic siting criteria," for evaluating the suitability of a proposed site based on consideration of the geologic, geotechnical, geophysical, and seismic characteristics of the proposed site. Geologic and seismic siting factors must include the safe shutdown earthquake (SSE) for the site and the potential for surface tectonic and non-tectonic deformation. The site-specific GMRS satisfies requirements of 10 CFR 100.23 with respect to development of the SSE.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for basic geologic and seismic information are given in Section 2.5.1 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants."

- Regional Geology: In meeting the requirements of 10 CFR 52.79(a)(1)(iii) and 10 CFR 100.23, LNP COL FSAR Section 2.5.1.1 will be considered acceptable if a complete and documented discussion is presented for all geologic (including tectonic and non-tectonic), geotechnical, seismic, and geophysical characteristics, as well as conditions caused by human activities, deemed important for safe siting and design of the plant.

- Site Geology: In meeting the requirements of 10 CFR 52.79(a)(1)(iii) and 10 CFR 100.23, and the guidance in RG 1.132, "Site Investigations for Foundations of Nuclear Power Plants"; Revision 2; RG 1.138, "Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants"; Revision 2; RG 1.198, "Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites" RG 1.206; and RG 1.208, LNP COL FSAR Section 2.5.1.2 will be considered acceptable if it includes a description and evaluation of geologic (including tectonic and non-tectonic) features, geotechnical characteristics, seismic conditions, and conditions caused by human activities at appropriate levels of detail within areas defined by circles drawn around the site using radii of 40 km (25 mi) for site vicinity, 8 km (5mi) for site area, and 1 km (0.6 mi) for site location.

In addition, the geologic characteristics should be consistent with appropriate sections from RG 1.132, Revision 2; RG 1.138, Revision 2; RG 1.198; RG 1.206; and RG 1.208.

2.5.1.4 *Technical Evaluation*

The NRC staff reviewed Section 2.5.1 of the LNP 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 basic geologic and seismic characteristics. 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 LNP COL application.

The staff reviewed the following information in the LNP COL FSAR.

AP1000 COL Information Item

- LNP COL 2.5-1

The NRC staff reviewed LNP COL 2.5-1 regarding the geologic, seismic, and geophysical information included in Section 2.5.1 of the LNP COL FSAR. The COL information item in Section 2.5.1 of the AP1000 DCD 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) geological history,

(4) evidence of paleoseismicity, (5) site stratigraphy and lithology, (6) engineering significance of geological 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.

Based on the discussion of the basic geologic and seismic information presented in LNP COL FSAR Section 2.5.1, the staff concludes that the applicant provided the information required to satisfy LNP COL 2.5-1.

The technical information presented in LNP COL FSAR Section 2.5.1 resulted from the applicant's review of existing geologic and seismicity data and published literature cited by the applicant; discussions with individuals who have conducted recent research in and around the site area; field reconnaissance studies in the site vicinity and site area and at the site location; lineament analyses using aerial photographs and remote sensing imagery; and detailed investigations performed for the LNP site, including subsurface borings, surface geophysical testing, and downhole geophysical logging and seismic testing. The applicant also provided limited information applicable to the LNP site as derived from the FSAR prepared by Florida Power Corporation (Florida Power Corporation, 1976) for the CR3, which is located about 18 km (11 mi) southwest of the LNP COL site. Through the review of LNP COL FSAR Section 2.5.1, 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.

LNP COL FSAR Section 2.5.1 includes geologic and seismic information the applicant collected in support of the vibratory ground motion analysis and the site-specific GMRS provided in 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 the review on geologic and seismic data published since the mid-to-late 1980s to assess whether these data indicate a need to update the existing seismic source models.

The staff visited the site in April 2009 (ML092600064), supported by technical experts from the U.S. Geological Survey (USGS), and interacted with the applicant and its consultants in regard to the geologic, seismic, geophysical, and geotechnical investigations being conducted for the LNP COL application. During this site visit, the staff examined core samples from the initial site characterization boreholes placed at the locations of containment structures and turbine buildings for LNP Units 1 and 2, as well as exposures of the Avon Park Formation along the Waccasassa River about 25 km (16 mi) northwest of the site. The core samples allowed staff to examine subsurface stratigraphy at the site, and the outcrops along the river permitted staff to observe and measure spacing and orientation of fractures in the Avon Park Formation. The staff also visited the site in September 2009 to examine core samples from the test grouting program. The staff noted grout uptake in a single vertical fracture intersected by one of the grout boreholes. Also during the September 2009 site audit, the staff examined exposures of the Avon Park Formation at the abandoned Gulf Hammock quarry about 19 km (12 mi) north-northwest of the LNP site, which again permitted staff to observe and measure spacing and orientation of fractures in the Avon Park Formation. In addition, in February 2010 at the applicant's records facility in Virginia, the staff examined boring logs, core photographs, and written core descriptions for 6 additional boreholes, located to be offset approximately 1.5 m (5 ft) from the position of the initial site characterization boreholes. These "offset" boreholes were drilled using controlled coring techniques to improve core recovery and further characterize soft zones postulated to mark horizons of low recovery in the initial site characterization boreholes for LNP Units 1 and 2. The two site visits and the examination of boring logs, core photographs, and core descriptions enabled the staff to assess and confirm the interpretations, assumptions, and conclusions the applicant made regarding the basic geologic and seismic information for the LNP site, including features related to karst development.

The following SER Sections 2.5.1.4.1, "Regional Geology," and 2.5.1.4.2, "Site Geology," present the staff's evaluation of the information the applicant provided in LNP COL FSAR Section 2.5.1 and the applicant's responses to RAIs for that FSAR section. In addition to the RAIs addressing specific technical issues related to regional and site geology of the LNP site, discussed in detail below, the staff also prepared several editorial RAIs to further clarify certain descriptive statements the applicant made in the FSAR and to qualify geologic features illustrated in FSAR figures. These editorial RAIs are not discussed in this technical evaluation. Also, RAIs related to geologic issues resolved in FSARs previously prepared for other sites in the CEUS are not discussed in detail in this technical evaluation for the LNP site, but rather addressed by cross-reference to and a summary of the pertinent information used to satisfactorily resolve the issues as presented in those FSARs.

2.5.1.4.1 Regional Geology

The staff focused the review of LNP COL FSAR Section 2.5.1.1 on the descriptions the applicant provided for physiography, topography, geologic history, stratigraphy, tectonic setting, and seismicity within the 320-km (200-mi) radius LNP site region. The staff also focused on the description of significant seismic sources outside the site region the applicant provided under the discussion of regional tectonic setting.

2.5.1.4.1.1 Regional Physiography and Topography

In FSAR Section 2.5.1.1.1, the applicant described the physiography and topography of the Coastal Plain physiographic province in the site region, including the Sea Island, East Gulf, and Floridian sections of that physiographic province. SER Figure 2.5.1-1 shows the location of the LNP site and its spatial relationship to these three sections of the Coastal Plain physiographic province. The LNP site lies within the Floridian section of the Coastal Plain province.

The staff focused the review of FSAR Section 2.5.1.1.1 on the applicant's discussion of the characteristics of rock units within the Coastal Plain physiographic province and the mechanism for and timing of the differential emergence of the Floridian Coastal Plain section of the Coastal Plain physiographic province in which the site lies. In RAI 2.5.1-13, the staff asked the applicant to clarify the use of the adjective "weak" when describing the limestones contained in the East Gulf Coastal Plain section of the site region. In response to RAI 2.5.1-13, the applicant stated that “weak” refers to these limestones being less resistant to erosion, without any implication related to mechanical strength of the rock unit, while “stronger” indicates a rock unit that is more resistant to erosion (e.g., sandstones). The applicant incorporated changes in LNP COL FSAR Section 2.5.1.1.1.1.2 to replace the adjective “weak” with the phrase “more easily eroded” when referring to limestones and shales, and “less easily eroded” when discussing sandstones.

Based on review of the response to RAI 2.5.1-13 and LNP COL FSAR Section 2.5.1.1.1.1.2, the staff concludes that the applicant adequately clarified the descriptive term “weak” as applied to the limestone units, which occur in the East Gulf Coastal Plain section of the site region. The staff makes this conclusion because the applicant clearly explained that “weak” refers to limestone and shale units that are less resistant to erosion due to its physical properties, rather than to any mechanical weakness that could pose a potential problem for stability of the foundation rock units at the LNP site. Consequently, the staff considers RAI 2.5.1-13 to be resolved.

In RAI 2.5.1-14, the staff asked the applicant to discuss the mechanism for and timing of the differential emergence of the Floridian Coastal Plain section of the Coastal Plain physiographic province in which the site lies in order to document that this emergence is not the result of Cenozoic (65.5 Ma to present) tectonic deformation. In response to RAI 2.5.1-14, the applicant summarized information from published literature cited by the applicant documenting that the observed elevation differences are the result of depositional and erosional processes primarily associated with sea level fluctuations, and that no evidence exists to suggest Cenozoic tectonic deformation as the causative mechanism. Based on robust data presented by Willett (2006), the applicant also documented calculations that show karst areas in Florida are losing about 1 m (3 ft) of limestone every 160,000 years due to dissolution, resulting in isostatic uplift of the Florida carbonate platform of as much as 58 m (190 ft) since early Quaternary time (i.e., < 2.6 Ma). The applicant further reported that Means (2009) suggested lithospheric flexure due to sediment loading as another non-tectonic uplift mechanism.

Based on review of the response to RAI 2.5.1-14, and independent review of published geologic information cited by the applicant, the staff concludes that the applicant documented that non-tectonic processes related to erosion, isostatic adjustment, and sea level fluctuations produced the differential emergence of the Floridian Coastal Plain section of the Coastal Plain physiographic province in which the site lies. Based on information derived from Willett (2006) and Means (2009), the staff further concludes that there is no evidence for Cenozoic tectonic deformation in the site area, and that the likelihood of neotectonic (i.e., < 5.3 Ma in age) deformation in the site region is negligible. The staff draws these conclusions because a preponderance of data collected by experts on geologic evolution of the site region strongly supports non-tectonic processes as the causative mechanism for emergence of the Florida Coastal Plain section. Consequently, the staff considers RAI 2.5.1-14 to be resolved.

Based on the review of LNP COL FSAR Section 2.5.1.1.1 and the responses to RAIs 2.5.1-13 and 2.5.1-14, the staff finds that the applicant provided a thorough and accurate description of regional physiography and topography in support of the LNP COL application.

2.5.1.4.1.2 Regional Geologic History

In FSAR Sections 2.5.1.1.2.1 through 2.5.1.1.2.3, the applicant discussed Late Proterozoic (> 542 Ma), Paleozoic (542 to 251 Ma), Mesozoic (251 to 65.5 Ma), and Cenozoic (65.5 Ma to present) geologic history of the LNP site region, including the Florida platform on which the site is located, concentrating on tectonic evolution and depositional history of sedimentary rock units for the site region and the platform. The applicant documented that tectonic deformation in the site region occurred mainly in pre-Cretaceous (> 65.5 Ma) time; that the Florida platform represents long-term sedimentation in a tectonically stable area as evidenced by undisturbed Upper Cretaceous (99.6 to 65.5 Ma) and Tertiary (65.5 to 2.6 Ma) strata on the platform; and that late Quaternary (< 2.6 Ma to present) deposits in the Florida Keys do not record significant uplift, subsidence, or tectonic deformation of the platform.

The staff focused the review of FSAR Section 2.5.1.1.2 on the Cenozoic depositional history of the Florida platform to ensure that no sedimentation patterns reflected Quaternary tectonic deformation in the site region. Based on independent review of the data sources the applicant provided, the staff concludes that tectonic deformation in the site region occurred mainly in pre-Cretaceous time because no existing data indicate younger Cenozoic (65.5 Ma to present) tectonic deformation. The staff further concludes that the Florida platform represents long-term sedimentation in a tectonically stable area since undisturbed Upper Cretaceous (99.6 to 65.5 Ma) and Tertiary (65.5 to 2.6 Ma) strata occur on the platform and no evidence exists for late Quaternary deformation.

Based on review of the LNP COL FSAR Section 2.5.1.1.2, the staff finds that the applicant provided a thorough and accurate description of the regional geologic history in support of the LNP COL application.

2.5.1.4.1.3 Regional Stratigraphy

In FSAR Section 2.5.1.1.3, the applicant described stratigraphic relationships for pre-Cretaceous (> 145.5 Ma), Cretaceous (145.5 to 65.5 Ma), and post-Cretaceous (< 65.5 Ma) rock units, which occur in the LNP site region. The applicant specifically addressed the foundation unit for LNP Units 1 and 2, the Middle Eocene (48.6 to 40.4 Ma) Avon Park Formation.

The staff focused the review of FSAR Section 2.5.1.1.3 on the applicant's descriptions of the Avon Park Formation. In RAI 2.5.1-4, the staff asked the applicant to describe the composition, thickness, lateral distribution, and material properties of a "shelf" occurring within the Avon Park Formation, as defined by low shear wave velocity (V_S) values. In response to RAI 2.5.1-4, the applicant stated that the "shelf" is a dolomitized stratigraphic horizon within the Avon Park Formation. The applicant indicated that this horizon exhibits little to no dip, and appears to underlie and extend laterally beyond the footprint of LNP Units 1 and 2. The applicant provided figures locating the dolomitized "shelf" horizon in relation to LNP Units 1 and 2, as well as tables summarizing the physical properties of this dolomitized horizon.

Based on its review of FSAR Section 2.5.1.1.3 and the applicant's response to RAI 2.5.1-14, the staff concludes that the applicant adequately described the stratigraphic "shelf" horizon within the Avon Park Formation, which underlies LNP Units 1 and 2. The staff makes this conclusion because the information provided by the applicant characterized this stratigraphic horizon in regard to its composition, thickness, lateral extent, material properties, and engineering parameters. Consequently, the staff considers RAI 2.5.1-4 to be resolved.

Based on review of LNP COL FSAR Section 2.5.1.1.3 and the applicant's response to RAI 2.5.1-4, the staff finds that the applicant provided a thorough and accurate description of the regional stratigraphy in support of the LNP COL application.

2.5.1.4.1.4 Regional Tectonic Setting

FSAR Section 2.5.1.1.4 discusses the tectonic setting of the site region. The applicant described the regional tectonic setting in terms of contemporary tectonic stress; structural setting and geophysical framework; tectonic features within a 320-km (200-mi) radius of the site; and significant seismic sources at a distance greater than 320 km (200 mi) from the LNP site. The staff focused the review of LNP COL FSAR Section 2.5.1.1.4 on the discussion of postulated tectonic features in the site region and possible significant seismic sources outside the site region, including the Charleston seismic source zone.

2.5.1.4.1.4.1 Tectonic Features in the Site Region

In RAI 2.5.1-17, the staff asked the applicant to discuss the data used by Barnett (1975) that postulated a basement fault passing through or near the site location, as suggested by FSAR Figure 2.5.1-222. In response to RAI 2.5.1-17, the applicant stated that Barnett (1975) did not provide detailed descriptions or justification for the locations of most of the basement faults he postulated, including the fault shown on FSAR Figure 2.5.1-222, which he inferred displaced pre-Middle Jurassic (> 161 Ma) basement rocks in the LNP site area. The applicant noted that, due to the scale of the maps presented by Barnett (1975), it was not possible to determine the exact location of the postulated basement structure relative to the site. The applicant indicated that Barnett (1975) based his interpretations of basement faulting on data from about eighty widely-spaced and sparsely-distributed wells that penetrated the basement, as well as well logs and geophysical and geologic data derived from published literature sources cited by the applicant. The applicant stated that the data cited by Barnett (1975) do not require a significant offset in the top of basement, as would be expected if a normal fault of large displacement existed, and that the structures postulated by Barnett (1975) are not expressed in gravity or magnetic maps for the site vicinity. Based on the fact the no data show anomalies to suggest faulting in the LNP site vicinity, the applicant concluded that no definitive evidence exists for faulting there.

Based on the fact that no current data suggest the presence of post-middle Jurassic faulting in the site vicinity, the staff concludes that the applicant provided sufficient information in the response to RAI 2.5.1-17 to document the speculative nature of the basement faults postulated by Barnett (1975), and that, if these basement structures exist, there is no evidence to demonstrate post-Middle Jurassic activity associated with the structures in the site vicinity. Consequently, the staff considers RAI 2.5.1-17 to be resolved.

In RAI 2.5.1-18, the staff asked the applicant to locate all regional tectonic structures discussed in FSAR Section 2.5.1.1.4.3, but which were not shown in referenced FSAR Figures 2.5.1-208 and 2.5.1-209, to enable a thorough assessment of tectonic features found in the LNP site region in regard to whether they may represent capable tectonic structures. In the response to RAI 2.5.1-18, the applicant incorporated changes to FSAR Section 2.5.1.1.4.3, including modifications to Figures 2.5.1-209 and 2.5.1-222, to further qualify the locations, ages, and types of deformation for tectonic structures in the site region.

Based on review of the applicant's response to RAI 2.5.1-18 and modifications implemented for figures and text in Revision 4 of LNP COL FSAR Section 2.5.1.1.4.3, the staff concludes that the applicant provided appropriate changes in Revision 4 of FSAR Section 2.5.1.1.4.3. The staff makes this conclusion because the modifications provided in Revision 4 of FSAR Section 2.5.1.1.4.3 locate all regional tectonic structures that lie within the LNP site region and qualify the ages and styles of deformation for these structures. Consequently, the staff considers RAI 2.5.1-18 resolved.

2.5.1.4.1.4.2 Charleston Area Tectonic Features

In RAI 2.5.1-21, the staff asked the applicant to summarize existing information on the following tectonic features postulated to occur in the Charleston area: the Ashley River, Charleston, Cooke, Drayton, Gants, and Woodstock faults. FSAR Figure 2.5.1-225 and Table 2.5.1-201 include these faults, but they are not discussed in detail in FSAR Section 2.5.1.1.4.4. In response to RAI 2.5.1-21, the applicant proposed changes to FSAR Section 2.5.1.1.4.4 and incorporated those changes in Revision 4 of LNP COL FSAR Section 2.5.1.1.4.4 to provide a discussion of the six tectonic features in the Charleston area included in FSAR Figure 2.5.1-225 and Table 2.5.1-201, but not initially discussed in the FSAR.

Based on review of the applicant's response to RAI 2.5.1-21 and the modifications included in Revision 4 of FSAR Section 2.5.1.1.4.4, the staff concludes that the applicant provided appropriate changes in Revision 4 of FSAR Section 2.5.1.1.4.4 because the modifications present a discussion of all tectonic features in the Charleston area. Consequently, the staff considers RAI 2.5.1-21 to be resolved.

In RAI 2.5.1-22, the staff asked the applicant to summarize the basis for the conclusion, presented in FSAR Section 2.5.1.1.4.4, that there is low confidence that the ECFS exists. In response to RAI 2.5.1-22, the applicant discussed several studies that assessed the ECFS as a potential seismic source, including the study for the North Anna ESP application as summarized in NUREG-1835 ("Safety Evaluation Report for an Early Site Permit (ESP) at the North Anna ESP Site"). In NUREG-1835, the NRC staff concluded that the geologic, seismic, and geomorphic evidence for the ECFS-North presented by Marple and Talwani (2000) is uncertain, and that most data apply to the southern and central segments of the ECFS. The applicant also pointed out that the VEGP ESP application (SNC, 2007) indicates that the ECFS-South segment is included in the Charleston area seismic source zone and, therefore, need not be incorporated as a separate and distinct seismic source for the LNP site.

Based on the detailed assessment of the ECFS for the North Anna ESP application as discussed in NUREG-1835 and as cited by the applicant in the response to RAI 2.5.1-22, the staff concludes that there is low confidence in the existence of the postulated northern and central segments of the ECFS. The staff further concludes that the updated Charleston seismic source model the applicant used incorporates the southern segment of the ECFS, which lies closest to the LNP site. Consequently, the staff considers RAI 2.5.1-22 to be resolved.

In RAI 2.5.1-45, the staff asked the applicant to discuss the potential tectonic significance of features in the vicinity of the Charleston seismic source, as shown in FSAR Figure 2.5.1-228, which Weems and Lewis (2002) interpreted to exhibit relative uplift during the last 34 Ma (i.e., possibly during Quaternary time). In response to RAI 2.5.1-45, the applicant stated that Weems and Lewis (2002) acknowledged that the areas shown in FSAR Figure 2.5.1-228, which they interpreted to possibly show uplift over the past 34 Ma based mainly on the irregular paleo-topographic surface shown by the bases of Oligocene (33.9 to 23 Ma) through Pliocene (5.3 to 2.6 Ma) units, could be explained either by buried erosional surfaces, syn-depositional or post-depositional tectonic warping, or a combination of those two factors. Based on examination of structure contour maps presented by Weems and Lewis (2002) drawn on the bases of the Oligocene through Pliocene units, the applicant concluded that uplift and subsidence patterns do not persist through time in the same locations, and that the intervening structural lows between the proposed uplifts are highly suggestive of erosion along ancient river channels. This conclusion drawn by the applicant agrees with that made by Southern Nuclear Company in its update of the Charleston seismic source for the VEGP site (SNC, 2006).

Based on the applicant's response to RAI 2.5.1-45 and the staff's independent review of the information presented by Weems and Lewis (2002), the staff concludes that any uplift that may have occurred in the vicinity of Charleston, as proposed by Weems and Lewis (2002) during the last 34 Ma, if it occurred, was pre-Quaternary (< 2.6 Ma) in age. The staff draws this conclusion because Weems and Lewis (2002) documented that the paleo-topographic relief observed at the base of one Oligocene formation in this vicinity could not have formed as a result of post-Oligocene (< 23 Ma) tectonic deformation based on the moderate dip and lack of topographic relief on an overlying unit of Upper Oligocene (28.4 to 23 Ma) age. This field relationship strongly suggests that no post-Oligocene tectonic uplift or subsidence occurred. Consequently, the staff considers RAI 2.5.1-45 to be resolved.

2.5.1.4.1.4.3 Earthquakes in Areas of Extended Crust

In RAI 2.5.1-24, the staff asked the applicant to discuss the potential for large-magnitude earthquakes in areas of extended continental crust, which includes the site region, based on interpretations presented in the current literature cited by the applicant. In response to RAI 2.5.1-24, the applicant indicated that Johnston and others (1994) used a global catalog of moderate to large historical seismicity from SCRs to determine that the largest SCR earthquakes (**M** > 7) occurred in areas of extended crust. The applicant noted that Johnston and others (1994) determined a mean magnitude of **M** 6.3 with a standard deviation of 0.5 for areas of non-extended crust, and a mean magnitude of **M** 6.4 with a standard deviation of 0.84 for extended crust. The applicant also reported that Schulte and Mooney (2005) presented an updated global earthquake catalog, which included **M** 4.5 or larger events for SCRs, and re-evaluated the correlation of intraplate seismicity with ancient extensional rifts, and that their study demonstrated that 52 percent of all seismic events occurred within extended crust. Based on limited borehole data, the applicant noted that crust in the LNP site region experienced some extension during the Mesozoic (251 to 65.5 Ma), although the total amount of crustal extension was minimal. The applicant confirmed that the maximum magnitude distribution for seismic sources in the LNP site region used in the updated seismic source model, discussed in detail in FSAR Section 2.5.2, captures an approximate range of **M** 4.5 to 7.7, such that the PSHA characterization for the LNP site allows for the possible occurrence of large earthquakes in the site region.

Based on the applicant's response to RAI 2.5.1-24, and an independent review of published information cited by the applicant related to large-magnitude earthquakes in areas of extended continental crust, the staff concludes that the applicant analyzed current data to assess the potential for large earthquakes in areas of extended crust, including the site region, and documented that the PSHA characterization for the LNP site properly allows for the possible occurrence of large earthquakes in the site region due to the magnitude range captured in the PSHA. The staff makes this conclusion because interpretations from the current literature cited by the applicant related to maximum magnitude of earthquakes that may occur in areas of extended continental crust, which the staff independently reviewed, support the applicant's statement that the PSHA for the LNP site allows for the occurrence of large earthquakes in the site region. Consequently, the staff considers RAI 2.5.1-24 to be resolved.

2.5.1.4.1.4.4 Staff Conclusions on Regional Tectonic Setting

Based on its review of LNP COL FSAR Section 2.5.1.1.4, the applicant's responses to RAIs 2.5.1-17, 2.5.1-18, 2.5.1-21, 2.5.1-22, 2.5.1-24, and 2.5.1-45, and changes incorporated in Revision 4 of FSAR Section 2.5.1.1.4, the staff finds that the applicant provided thorough and accurate descriptions of the regional tectonic setting of the LNP site, including contemporary tectonic stress, regional structural setting and geophysical framework, regional tectonic structures within a 320-km (200-mi) radius of the site, significant seismic sources at a distance greater than 320 km (200 mi) from the site, and regional seismicity. The staff also concludes that the descriptions provided in LNP COL FSAR Section 2.5.1.1.4 reflect the current literature cited by the applicant and state of knowledge and meet the requirements of 10 CFR 52.79(a)(1)(iii) and 10 CFR 100.23.

2.5.1.4.2 Site Geology

NRC staff focused the review of LNP COL FSAR Section 2.5.1.2, "Site Geology," on the descriptions the applicant provided for physiography, topography, geomorphology, geologic history, stratigraphy, structural geology, geologic hazard and engineering geology within the 40 and 8 km (25 and 5 mi) LNP site vicinity and area, respectively. The staff also focused on the descriptions the applicant provided for certain of these topics for the area within 1 km (0.6 mi) of the site (i.e., the site location). The staff concentrated specifically on the applicant's descriptions of the geologic characteristics, which may enhance the development of karst, including regional, site vicinity, site area, and site location fracture patterns, and of the evidence that the site vicinity has been tectonically quiescent since the beginning of Cretaceous time (i.e., 145.5 Ma).

2.5.1.4.2.1 Site Physiography, Topography, and Geomorphology

In LNP COL FSAR Section 2.5.1.2.1, the applicant described physiography, topography and geomorphology of the LNP site vicinity and site area. The applicant stated that the LNP site lies within the Gulf Coastal Lowlands geomorphic province of the midpeninsular physiographic zone, and that this geomorphic province represents an old karst terrain overlain by marine terrace sediments deposited on a tectonically stable Florida platform during previous higher sea level stands.

Based on review of LNP COL FSAR Section 2.5.1.2.1, as well as independent review of current literature cited by the applicant on physiography, topography, and geomorphology of the site vicinity and site area, the staff finds that the applicant provided a complete and accurate description of site physiography, topography, and geomorphology in support of the LNP COL application

2.5.1.4.2.2 Site Vicinity Geologic History

FSAR Section 2.5.1.2.2 summarizes geologic history of the Florida platform, which includes the site vicinity, from Late Proterozoic (> 542 Ma) to the present. The applicant stated that the Florida Platform has been tectonically quiescent from Cretaceous (145.5 to 65.5 Ma) into Holocene (10,000 years to present) time. The applicant noted that sea level fluctuations, rather than tectonic events, affected sediment distribution in the Florida platform throughout the Neogene (23 to 2.6 Ma) and Quaternary (2.6 Ma to present), and that sea level rose to its present level from Pleistocene (2.6 Ma to 10,000 years) to the present time.

Based on review of LNP COL FSAR Section 2.5.1.2.2, as well as independent review of current literature cited by the applicant on the geologic and tectonic setting of the Florida platform, which documented that the site vicinity has been tectonically quiescent since the start of Cretaceous time, the staff finds that the applicant provided a complete and accurate description of site vicinity geologic history in support of the LNP COL application.

2.5.1.4.2.3 Site Vicinity and Site Area Stratigraphy

FSAR Section 2.5.1.2.3 describes the stratigraphy of the site vicinity and site area. The applicant stated that the lowermost and oldest stratigraphic units are Paleozoic (542 to 251 Ma) shales and quartzite sands overlain by Triassic (252 to 201.6 Ma) diabase. Cretaceous (145.5 to 65.5 Ma) and Cenozoic (65.5 Ma to present) carbonates, consisting of both limestone and dolomite overlain by undifferentiated sediments (surficial sands, clayey sands, and alluvium) of Pleistocene (2.6 Ma to 10,000 yr) to Holocene age (10,000 years to present), comprise the uppermost stratigraphic units. The staff focused the review of FSAR Section 2.5.1.2.3 on aspects of the stratigraphy that may be indicative of karst in the site vicinity and site area.

2.5.1.4.2.3.1 Surficial Quaternary Deposits

In RAI 2.5.1-8, the staff asked the applicant to evaluate the possibility that aerial distribution of thicker surficial Quaternary deposits in areas of lower surface topography may reflect local collapse above dissolution cavities at depth, which allowed deposition of thicker surficial deposits. From cross sections based on borehole data, illustrated in FSAR Figures 2.5.4.2-203A and 2.5.4.2-202A, thickness of Quaternary (2.6 Ma to present) sediments varies from less than 3 m (10 ft) in the site area to at least 24 m (80 ft) at locations near LNP Units 1 and 2. In response to RAI 2.5.1-8, the applicant stated that erosional episodes related to sea level fluctuations removed sediment from the Ocala platform, eventually exposing the Upper Eocene (about 40.4-33.9 Ma) Avon Park Formation, upon which Quaternary sediments accumulated to variable thicknesses. The applicant noted that the erosional surface atop the carbonate sediments in the LNP site region includes incised paleochannels filled with Quaternary sediments, some of which exhibit up to 30 m (98 ft) of relief. Because of the scarcity of dissolution voids encountered in the LNP site borings and the documented erosional and depositional history of the site vicinity, the applicant concluded that the most plausible interpretation of the increased thickness of Quaternary sediments observed in the borings is deposition in paleochannels. As part of the response to RAI 2.5.1-8, the applicant proposed changes to FSAR Sections 2.5.1.2.1.1, 2.5.1.2.1.3, 2.5.1.2.3.3, 2.5.1.2.3.6, 2.5.1.2.5.2, and 2.5.1.2.5.3 to further clarify information regarding LNP site stratigraphy. The staff finds these changes acceptable and verified that the applicant incorporated the changes in LNP COL FSAR Section 2.5.1.2.

Based on review of the applicant’s response to RAI 2.5.1-8 and the changes implemented in FSAR Section 2.5.1.2, the staff concludes that sediment-filled paleochannels are an acceptable explanation for the observed thick Quaternary sediments in the LNP borings. The staff draws this conclusion because there is evidence for this mode of sediment accumulation in the site region, and site characterization boreholes revealed only a few small subsurface voids. In addition, during the site audit conducted in September 2009, the staff confirmed that there is a paucity of subsurface dissolution cavities at the LNP site based on grout uptake in the slanted boreholes drilled for the grout testing program. In February 2010, the staff also examined boring logs, core photographs, and written core descriptions for the six “offset” boreholes, drilled using controlled coring techniques to improve core recovery, which documented that the low recovery horizons noted in the initial site characterization boreholes for LNP Units 1 and 2 marked soft zones in the normal stratigraphic sequence, rather than large subsurface dissolution voids. Consequently, the staff considers RAI 2.5.1-8 to be resolved.

In RAI 2.5.1-9, the staff asked the applicant to discuss whether reactivity to hydrochloric acid (HCL) was the sole test performed to differentiate unconsolidated Quaternary deposits from calcareous silts derived from weathered Avon Park limestone in site characterization boreholes drilled at LNP Units 1 and 2. In response to RAI 2.5.1-9, the applicant stated that Quaternary clastic sediments at the LNP site consist mainly of well-sorted fine quartz sands and silty sands with interbedded clays, and show little reaction to HCL due to a lack of carbonate. The applicant stated that weathered Avon Park Formation carbonates typically lack clastic materials.

Based on review of the applicant's response to RAI 2.5.1-9, the staff concludes that the applicant clarified the additional criterion used to distinguish unconsolidated Quaternary deposits from underlying weathered Avon Park Formation limestone to enable a reasonable estimate of the thickness of Quaternary deposits at the LNP site. The staff makes this conclusion because the observed variation in clastic content of these two stratigraphic horizons is definitive when coupled with the HCL test. Consequently, the staff considers RAI 2.5.1-9 to be resolved.

2.5.1.4.2.3.2 Stratigraphic Data from Boreholes

In RAIs 2.5.1-35 and 2.5.1-48, the staff asked the applicant to explain how Rupert (1988) derived his lithologic descriptions for the deep petroleum exploration wells that penetrated the Avon Park Formation in the site vicinity, and to present the criteria used to conclude that washout of soft carbonate layers produced the no-return and no-recovery zones noted by Rupert (1988) in the logs for these deep petroleum wells, rather than open or filled dissolution voids. In the responses to RAIs 2.5.1-35 and 2.5.1-48, the applicant stated that Rupert (1988) relied on Vernon (1951) and the Florida Geological Survey (FGS) for lithologic descriptions, and noted that none of the driller's logs from the FGS reported dissolution voids in the upper 305 m (1,000 ft) of the deep petroleum exploration boreholes, which passed through the Avon Park Formation. In addition to the previous deep petroleum test wells, which penetrated the Avon Park Formation, the applicant analyzed cores from the LNP site taken from borings that penetrated to 152 m (500 ft) below the ground surface as part of the LNP site geotechnical investigations program and noted that Eocene (55.8 to 33.9 Ma) formations in the site area, including the Avon Park Formation, commonly contain interbedded hard (dolomite) and soft (weathered limestone) horizons. The applicant acknowledged that such a stratigraphic sequence requires careful drilling methods to avoid low core recovery, and reported that initial drilling in the Avon Park Formation often resulted in variable recovery rates. To determine that the poor recovery zones resulted from washout of soft carbonate horizons, the applicant drilled six supplemental boreholes at the LNP site located to be offset approximately 1.5 m (5 ft) from the position of the initial site characterization boreholes. The applicant used controlled coring techniques to improve core recovery and documented the presence of soft zones, rather than dissolution voids, at depth. Based on these field data, the applicant concluded that the no-return and no-recovery zones detected in core samples from the LNP site resulted from washout of soft horizons in the normal stratigraphic sequence.

Based on review of the applicant's responses to RAIs 2.5.1-35 and 2.5.1-48, as well as direct examination of lithologic and geophysical logs for the deep petroleum wells the applicant provided plus review of core samples from grout test holes during the September 2009 site audit and of boring logs, core photographs, and written core descriptions from the six supplemental "offset" boreholes located at the LNP site during February 2010, the staff concludes that the missing zones in the Avon Park Formation are due to washouts of softer horizons in the normal stratigraphic sequence, rather than to large open or filled dissolution voids. Examination of cores from the grout test holes and of data from the six "offset" supplemental boreholes did not reveal the presence of large dissolution voids in the Avon Park Formation at the LNP site. The offset boreholes used minimal down-pressures, lower drilling fluid pressures, slower drilling rates, and a larger diameter core barrel specifically to improve core recovery and determine if missing zones in the Avon Park Formation resulted from voids or washout of soft zones in the normal stratigraphic sequence. These supplemental data documented that the no-recovery zones logged in the initial site characterization boreholes resulted from washout of soft zones, rather than dissolution voids. Therefore, the staff draws this conclusion because the preponderance of field data from boreholes at LNP Units 1 and 2, including the data directly reviewed by NRC staff, strongly supports this interpretation. Consequently, the staff considers RAIs 2.5.1-35 and 2.5.1-48 to be resolved.

Based on review of LNP COL FSAR Section 2.5.1.2.3, review of the applicant's responses to RAIs 2.5.1-8, 2.5.1-9, 2.5.1-35 and 2.5.1-48 and the changes implemented in FSAR Section 2.5.1.2, and independent review of borehole data as described above, the staff finds that the applicant provided a complete and accurate description of site vicinity and site area stratigraphy in support of the LNP COL application.

2.5.1.4.2.4 Site Vicinity and Site Area Structural Geology

In FSAR Section 2.5.1.2.4, the applicant discussed structural geology of the LNP site vicinity and site area. The applicant stated that recent geologic mapping shows no faults within the 40-km (25-mi) radius site vicinity, and that no known structural features have been identified at the site location within a 1-km (0.6-mi) radius of the site. The applicant also discussed regional fracture systems in Florida as initially defined by Vernon (1951), the relationship between those regional fracture systems and smaller-scale fracture patterns near the site, and differing interpretations of the Ocala Platform. The staff focused the review of FSAR Section 2.5.1.2.4 on the characteristics of regional and local fracture systems, including the relationships between fractures and surficial features related to karst development; origin of the Ocala Platform; and postulated tectonic structures in the LNP site vicinity.

2.5.1.4.2.4.1 Observed Fracture Patterns

In RAI 2.5.1-2, the staff asked the applicant to explain whether the local (i.e., outcrop-scale) fractures observed and measured in the site area, referred to as a "subset" of the regional fracture system by the applicant, are smaller-scale fractures that parallel regional fracture trends. The staff also asked the applicant to discuss whether these local fractures exercise control on dissolution. In response to RAI 2.5.1-2, the applicant indicated that "local fractures" refer to vertical outcrop-scale fractures, such as those observed in the Avon Park Formation both along the Waccasassa River and at the abandoned Gulf Hammock quarry, while "regional fractures" are those linear features, identified by Vernon (1951) using aerial imagery, which extend across the site region. Due to the similarity in orientations of these two different scales of fractures, the applicant concluded that the local fractures can be interpreted as smaller-scale features, which reflect the regional fracture system identified by Vernon (1951). Based on field observations of local fracture systems and examination of regional lineament patterns on aerial imagery, the applicant also concluded that local and regional fracture systems strongly influence local dissolution because fractures act as conduits for groundwater flow, and that fractures exercise strong control on dissolution in the site vicinity and site area, particularly where the vertical fractures intersect near-horizontal bedding planes. The applicant cited Dr. T. Scott (personal communications, June 2009) of the FGS, who stated, based on his field observations, that fractures are common in limestone and dolostone quarries in the site vicinity; that fractures in limestones are noticeably enlarged by dissolution; and that fractures in dolostones show less enlargement and limited void development due to dissolution. The applicant also noted consistency between orientations of aligned wetlands and surface depressions associated with mapped lineaments at the LNP site and trends of fracture sets observed and measured in the CR3 site excavations.

Based on review of the applicant's response to RAI 2.5.1-2, as well as direct observation and measurement of local fracture systems along the Waccasassa River in April 2009 and at the Gulf Hammock quarry in September 2009, which enabled a comparison of orientations of the regional and local fracture systems, the staff concludes that outcrop-scale fractures in the Avon Park Formation share a common orientation and likely represent two different scales of the same fracture system. Based on strong confirmation from field data, the staff concludes that fractures exercise strong control on dissolution, and consequently karst development, in the site vicinity and site area, particularly where vertical fractures intersect horizontal bedding planes. Consequently, the staff considers RAI 2.5.1-2 to be resolved.

In RAI 2.5.1-10, the staff asked the applicant to explain the basis for distinguishing "primary" and "secondary" fractures at both local and regional scales, and to provide further description of the local fracture sets in regard to their characteristics and possible origin (i.e., tectonic or non-tectonic). In response to RAI 2.5.1-10, the applicant stated that "primary" and "secondary," as applied to local fractures observed in the Avon Park Formation at the Gulf Hammock quarry and along the Waccasassa River, reflect fracture prominence and frequency to be consistent with descriptions of "major" (primary) and "minor" (secondary) regional fracture sets inferred from photolineament analysis. That is, primary, or major, fractures are most prominent and occur most frequently at both local and regional scales. Based on field measurements of fractures in outcrops at the Gulf Hammock quarry and along the Waccasassa River, the applicant reported that the dominant strike directions of the primary fracture sets are N39W and N51E (i.e., orthogonal fractures), while the secondary fracture sets trend approximately N-S and E-W. The applicant noted that it is not currently possible to define a specific mechanism for development of the primary and secondary fractures sets.

Based on the applicant's response to RAI 2.5.1-10, as well as direct observation and measurement of the local fracture systems in outcrops of the Avon Park Formation along the Waccasassa River in April 2009 and at the Gulf Hammock quarry in September 2009, which provided independent observation of the field relationships, the staff concludes that the distinction the applicant made between primary and secondary fractures is correct and that the orientations of these fracture sets are N39W and N51E (primary) and N-S and E-W (secondary). Based on direct observation of field characteristics of the fractures, the staff also concludes that a specific causative mechanism for the fracture sets cannot be deduced from the field relationships and has not currently been determined by area experts. Consequently, the staff considers RAI 2.5.1-10 to be resolved.

In RAI 2.5.1-39, the staff asked the applicant to discuss the relationship of fractures mapped at the CR3 site to fracture patterns expected to occur at the LNP site, and to regional fracture systems that control stream drainage and sinkhole alignment patterns; to compare the spacing of regional fracture sets with spacing of fractures measured at the CR3 site and anticipated to occur at the LNP site; and to explain why fracture sets interpreted as conjugate, implying that they are tectonically-induced shear fractures, geometrically appear to be orthogonal. In response to RAI 2.5.1-39, the applicant acknowledged that characterization of fractures is important for identifying and mitigating potential hazards related to karst. The applicant reported that the FSAR for CR3 did not provide detailed information about spacing or orientations of fractures observed in the excavation for that plant, so that comparisons of fracture data from the CR3 site could not be made with regional fracture sets or fractures expected at the LNP site. However, the applicant noted that orientations of lineaments defined by slope breaks or alignment of circular depressions and associated wetlands in the LNP site vicinity are consistent with trends of the fracture sets reported for the CR3 site excavation. The applicant stated that, although bedrock exposures at the LNP site location are insufficient to evaluate length or spacing of fracture sets in the Avon Park Formation at the site, fracture spacing observed at the Gulf Hammock quarry and along the Waccasassa River are likely representative of fracture spacing at the LNP site. The applicant also indicated that the fracture sets initially referred to as conjugate are orthogonal based on observed fracture geometry. The applicant incorporated changes in LNP COL FSAR Section 2.5.1.2.4 to further clarify fracture characteristics.

Based on review of the applicant's responses to RAIs 2.5.1-2, 2.5.1-10, and 2.5.1-39 and the changes implemented in LNP COL FSAR Section 2.5.1.2.4, as well as independent review of existing fracture data and direct field observation and measurement of fracture patterns in outcrops of the Avon Park Formation along the Waccasassa River (April 2009) and at the Gulf Hammock quarry (September 2009), the staff concludes that orientations and spacings of fractures observed in the site vicinity and site area and suggested by lineament studies likely reflect orientations and spacing of fractures at the LNP site. The staff also concludes that fractures exercise strong control on dissolution and karst development. The staff draws these conclusions because the preponderance of data from both outcrop studies and lineament analyses do not indicate unique fracture orientations and spacings for the site vicinity, site area, or site location, and do support the interpretation that fractures control dissolution and karst development. Consequently, the staff considers RAIs 2.5.1-2, 2.5.1-10, and 2.5.1-39 to be resolved.

2.5.1.4.2.4.2 The Ocala Arch (or Platform)

In RAI 2.5.1-11, the staff asked the applicant to discuss the origin of the Ocala arch (or platform) in regard to whether it is tectonic or non-tectonic, including any possible association of regional and local fracture sets with development of the Ocala arch. In response to RAI 2.5.1-11 regarding origin of the Ocala arch, the applicant stated that the consensus of knowledgeable FGS geologists is that this feature developed due to differential subsidence, erosion, and sedimentation, rather than as a result of tectonic uplift, and the applicant provided information to document this interpretation. The applicant stated further that the Ocala arch does not exhibit fracture patterns that are uniquely different from the prominent regional fracture systems, which occur statewide, or from the local fracture patterns, which reflect the same trends as the regional fracture systems and are, therefore, related to its genesis.

Based on review of the applicant's response to RAI 2.5.1-11, including independent review of information from area experts at the FGS provided by the applicant, the staff concludes that the applicant provided current information regarding origin of the Ocala arch and possible association of fractures with the platform. The staff draws this conclusion because the applicant assessed the existing data related to origin of the Ocala arch with due consideration for the most current interpretations by FGS geologists, the recognized area experts, who interpret the Ocala arch as non-tectonic in origin and state that regional and local fracture patterns are not unique to the platform. Consequently, the staff considers RAI 2.5.1-11 to be resolved.

2.5.1.4.2.4.3 Postulated Faults and Identification Criteria

In RAI 2.5.1-38, the staff asked the applicant to summarize the information leading to the conclusion that no faults occur within the site vicinity, and to discuss the criteria applied to distinguish faults from fractures. In response to RAI 2.5.1-38, the applicant summarized pertinent data collected by FGS geologists, including geologic maps, cross sections, and structure contour maps, used to determine that no faults occur in the site vicinity (e.g., a statewide 1:750,000-scale geologic map and cross sections from Scott and others, 2001; a 1:126,720-scale geologic map of Levy County from Campbell, 1992; a 1:500,000-scale geologic map of the Floridian aquifer system from Knapp, 1979; and structure contour maps developed by Arthur and others, 2008). None of these data sources developed by area experts from the FGS showed discontinuities or anomalies resulting in the interpretation of surface or subsurface faults in the site vicinity. However, the applicant noted that Arthur et al. (2008) postulated two short segments of a northwest-trending subsurface fault just outside the site vicinity, located about 42 km (26 mi) southeast of the LNP site at its nearest point, based on abrupt changes in thickness in the Suwannee Limestone, as suggested by their structure contour maps. The applicant indicated that there is no surface expression of this postulated fault documented in the current literature cited by the applicant and, if it exists, it is pre-Quaternary (> 2.6 Ma) in age since there is no disruption of Quaternary sediments overlying the inferred fault. Finally, the applicant defined several standard criteria used to distinguish faults from fractures in the site vicinity and site area, all of which depend on finding geologic evidence of displacement along the fault surface as indicated by the presence of sheared materials; visible fault offset or offset inferred from geologic map data; anomalies that suggest truncation or offset of geologic materials; or deposits and geomorphic surfaces disrupted by folding or tilting. By applying these criteria and considering the data collected by FGS geologists, the applicant concluded that no faults occur within the site vicinity.

Based on review of the applicant's response to RAI 2.5.1-38, as well as independent review of pertinent published literature provided by the applicant and data related to structural geology of the site vicinity and site area, including borehole information, the staff concludes that no current data support the existence of faults in the LNP site vicinity or site area. The staff makes this conclusion because the information provided by the applicant, and reviewed by the staff, documented the geologic map data used to assess the presence of faulting. In addition, the staff concludes that the criteria the applicant used to assess the presence of faulting in the site area and site vicinity are the standard criteria for recognition of faults based on field data. Consequently, the staff considers RAI 2.5.1-38 to be resolved.

Based on review of LNP COL FSAR Section 2.5.1.2.4, the applicant's responses to RAIs 2.5.1-2, 2.5.1-10, 2.5.1-11, 2.5.1-38, and 2.5.1-39 and associated changes implemented in FSAR Section 2.5.1.2.4, as well as independent review of pertinent literature cited by the applicant and data and direct field observation of fractures in the Avon Park Formation, the staff finds that the applicant provided a complete and accurate description of structural geology of the site vicinity and site area in support of the LNP COL application.

2.5.1.4.2.5 Site Location Geology

In FSAR Section 2.5.1.2.5, the applicant discussed geology of the site location, including geomorphology, stratigraphy, and karst development. The staff focused the review of FSAR Section 2.5.1.2.5 on the applicant's discussion of factors governing karst development and possible size of subsurface dissolution cavities at the site location.

2.5.1.4.2.5.1 Potential for Rapid Groundwater Flow Conduits

In RAI 2.5.1-31, the staff asked the applicant to discuss available information related to the existence of underground conduits capable of accommodating rapid groundwater flow at or near the LNP site. In RAI 2.5.1-47, the staff asked the applicant to provide a reference for a statement included in the response to RAI 2.5.1-31 that no springs of any noticeable magnitude exist within the LNP site vicinity. In responses to RAIs 2.5.1-31 and 2.5.1-47, the applicant stated that the LNP site lies in a zone of very low recharge, and cited Upchurch (personal communication, 2009) to document the absence of significant springs within the outcrop area of the Avon Park Formation, including the site vicinity. The applicant presented a map modified from Maddox (1993), which shows that no known caves occur within the outcrop area of the Avon Park Formation in Levy and Citrus Counties. Scott and others (2004) reported only two small springs near the LNP site, namely Big King and Little King Springs, which lie to the north-northwest and within 8 km (5 mi) of the site. The applicant concluded that few voids, and no large ones, occurred in the LNP site characterization borings, and reiterated that the upper 150 m (50 ft) of the Avon Park Formation consists primarily of dolomitized limestone (i.e., dolostone), which is less susceptible to dissolution than pure limestone.

Based on review of the applicant's responses to RAIs 2.5.1-31 and 2.5.1-47, as well as independent examination of cores and borehole logs from the LNP site in September 2009 and February 2010 that did not reveal interconnected underground voids or extensive fractures in the subsurface, the staff concludes that no evidence exists for interconnected underground conduits capable of accommodating rapid groundwater flow at or near the LNP site. The staff draws this conclusion because no springs of significant magnitude occur at or near the LNP site, and the site characterization core samples directly examined by staff did not contain interconnected or large voids in the subsurface. Consequently, the staff considers RAIs 2.5.1-31 and 2.5.1-47 to be resolved.

2.5.1.4.2.5.2 Size of Subsurface Dissolution Cavities

The staff requested that the applicant clarify information related to the possible maximum size of subsurface dissolution cavities as provided in a supplemental discussion of the potential for karst development at the site location (Progress Energy, 2008). In RAIs 2.5.1-5 and 2.5.1-7, the staff asked the applicant to address the uncertainty in the estimate of a maximum lateral extent for dissolution cavities of 3 m (10 ft), as cited in the supplemental discussion, and to discuss the potential for coalescing dissolution cavities at depth below LNP Unit 1 or LNP Unit 2. In responses to RAIs 2.5.1-5 and 2.5.1-7, the applicant stated that conservative parameters applied in the analysis of size of subsurface karst features based on grout uptake volume accounted for uncertainties in the subsurface data used to estimate the maximum size of dissolution voids. These conservative parameters included increasing grout volumes used in the void size analysis above the grout uptake volumes calculated from borehole data, specifically by 50-percent for vertical fractures and 100-percent for horizontal bedding planes. The use of the parameters resulted in the applicant defining a dissolution cavity with a maximum lateral dimension of 3 m (10 ft), whereas the maximum void size calculated from actual borehole data was 1.6 m (5.3 ft) in lateral extent. The applicant pointed out that the size of the dissolution cavity used in the analysis is 1.9 times the size of the cavity calculated from borehole data, and thus concluded that the estimate of maximum size of subsurface dissolution cavities presented in the supplemental discussion was conservative. The applicant noted that the degree of dolomitization of the Avon Park Formation, a process, which lowers the likelihood of dissolution, decreased the potential for coalescence of subsurface dissolution cavities. The applicant provided information documenting the fact that dolomites dissolve less readily than pure limestones in response to RAI 2.5.1-1 discussed below in SER Section 2.5.1.4.2.6, “Site Area Geologic Hazard Evaluation.”

Based on the review of the applicant’s responses to RAIs 2.5.1-5 and 2.5.1-7, as well as independent examination of supporting field data from grout test cores in September 2009 and the six “offset” boreholes drilled using controlled boring techniques to improve core recovery and enable assessment of subsurface dissolution cavities and fractures in February 2010, the staff concludes that the estimate of a maximum void size of 3 m (10 ft) in lateral extent is conservative. The staff makes this conclusion because the preponderance of field data indicates that large subsurface dissolution cavities do not occur in the Avon Park Formation at the site location. The supporting field data examined during the September 2009 site audit specifically showed grout uptake only in a single vertical fracture intersected by one of the test grouting boreholes, and no large dissolution cavities occurred in any of the boreholes. The supporting data examined in February 2010 enabled the staff to conclude these data indicate that the low recovery horizons noted in the initial site characterization boreholes for LNP Units 1 and 2 (as examined by staff during the site visit in April 2009) mark soft zones in the normal stratigraphic sequence, rather than large subsurface dissolution cavities. Consequently, the staff considers RAIs 2.5.1-5 and 2.5.1-7 to be resolved.

In RAIs 2.5.1-12 and 2.5.1-46, the staff asked the applicant to discuss what the scale of surficial features may suggest in regard to a maximum lateral dimension for dissolution voids in the subsurface. In responses to RAIs 2.5.1-12 and 2.5.1-46, the applicant indicated that surface morphology of the LNP site is characterized by shallow depressions, classified as solution sinkholes, which vary in size from small, well-defined depressions less than 50 m (64 ft) in diameter and 1 to 2 m (2 to 6 ft) in depth to large, irregular, shallow depressions ranging up to 600 m (2,000 ft) wide. Based on Sinclair and Stewart (1985), the applicant reported that the diameter of these shallow, surficial solution sinkholes observed at the LNP site is not indicative of the size of expected subsurface karst features. Following Sinclair and Stewart (1985), the applicant stated that dissolution is most active at the limestone surface where dissolution features develop, commonly along fractures that allow water to easily percolate into the subsurface, dissolve the limestone, and transport insoluble residues, such that these features indicate shallow dissolution only. The applicant further indicated that deep dissolution does not commonly occur because subsidence of the soil layer occurs as the surface of the limestone dissolves and seals the bottom of the shallow depression, forming a marsh or lake in the depression. The applicant stated that this shallow dissolution process produced the undulating topography characterized by the shallow depressions, which are common over large parts of Florida and which dominate the LNP site.

Based on review of the applicant's responses to RAIs 2.5.1-12 and 2.5.1-46, as well as independent review of Sinclair and Stewart (1985) and other pertinent published literature cited by the applicant, the staff concludes that the shallow solution sinkhole depressions, which dominate the surface of the LNP site, are surficial sinkholes that do not reflect deep dissolution cavities. The staff makes this conclusion because experts in the region have documented this interpretation based on borehole data that do not reveal deep dissolution cavities beneath these solution sinkholes. Consequently, the staff considers RAIs 2.5.1-12 and 2.5.1-46 to be resolved.

Based on review of LNP COL FSAR Section 2.5.1.2.5, the applicant's responses to RAIs 2.5.1-5, 2.5.1-7, 2.5.1-12, 2.5.1-31, 2.5.1-46, and 2.5.1-47, as well as independent review of pertinent literature cited by the applicant and data and direct observation of grout test cores in September 2009 and examination of information from the six "offset" boreholes drilled using controlled boring techniques to improve core recovery in February 2010, the staff finds that the applicant provided a complete and accurate description of site location geology in support of the LNP COL application.

2.5.1.4.2.6 Site Area Geologic Hazard Evaluation

FSAR Section 2.5.1.2.6 presents an evaluation of the geologic hazards at the LNP site. The applicant noted that the LNP site is located in an area of infrequent and low seismicity, and that no capable tectonic sources occur in the site area. The applicant did not indicate whether field reconnaissance studies or literature searches cited by the applicant were performed to determine if paleoliquefaction features (i.e., indicators of prehistoric earthquake activity) occur in the site region, vicinity, or area. The applicant concluded that the only geologic hazard identified in the LNP site area is potential surface deformation resulting from carbonate dissolution and collapse or subsidence related to karst development.

The staff focused the review of FSAR Section 2.5.1.2.6 on qualification of the dissolution rates cited for development of karst at the LNP site, and whether paleoliquefaction features may exist in the site region, site vicinity, or site area as indicators of prehistoric seismic events.

2.5.1.4.2.6.1 Proposed Dissolution Rates

In RAI 2.5.1-1, the staff asked the applicant to summarize the technical basis for the dissolution rates cited in the LNP COL FSAR, and to document the statement in the FSAR that dolomitized limestone dissolves more slowly than pure limestone. In response to RAI 2.5.1-1, the applicant indicated that a comparison of the more dolomitized Avon Park Formation with the less dolomitized Ocala Formation at the CR3 site provided the dissolution rate of less than 1E-4 percent per year proposed for the Avon Park Formation at the LNP site. The applicant stated that the dissolution rate for the Ocala Formation at the CR3 site, 1E-4 percent per year, calculated out to 6E-3 percent over the projected 60-year life of that plant. Regarding the degree of dolomitization of the Avon Park Formation at the LNP site, which converts limestone to dolomite, the applicant reported that 18 of 20 samples from the LNP site analyzed during LNP site characterization investigations exhibited a high degree of dolomitization, containing less than 50 percent calcium carbonate ($CaCO_3$). The applicant reported that Easterbrook (1999) documented that about 60 percent $CaCO_3$ is necessary to form karst, and about 90 percent may be required to fully develop karst. Also citing Easterbrook (1999), the applicant stated that dolomites, composed of calcium-magnesium carbonate [$CaMg(CO_3)_2$], have a lower permeability than non-dolomitized limestones. This characteristic diminishes dissolution and karst formation. The applicant concluded that the potential for dissolution and karst formation at the LNP site during the life of the plant is not significant, and added that a monitoring program would be established for the LNP plant to confirm this low dissolution rate as part of the groundwater monitoring program.

Based on review of the applicant's response to RAI 2.5.1-1, as well as an independent review of the references cited therein, the staff concludes that there is a strong technical basis for the proposed low dissolution rate at the site location. The staff draws this conclusion because characterization of the Avon Park Formation indicates that this unit is dolomitized at depth, and there is a preponderance of published information to document that dolomites and dolomitic limestones have much lower dissolution rates than pure limestones. Consequently, the staff considers RAI 2.5.1-1 to be resolved. The staff further concludes that the only geologic hazard identified in the LNP site area is potential non-tectonic surface deformation resulting from collapse or subsidence related to karst development. The staff addresses this potential hazard in SER Section 2.5.3.4.8.

2.5.1.4.2.6.2 Paleoliquefaction Features

In RAI 2.5.1-41, the staff asked the applicant to discuss the efforts undertaken to document the presence or absence of paleoliquefaction features in the site region, site vicinity, and site area, or to explain why such efforts were not thought to be necessary. In response to RAI 2.5.1-41, the applicant stated that no published or unpublished reports reviewed during site characterization or preparation of FSAR Section 2.5 identified paleoliquefaction features in the LNP site region. In addition, based on discussions with Dr. T. Scott of the FGS (personal communications, 2009), the applicant confirmed that no paleoliquefaction features have been reported anywhere in Florida. The applicant also discussed observations made during field reconnaissance in the LNP site vicinity and site area, which resulted in the suggestion that detailed studies, would not likely provide data useful for evaluating the occurrence, location, or size of prehistoric earthquakes in the LNP site vicinity and area. The applicant indicated that a paucity of exposures and limited stratigraphy favorable for liquefaction in the site vicinity, including along major drainages, rendered it difficult to document the presence or absence of paleoliquefaction features. Therefore, based on existing information documenting that no reported paleoliquefaction features occur in the site region and that Florida currently has a low risk of earthquakes, communications with a knowledgeable expert from the FGS indicating that no paleoliquefaction features have been observed in Florida, and the existence of only sparse exposures, which lack materials favorable for liquefaction, the applicant stated that detailed paleoliquefaction studies were not performed to assess the possibility of prehistoric earthquakes in the site region, site vicinity, or site area.

Based on review of the applicant's response to RAI 2.5.1-41, the staff concludes that paleoliquefaction features are not likely to exist in the site region, site vicinity, or site area. The staff draws this conclusion because investigations by experts knowledgeable about the geology and seismicity of Florida have not demonstrated the existence of paleoliquefaction features anywhere in the State of Florida. In addition, the Florida platform on which the LNP site is located reflects regional tectonic quiescence since the Cretaceous (145.5 Ma) as discussed in FSAR Section 2.5.1.1.2, and there is no geologic or geomorphic evidence of Quaternary (2.6 Ma to present) faulting as discussed in FSAR Section 2.5.3. Consequently, the staff considers RAI 2.5.1-41 to be resolved.

Based on review of FSAR Section 2.5.1.2.6 and the applicant's responses to RAIs 2.5.1-1 and 2.5.1-41, the staff finds that the applicant provided a complete and accurate description of potential geologic hazards in the site area in support of the LNP COL application.

2.5.1.4.2.7 Site Engineering Geology Evaluation

FSAR Section 2.5.1.2.7 discusses site engineering geology, including engineering behavior of soil and rock; zones of alteration, weathering, and structural weakness; karst features; and deformation zones. The applicant indicated that FSAR Section 2.5.4 discusses engineering behavior of soil and rock materials at the site, and that, if any karst features occur in the LNP foundation rocks, then they will be addressed through appropriate design considerations as explained in that FSAR section. The applicant stated that no zones of structural weaknesses (e.g., extensive fracture zones or faults) have been identified at the LNP site; that the Avon Park Formation does exhibit weathering alteration and varying degrees of dissolution; and that, with the exception of possible paleosinkholes, no deformation zones have been encountered.

Based on the review of FSAR Section 2.5.1.2.7, as well as independent review of current literature cited by the applicant related to geologic and geotechnical characteristics of the LNP site, the staff finds that the applicant provided a complete and accurate description of site engineering geology in support of the LNP COL application.

2.5.1.5 *Post Combined License Activities*

There are no post-COL activities related to FSAR Section 2.5.1. However, in SER Section 2.5.3.4.8 ("Potential for Surface Deformation at the Site"), the staff identified a geologic mapping

License Condition related to FSAR Section 2.5.3.8.1 as the responsibility of the COL licensee. SER Section 2.5.3.5 addresses this License Condition.

2.5.1.6 *Conclusion*

The NRC staff reviewed the application and checked the referenced DCD. The staff confirmed that the applicant addressed the required information related to basic geologic and seismic characteristics, and that there is no outstanding information expected to be addressed in the LNP COL FSAR related to these characteristics. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

As set forth above, the staff has reviewed the information in LNP COL 2.5-1 and finds that the applicant provided a thorough characterization of basic geologic and seismic information for the LNP site, 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 the COL site, in accordance with NRC regulations provided in 10 CFR 100.23 and 10 CFR 52.79(a)(1)(iii) and the guidance provided in RG 1.208. Based on the applicant's geologic investigations of the site region and site area, 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 and rock units at the site. The staff also concludes that there is no potential for the effects of human activity (i.e., mining activity or ground water injection or withdrawal) to compromise the safety of the site. Therefore, the staff concludes that the proposed COL site is acceptable from the standpoint of basic geologic and seismic information and meets the requirements of 10 CFR 52.79(a)(1)(iii) and 10 CFR 100.23.

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 ground motion response spectrum (GMRS), which must meet the regulations for the Safe Shutdown Earthquake (SSE) provided in 10 CFR 100.23. The GMRS is defined as the free-field horizontal and vertical GMRS 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 miles) 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*

Section 2.5 of the LNP COL FSAR, Revision 9, incorporates by reference Section 2.5.2 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Section 2.5.2, the applicant provided site-specific information to address the following:

AP1000 COL Information Item

- LNP COL 2.5-2

The applicant provided additional information in LNP COL 2.5-2 to address COL Information Item 2.5-2. LNP 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, PSHA, seismic wave transmission characteristics and the SSE ground motion.

- LNP COL 2.5-3

The applicant provided additional information in LNP 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

FSAR Section 2.5.2.1 describes the development of a current earthquake catalog for the LNP Units 1 and 2 site. The applicant used the methodology provided in RG 1.208 by starting with the EPRI- Seismicity Owners Group (SOG) historical earthquake catalog (EPRI NP-4726-A, 1988), which is complete from 1627 to 1984. The applicant updated EPRI-SOG's historical earthquake catalog with seismicity from 1985 through December 2006 using current seismicity catalogs. The current seismicity catalogs include data from the Advanced National Seismic System (ANSS), the International Seismological Centre (ISC), Virginia Tech Seismological Observatory's Southeastern U.S. Seismic Network, and the U.S. Geological Survey (USGS) National Earthquake Information Center (NEIC). The applicant deleted duplicate entries for the final updated catalog and converted the different magnitude scales used by the catalogs to body wave magnitude (m_b), which is the scale used in the EPRI-SOG catalog.

The applicant's seismicity catalog update includes the seismicity data from the Bellefonte Geotechnical, Geological, and Seismological (GG&S) earthquake catalog (TVA, 2006) extended to latitude 23°N and longitude 107°W, and through December 2006. This extended coverage includes the LNP Units 1 and 2 320-km (200-mi) site radius and seismicity throughout the Gulf of Mexico. These were not included in the Bellefonte GG&S earthquake catalog. The geographic distribution of earthquakes in the applicant's updated earthquake catalog is provided in SER Figure 2.5.2-1.

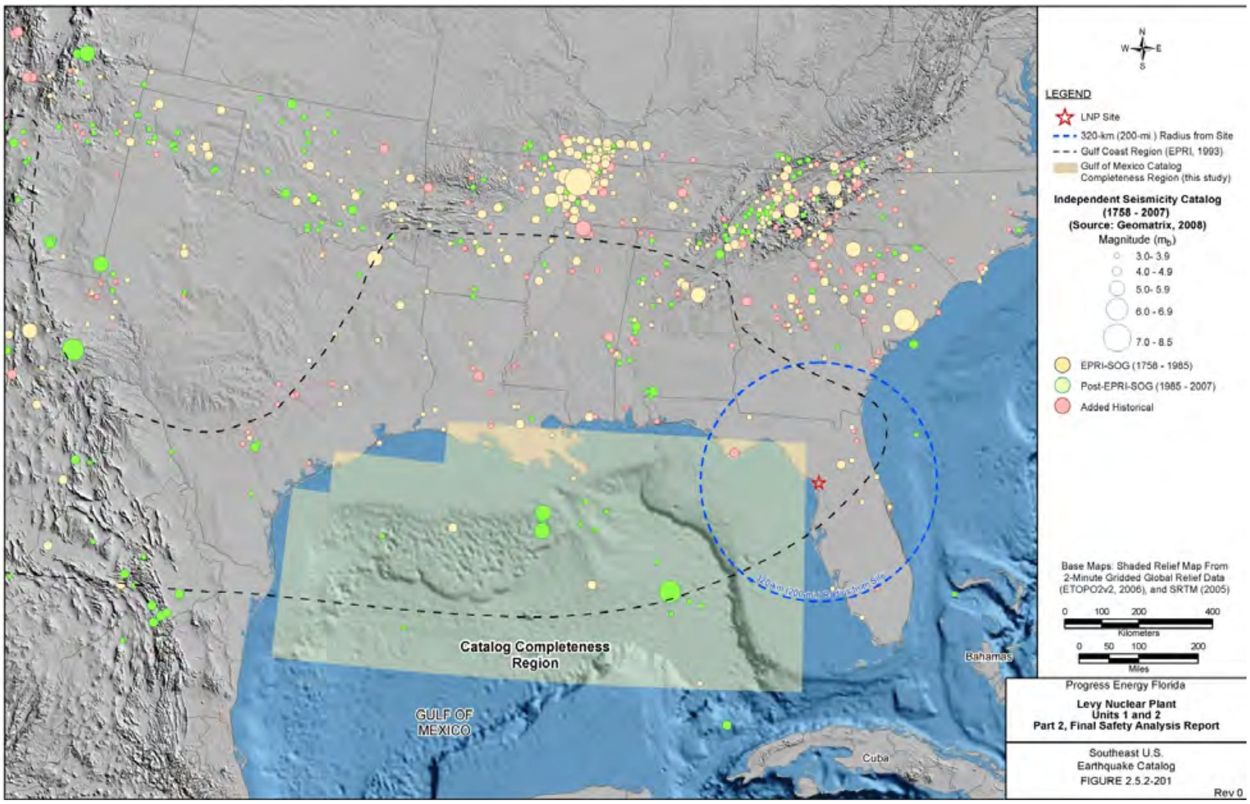


Figure 2.5.2-1. Topography and Bathymetry Map Showing the Applicant's Updated Earthquake Catalog and the Location of the LNP Site (FSAR Figure 2.5.2-201)

2.5.2.2.1.1 Earthquakes that May Influence Seismic Hazard at the LNP Site

In FSAR Section 2.5.2.1.2, the applicant identified three regions where earthquakes occur that may significantly influence the seismic hazard at the LNP Units 1 and 2 site. The first is the LNP Units 1 and 2 site region, which encompasses seismicity within the 320-km (200-mi) site radius. The second area includes the earthquakes in the Gulf of Mexico and the third region contains the historic earthquakes of the Charleston, South Carolina region. The applicant's description of these regions is summarized below.

2.5.2.2.1.1.1 Earthquakes within the 320-km (200-mi) LNP Site Region

The applicant noted that there are fifteen earthquakes with m_b greater than or equal to 3 located within 320 km (200 mi) of the LNP Units 1 and 2 site, which is shown in SER Figure 2.5.2-1. As described in the applicant's updated earthquake catalog, event magnitudes do not exceed m_b of 4.3 and the earthquakes occurred between the years 1826 and 2006. Out of these earthquakes thirteen earthquakes have magnitudes (m_b) between 3 and 4 ($3 \leq m_b < 4$) and two earthquakes have magnitudes greater than 4 ($4 \leq m_b < 4.3$). The applicant noted that estimates of Modified Mercalli Intensity (MMI) and strong motion records are not available for these earthquakes. The largest earthquake within the LNP site region occurred on January 13, 1879, near St. Augustine, Florida at a distance of 76 km (47 mi) from the LNP site, and at a magnitude of m_b 4.3.

Earthquakes in the Gulf of Mexico

As shown in SER Figure 2.5.2-1, the southwestern portion of the 320-km (200-mi) site region extends into the Gulf of Mexico. However, the original EPRI-SOG earthquake catalog covers only a small portion of the Gulf of Mexico along the US coastline. The applicant updated the original EPRI-SOG catalog with seismicity within the Gulf of Mexico between the latitude 24° North (N) to 32° N and longitude 100° West (W) to 83° W. This update was prompted by the occurrence of two moderate-sized seismic events in the Gulf region. These two events (m_b 4.9 on February 10, 2006 and m_b 6.0 on September 10, 2006) are shown in SER Figure 2.5.2-1. The applicant calculated the magnitude of these events from the average of catalog reported m_b estimates and m_b estimates converted from other magnitude scales as reported in FSAR Table 2.5.2-201. The applicant noted that the m_b 6.0 earthquake is the closest Gulf of Mexico earthquake to the LNP Units 1 and 2 site. The effect of this earthquake was felt in Crystal River, Florida, which is located approximately 16 km (10 mi) south of the LNP site. Reports from Crystal River indicate an MMI of IV, which generally means that the ground motion resulting from the earthquake was moderately felt by Crystal River residents but no damage was sustained. To characterize periods of catalogue completeness for the Gulf of Mexico, the applicant adopted the procedure used in the EPRI-SOG study and divided the seismicity catalog into time frames and the event magnitude scale into intervals and determined a probability of completeness for each interval. The applicant's probabilities of detection for the Gulf of Mexico Completeness Region are listed in FSAR Table 2.5.2-211.

Historic Earthquakes of the Charleston, South Carolina Region

The September 1, 1886, Charleston, South Carolina earthquake is the largest (m_b 6.8) known event to occur in the southeastern United States. According to the LNP updated earthquake catalog, the event was located 494 km (307 mi) north of the LNP Units 1 and 2 site. Ground motion associated with the event was felt throughout northern Florida and the effects of several aftershocks were felt as far as Jacksonville, Florida, which is located approximately 217 km (135 mi) northeast of the LNP Units 1 and 2 site.

2.5.2.2.2 Geologic and Tectonic Characteristics of the Site and Region

FSAR Section 2.5.2.2 describes the original EPRI-SOG (EPRI, 1988) seismic source models that contribute to 99 percent of the total hazard at the LNP Units 1 and 2 site. These contributing EPRI-SOG sources are from the 1989 EPRI-SOG PSHA study. In that study, EPRI-SOG analyzed seismic source models for the Crystal River Unit 3 Nuclear Generating Plant (CR3) located about 15 km (10 mi) from the LNP Units 1 and 2 site. The applicant began its assessment of seismic sources at the LNP Units 1 and 2 site using the sources found to contribute to 99 percent of the total hazard at the CR3 site based on the 1989 EPRI-SOG study. EPRI-SOG designated six earth science teams (ESTs) to develop seismic source models for the Central and Eastern United States (CEUS), which were completed in 1986. The applicant also reviewed available geological, seismological, and geophysical data since the late 1980's to evaluate the need for modifications to the original EPRI-SOG ESTs' seismic source models. SER Section 2.5.2.2.4 describes the applicant's sensitivity studies of these potential source zone updates as well as potential new seismic sources.

2.5.2.2.2.1 Summary of EPRI Seismic Sources

Consistent with RG 1.208, the applicant used the 1986 EPRI-SOG seismic source model for the CEUS as a starting point for its seismic source characterization of the LNP Units 1 and 2 site. The 1986 EPRI-SOG seismic source model is comprised of input from six independent ESTs that include the Bechtel Group, Dames & Moore, Law Engineering, Rondout Associates, Weston Geophysical Corporation, and Woodward-Clyde Consultants (WCC). The 1989 EPRI-SOG study (EPRI, 1989) subsequently incorporated each of the EST models into a PSHA for nuclear power plant sites in the CEUS. FSAR Section 2.5.2.2.1 and FSAR Tables 2.5.2-202 through 2.5.2-207 detail the primary seismic sources developed by each of the six ESTs that contributed to 99 percent of the total hazard at the CR3 site and were assessed by the applicant for contributing to the hazard at the LNP Units 1 and 2 site. The seismic source models developed by the six ESTs are briefly described below.

2.5.2.2.2.1.1 Bechtel Group

Five Bechtel Group seismic source zones contributed to 99 percent of the total hazard at the CR3 site. The Charleston Area (H) and Faults (N3) sources represent locations for the 1886 Charleston earthquake and have an assigned maximum m_b of 7.4. The Atlantic Coastal Region (BZ4) has an assigned maximum m_b of 7.4 and is a background source that encompasses eastern Mesozoic basins and the Charleston area sources. The Gulf Coast Zone (BZ1) is a background source that encompasses most of the site region and extends from western Texas to eastern Florida, while the Southern Appalachians Region (BZ5) covers the large area of the southern Appalachians to the north of the site. Both the Gulf Coast Zone and the Southern Appalachians Region have an assigned maximum m_b of 6.6.

2.5.2.2.2.1.2 Dames & Moore

Six seismic sources defined by Dames & Moore contributed to 99 percent of the total hazard at the CR3 site. The South Cratonic Margin (41) covers the continental margin region and has an assigned maximum m_b of 7.2. The Southern Appalachian Mobile Belt source (53) has an assigned maximum m_b of 7.2 and characterizes rocks that have undergone multiple periods of deformation. The Charleston region (54) and Charleston Mesozoic Rift (52) sources represent sources for the 1886 Charleston earthquake and the surrounding area and have assigned maximum m_b of 7.2. The Southern Coastal Margin source (20) is a large background source that extends from Mexico, along the Texas coastal plain to eastern Florida with an assigned maximum m_b of 7.2. The Paleozoic (Appalachian) Fold Belt source (4) covers the folded mountain belt from New York to Alabama and has an assigned maximum m_b of 7.2.

2.5.2.2.2.1.3 Law Engineering

Eight Law Engineering seismic source zones contributed to 99 percent of the total hazard at the CR3 site. The Eastern Basement (17) source encompasses a larger area of buried Precambrian-Cambrian normal faults, where the assigned maximum m_b is 6.8. The Eastern Basement Background source (217) covers a pattern of magnetic anomalies and a negative Bouguer gravity signature, where the assigned maximum m_b is 5.7. The Reactivated Normal Faults source (22) has a maximum m_b of 6.8 and describes seismicity along the Eastern Seaboard region. The Charleston source (35) represents a source for the 1886 Charleston earthquake and the surrounding area and has an assigned maximum m_b of 6.8. The Mesozoic Basin source (8) encompasses the northeast-trending troughs of Triassic to early Jurassic age with a maximum m_b of 6.8. The South Coastal Block Zone (126) encompasses most of the site region and extends from Mexico, through Texas, to eastern Florida and has an assigned maximum m_b of 4.9. The Eastern Piedmont source (107) is located north of the LNP site region and has a maximum m_b of 6.8. The Brunswick source (108) is a background source that characterizes a basement terrane and has an assigned maximum m_b of 6.8.

2.5.2.2.2.1.4 Rondout Associates

Six seismic source zones defined by Rondout Associates contributed to 99 percent of the total hazard at the CR3 site. The Southern New York–Alabama Lineament source (13) characterizes a change in the regional magnetic anomaly pattern in basement rocks with an EPRI assigned maximum m_b of 6.5. The Charleston source (24) represents a source for the 1886 Charleston earthquake, including the Ashley River and Woodstock Faults, using a maximum m_b of 7.0. The Southern Appalachians source (25) characterizes anomalies associated with the New York-Alabama lineament and has an assigned maximum m_b of 7.0. The South Carolina Zone source (26) has a maximum m_b of 6.8 and encompasses cross-cutting fracture zones on the aeromagnetic map of South Carolina. The Appalachian Crust source (49) encompasses the location of the LNP site and is a background source of crust made of an accretionary terrane formed after the Precambrian. This source has an assigned maximum m_b 5.8. The Gulf Coast to Bahamas Fracture Zone (51) encompasses most of the site region and extends from Mexico and Texas to eastern Florida. The maximum assigned m_b for this zone is 5.8.

2.5.2.2.2.1.5 Weston Geophysical Corporation

Six Weston Geophysical Corporation seismic source zones contributed 99 percent of the total hazard at the CR3 site. The New York–Alabama–Clingman Block source (24) characterizes a linear block of seismicity in the Southern Appalachians, where the assigned maximum m_b is 6.6. The Charleston source (25) is localized on and around the city of Charleston, South Carolina and represents a source for the 1886 Charleston earthquake using a maximum m_b of 7.2. The South Carolina Zone source (26) describes the larger region of the state of South Carolina using an assigned maximum m_b of 7.2. The Southern Appalachian source (103) is a background zone located north of the site region and has a maximum assigned m_b of 6.6. The Southern Coastal Plains source (104) is a background source of the south coastal plain seismicity zone and has an assigned maximum m_b of 7.2. The Gulf Coast Zone (107) is a large areal source that extends from Mexico and Texas to eastern Florida and encompasses most of the site region. The maximum m_b assigned is 6.0.

2.5.2.2.2.1.6 Woodward-Clyde Consultants

Four Woodward-Clyde Consultants seismic source zones were found to contribute 99 percent of the total hazard at the CR3 site. The Greater South Carolina sources (29, 29A, and 29B) characterize seismicity in South Carolina, Georgia and western North Carolina with an assigned maximum m_b of 7.4. The Charleston source (30) represents a local source for the 1886 Charleston earthquake with a maximum m_b of 7.5. The Blue Ridge Zone and Alternative sources (31 and 31A) extend from the south to the central Appalachians and have a maximum m_b of 7.0. The Crystal River source (B36) encompasses most of the state of Florida, including the CR3 site and LNP sites and has maximum m_b of 6.5.

2.5.2.2.2.2 Post-EPRI Seismic Source Characterization Studies

In accordance with the guidance in RG 1.208, the applicant reviewed seismic source characterization studies published since the original EPRI-SOG (1988) study to assess the need to update the 1986 EPRI-SOG seismic source parameters. Based on LNP's updated seismicity catalog and the results of several post-ERPI studies (Frankel et al., 2002; SCDOT, 2003; and SNC, 2006), the applicant updated the EPRI-SOG (1988) characterizations of the Charleston seismic source zone and the Gulf Coastal Source Zones (GCSZ).

2.5.2.2.2.2.1 USGS National Seismic Hazard Mapping Project

The applicant stated that as part of the 2002 update of the National Seismic Hazard Maps, the USGS developed a model of the Charleston source that incorporates available data regarding recurrence, M_{max}, and geometry of the source zone. The USGS model used two equally weighted source geometries: (1) an areal source enveloping most of the tectonic features and liquefaction data in the greater Charleston area, and (2) a north-northeast-trending elongated areal source enveloping the southern half of the southern segment of the proposed East Coast fault system. For maximum moment magnitudes (M_{max}), the study defines a distribution of M 6.8 (0.20), 7.1 (0.20), 7.3 (0.45), 7.5 (0.15). For recurrence, USGS (Frankel et al., 2002) adopted a mean paleoliquefaction-based recurrence interval of 550 years and represented the uncertainty with a continuous lognormal distribution. The applicant chose to update the EPRI Charleston seismic source using the Updated Charleston Seismic Source (UCSS) model presented in Southern Nuclear Company's (SNC) ESP (SNC, 2007) application for VEGP Units 3 and 4, which is discussed below.

2.5.2.2.2.2.2 Updated Charleston Seismic Source Zone

The site of the 1886 large-magnitude Charleston, SC earthquake lies approximately 494 km (307 mi) north of the LNP Units 1 and 2 sites. The applicant included the UCSS zone developed by SNC (2006), because the Charleston area is the closest principle source of seismic activity to the LNP Units 1 and 2 site. SNC's new zone accounts for updated information regarding the location, size, and rate of earthquake occurrence for large-magnitude earthquakes in the vicinity of Charleston, SC. The UCSS model includes four possible source regions as shown in FSAR Figure 2.5.2-213. In the model, the four seismic sources are treated as potential zones capable of producing large earthquakes. The size of the characteristic earthquake is assumed to vary from magnitude M_{max} 6.7 to 7.5 in each of these four alternative source zones. The applicant used these seismic source geometries and modeled the occurrence of large repeated earthquakes in the Charleston region. Since the distance between the updated Charleston sources and the LNP site is relatively far at 494 km (307 mi), the applicant updated only the EPRI-SOG (1988) source models for the large-magnitude earthquakes within the Charleston source zone. The applicant assumed that smaller magnitude earthquakes of less than 6.7 at such large distances would not significantly affect the seismic hazard at the LNP Units 1 and 2 sites. Therefore the applicant retained the 1986 EPRI-SOG Charleston sources but limited the Mmax in those sources to m_b 6.6.

2.5.2.2.2.2.3 Gulf Coastal Source Zone

The applicant's updated earthquake catalog includes the two 2006 Gulf of Mexico earthquakes that exceed the bounds of the upper end of the M_{max} distributions for a few EPRI-SOG source models for the Gulf Coast. These earthquakes are the February 10, 2006, m_b of 4.9 earthquake and the September 10, 2006, m_b of 6.0 earthquake. Because of this, the applicant revised five of the six ESTs' M_{max} distributions for GCSZ background sources that contain the LNP site. The applicant's updates to the GCSZs are the same as those made in the South Texas Project (STP) Units 3 and 4 COL application (STPNOC, 2008) and are listed in SER Table 2.5.2-1. The applicant concluded that the increases in M_{max} adequately accounts for the February 10 and September 10, 2006, earthquakes and any potential association between the earthquakes within the Gulf of Mexico and proposed normal faults along the edge of the continental shelf.

Table 2.5.2-1. EPRI-SOG EST GCSZ updates from the STP Unit 3 and 4 COLA. (FSAR Table 2.5.2-209)

EPRI-SOG EST	SOURCE	DESCRIPTION	PROBABILITY OF ACTIVITY	M_{max} DISTRIBUTIONS EPRI-SOG (1989) m_b [WEIGHTS]	UPDATED M_{max} DISTRIBUTIONS STP Unit 3 and 4 (STPNOC, 2008) m_b [WEIGHTS]
Bechtel Group	BZ1	Gulf Coast	1.0	5.4 [0.1] 5.7 [0.4] 6.0 [0.4] 6.6 [0.1]	6.1 [0.1] 6.4 [0.4] 6.6 [0.5]

Table 2.5.2-1. EPRI-SOG EST GCSZ updates from the STP Unit 3 and 4 COLA. (FSAR Table 2.5.2-209)

EPRI-SOG EST	SOURCE	DESCRIPTION	PROBABILITY OF ACTIVITY	M_{max} DISTRIBUTIONS EPRI-SOG (1989) m_b [WEIGHTS]	UPDATED M_{max} DISTRIBUTIONS STP Unit 3 and 4 (STPNOC, 2008) m_b [WEIGHTS]
Dames & Moore	20	South Coastal Margin	1.0	5.3 [0.8] 7.3 [0.2]	5.5 [0.8] 7.3 [0.2]
Law Engineering	126	South Coastal Block	1.0	4.6 [0.9] 4.9 [0.1]	5.5 [0.9] 5.7 [0.1]
Rondout Associates	51	Gulf Coast to Bahamas Fracture Zone	1.0	4.8 [0.2] 5.5 [0.6] 5.8 [0.2]	6.1 [0.3] 6.3 [0.55] 6.5 [0.15]
Weston Geophysical Corporation	107	Gulf Coast	1.0	5.4 [0.71] 6.0 [0.29]	6.6 [0.89] 7.2 [0.11]
Woodward-Clyde Consultants	B43	Central US Backgrounds	NA	4.9 [0.17] 5.4 [0.28] 5.8 [0.27] 6.5 [0.28]	No update

2.5.2.2.3 Correlation of Earthquake Activity with Seismic Sources

FSAR Section 2.5.2.3 describes the correlation of updated seismicity with the EPRI-SOG seismic source models. As described above, the applicant created an updated seismicity catalog covering the LNP site region as part of FSAR Section 2.5.2.1.1. The applicant compared the distribution of earthquake epicenters from the updated seismicity catalog with the seismic sources characterized by each of the EPRI-SOG ESTs, and drew the following conclusions:

- There are no new identifiable seismic sources or active geologic features within the 320-km (200-mi) radius site region and all earthquake activity follows the pattern identified in the EPRI-SOG characterizations. The updated earthquake catalog has spatial patterns and estimated seismicity occurrence rates similar to that of the EPRI-SOG earthquake catalog. Therefore, the applicant made no significant revisions to the EPRI-SOG seismic source geometries or recurrence rates.

- The two 2006 earthquakes that occurred in the Gulf of Mexico are not covered by the M_{max} used by some of the EPRI-SOG ESTs for their Gulf Coast seismic source models. As a result, the applicant revised some of the ESTs' M_{max} distributions for its Gulf Coast models.

- The 1886 Charleston, South Carolina earthquake is the largest historical earthquake to occur in the southeastern United States and the applicant considered this event the closest principle source of seismic activity to the LNP Units 1 and 2 site. The EPRI-SOG teams considered the 1886 earthquake, but more recent studies have further studied alternative source locations, M_{max} values, and large-magnitude recurrence rates. Therefore, the applicant incorporated the findings of the other studies to more adequately characterize the Charleston seismic zone.

2.5.2.2.4 Probabilistic Seismic Hazard Analysis and Controlling Earthquake

FSAR Section 2.5.2.4 presents the results of the applicant's probabilistic seismic hazard analysis (PSHA) for the LNP Units 1 and 2 site. In performing its PSHA, the applicant followed the guidance provided in RG 1.208 to determine the seismic hazard curves and controlling earthquakes for the LNP Units 1 and 2 site. The applicant based its analyses on the original EPRI hazard study (1989) and used the seismic sources identified in EPRI-SOG's 1988 study and updated them as necessary. The PSHA curves generated by the applicant represented generic hard rock conditions characterized by a V_S in excess of 2.7 kilometers per second (km/s) (9,000 feet per second (fps)). The applicant also described the earthquake potential for the site in terms of the 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. Before determining the controlling earthquakes, the applicant updated five of the six GCSZ defined by EPRI (1989) and used the new ground motion models described below.

2.5.2.2.4.1 PSHA Inputs

Before performing the PSHA, the applicant updated the GCSZ inputs from the original 1989 EPRI study and used the updated EPRI (2004, 2006) ground motion models instead of the ground motion models used in the original EPRI study (1989).

2.5.2.2.4.1.1 Seismic Source Model

In order to conduct PSHAs 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 identification of active seismic source zones in the area, compilation of a comprehensive list of earthquakes from the historical records and earthquakes that were recorded instrumentally, determination of earthquake occurrence rates in each of the seismic zones and their probability of occurrence, estimation of maximum magnitudes, and choosing ground motion prediction equations relevant to that region. As summarized above in SER Section 2.5.2.2.2, the seismic sources in the EPRI-SOG study consisted of six alternative seismic source models developed by six ESTs for the CR3 site. The applicant used these seismic source models as the starting point and updated them based on available new information. The applicant modified the EPRI-SOG source models as follows:

- For all sources identified as the Charleston source from each of the six EPRI EST models the Mmax was limited to m_b 6.6. The UCSS source model (SNC, 2006) was used to represent Charleston repeated large magnitude earthquakes (Mmax 6.7 to

7.5).Revised M_{max} distributions for five of the six EPRI EST seismic source models within the region of the GCSZ that contain the LNP Units 1 and 2 site, consistent with the updates made in the STP Units 3 and 4 COL application (STPNOC, 2008), as described in SER Table 2.5.2-1.

2.5.2.2.4.1.2 Ground Motion Models

The applicant used the ground motion models developed by the 2004 EPRI-sponsored study (EPRI, 2004) for the updated PSHA. The 2004 EPRI project reviewed the latest knowledge of CEUS ground motions. The study updated equations estimating median spectral acceleration and associated uncertainties as a function of earthquake magnitude and distance throughout the CEUS. The applicant modeled epistemic uncertainty using multiple ground motion equations with weights and multiple estimates of weighted aleatory uncertainty, which arises due to inherent randomness in data. The 2006 EPRI study found that the aleatory uncertainties were too large in EPRI (2004), thus resulting in an overestimation of seismic hazard. Therefore, the applicant used the 2004 EPRI ground motion models with the update of the 2006 EPRI aleatory uncertainty equations.

2.5.2.2.4.1.3 PSHA Sensitivity Analysis

Consistent with RG 1.208, the applicant evaluated potential impacts of new data and information in its seismic hazard calculations. The applicant provided sensitivity study results to evaluate the impacts of the proposed changes to the seismic parameters used in the PSHA calculations. These changes are categorized in four different areas: 1) selection of EPRI-SOG seismic sources near the LNP site; 2) updated source models for the Charleston, South Carolina region; 3) updated maximum magnitude distributions for the GCSZ; and 4) updated seismicity parameters for the GCSZ.

The applicant examined sources within the LNP 320-km (200-mi) site radius and sources at larger distances that could affect the site, such as Charleston, South Carolina. The sensitivity analysis assesses seismic hazard to establish any seismic source whose contribution to the total hazard exceeds 1 percent in the frequency of exceedance in the 10^{-4} and 10^{-5} range and, therefore, should be included in the hazard calculations.

The applicant concluded that the effect of both the Gulf of Mexico parameter updates and the Charleston source update resulted in an appreciable increase in the hazard. Therefore, the applicant incorporated these modifications into the updated PSHA for the LNP Units 1 and 2 site.

2.5.2.2.4.2 PSHA Methodology and Calculation

Using the updated EPRI-SOG seismic source characteristics and new ground motion models (EPRI, 2004) with updated uncertainties as inputs (EPRI, 2006), the applicant performed PSHA calculations for peak ground acceleration (PGA) and spectral acceleration at frequencies of 0.5, 1.0, 2.5, 5, 10, 25, and 100 Hertz (Hz). Following the guidance in RG 1.208, the applicant performed PSHA calculations assuming generic hard rock site conditions with a V_S of 2.8 km/s (9,200 fps).

2.5.2.2.4.3 PSHA Results

The applicant's PSHA results for the LNP Units 1 and 2 site are described in FSAR Section 2.5.2.4. The applicant performed the PSHA calculations using the EPRI-SOG seismic sources described in SER Section 2.5.2.2.2. Additionally, the applicant incorporated SNC's UCSS (2006) update of the large-magnitude Charleston, South Carolina source zone and the updates to the GCSZ, as described in SER Table 2.5.2-1. Site seismic hazard characteristics are quantified by the seismic hazard curves from the PSHA. The hazard curves were developed identifying and characterizing each seismic source that contributed to 99 percent of the seismic hazard at the LNP site. Using the hazard curves, the applicant developed UHRS, which are the spectral accelerations that have an equal likelihood of exceedance at different natural frequencies. FSAR Figures 2.5.2-226 through 2.5.2-232 illustrate the applicant's mean and 5th, 16th, 50th, 84th, and 95th fractile hard rock hazard curves for the PGA and spectral acceleration at frequencies of 25, 10, 5, 2.5, 1, and 0.5 Hz. SER Figure 2.5.2-2 shows the mean UHRS for the 10^{-3}, 10^{-4}, 10^{-5}, and 10^{-6} annual frequencies of exceedance for hard rock conditions.

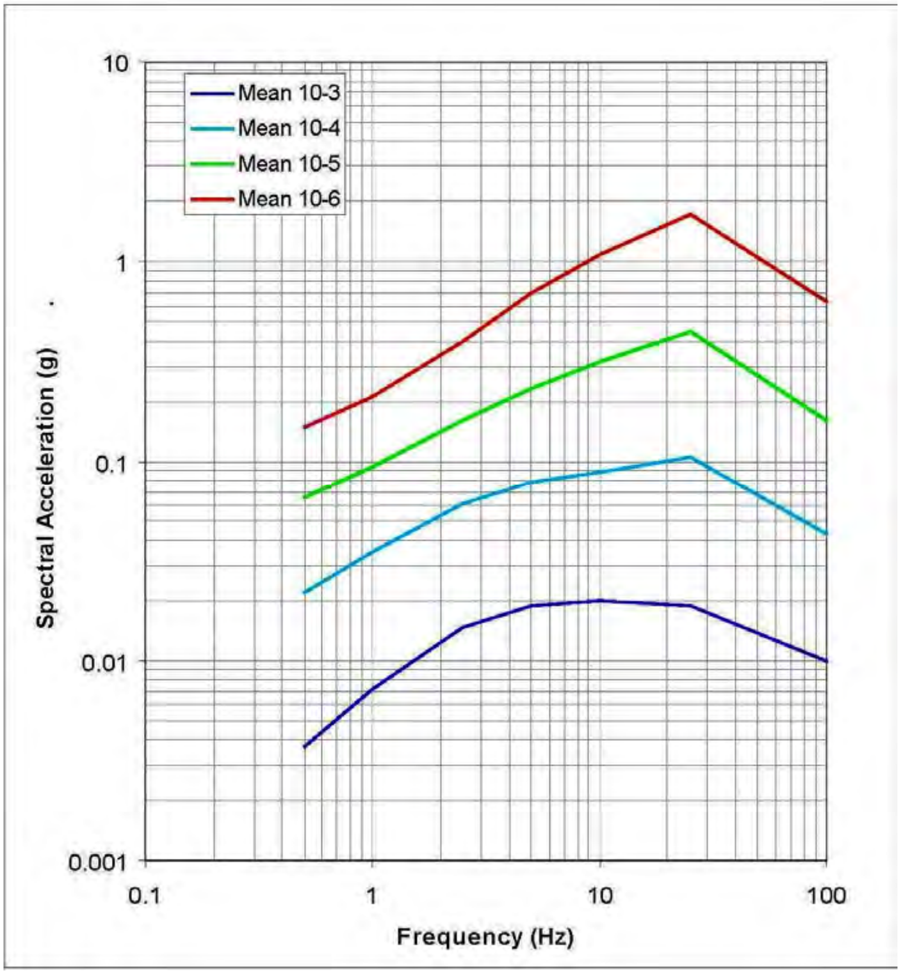


Figure 2.5.2-2. UHRS for the LNP 1 and 2 Site for Generic CEUS Hard Rock Conditions (FSAR Figure 2.5.2-238)

FSAR Section 2.5.2.4.4.2 describes the earthquake potential for the site in terms of the most likely earthquake magnitudes and source-to-site distances, which are referred to as ‘controlling earthquakes’. The applicant determined the controlling earthquakes that dominate low-frequencies (LF) and the high frequencies (HF), 1 and 2.5 Hz and 5 and 10 Hz, respectively. To determine the controlling earthquakes, the applicant deaggregated the PSHA at selected probability levels. The procedure the applicant used is outlined in RG 1.208. The applicant performed the deaggregation of the mean 10^{-3}, 10^{-4}, 10^{-5}, and 10^{-6} PSHA hazard results.

For use in the applicant’s site response analysis, which is summarized in the next SER section, the applicant developed “deaggregation earthquakes” (DE) from the controlling earthquakes. The DE parameters are listed in SER Table 2.5.2-2 and the applicant used these earthquakes to reflect the weighted distribution of earthquakes contributing to the hazard at the site. The applicant defined the weight of each DE by the relative contribution of the earthquake in a magnitude-distance domain of the total hazard. Using the EPRI median ground motions, the EPRI aleatory variability models, and the spectral shape functions from NUREG/CR-6728 CEUS ground motions, the applicant then developed smooth response spectra to represent each of the DE listed in SER Table 2.5.2-2.

Table 2.5.2-2. Deaggregation Earthquake Parameters (FSAR Table 2.5.2-221)

FREQUENCY RANGE (Hz)	MEAN ANNUAL FREQUENCY OF EXCEEDANCE	DEAGGREGATION EARTHQUAKES (DE)		
		MAGNITUDE (m_b)	DISTANCE (km [mi])	WEIGHT
1 and 2.5	10^{-4}	5.5 6.3 7.1	20.2 (12.5) 72 (45) 459 (285)	0.105 0.052 0.843
5 and 10	10^{-4}	5.4 6.2 7.1	27.7 (17.2) 70 (43) 455 (282)	0.320 0.077 0.603
1 and 2.5	10^{-5}	5.6 6.4 7.2	12.2 (7.5) 45 (28) 456 (283)	0.218 0.112 0.670
5 and 10	10^{-5}	5.4 6.3 7.2	13.6 (8.4) 29 (18) 453 (281)	0.615 0.156 0.229
1 and 2.5	10^{-6}	5.7 6.5 7.2	8.9 (5.5) 32 (20) 455 (282)	0.400 0.240 0.360
5 and 10	10^{-6}	5.4 6.4 7.2	8.9 (5.5) 15 (9.3) 450 (279)	0.681 0.297 0.022

2.5.2.2.3 Seismic Wave Transmission Characteristics of the Site

FSAR Section 2.5.2.5 describes the procedure the applicant used to assess the effects of soils on seismic wave transmission beneath the site. The hazard curves generated by the PSHA are defined for generic hard rock conditions characterized by a V_S of 2.8 km/s (9,200 fps). For the LNP Units 1 and 2 site, these hard rock conditions exist at a depth of 1,300 m (4,300 ft) beneath the ground surface, while materials with lower velocities exist in the upper 1,300 m (4,300 ft). To determine the near-surface UHRS, the applicant used Approach 2B outlined in NUREG/CR-6728). Following Approach 2B, the applicant: (1) developed soil models for the LNP Units 1 and 2 site; (2) randomized the soil profiles to account for variability; and (3) performed the final site response analysis.

In FSAR Section 2.5.2.5, the applicant described how it performed two sets of site response analyses. The applicant used one analysis to develop the site specific GMRS and the second analysis to perform the soil structure interaction (SSI) analyses. For the SSI analyses inputs, the applicant developed the performance based surface response spectra (PBSRS) and foundation input response spectra (FIRS). While the applicant described development of PBSRS and FIRS in FSAR Section 2.5.2.5, the summary and evaluation of the PBSRS, FIRS, and SSI analyses are described in SER Section 3.7.1.

2.5.2.2.3.1 Site Response Model

The applicant developed site-specific shallow V_S models for the upper 152 m (500 ft) based on the results of 18 compression (P) and shear (S) wave P-S suspension logging and downhole velocity survey wells and used four deep wells to make stratigraphic and velocity determinations to 1,676 m (5,500 ft) depth. The applicant estimated that the subsurface geology at the LNP Units 1 and 2 site consists of approximately 1,300 m (4,300 ft) of Cretaceous and Cenozoic limestone and dolomite and 1.8 m (6 ft) of Quaternary sands at the surface. The median shear wave velocity profile was added to the base of the shallow profiles to create the applicant's initial velocity profiles for site response analysis.

The applicant also estimated the parameter kappa (κ) as input into the site response analysis. Kappa is the near-surface damping parameter, which is an estimate of the dissipation of seismic energy of the site during an earthquake due to damping within soil layers and waveform scattering at layer boundaries. The applicant used two sets of modulus reduction and damping relationships to account for the potential of nonlinear behavior in the approximately 18.3 m (60 ft) of partly-to-moderately weathered limestone that occurs at a depth range of 48.8 to 67.1 m (160 to 220 ft). The remaining rock layers are assumed to behave linearly during seismic shaking.

The applicant's analysis resulted in the V_S profiles for the LNP Units 1 and 2 site illustrated in SER Figure 2.5.2-3, which were used in the applicant's GMRS analysis.

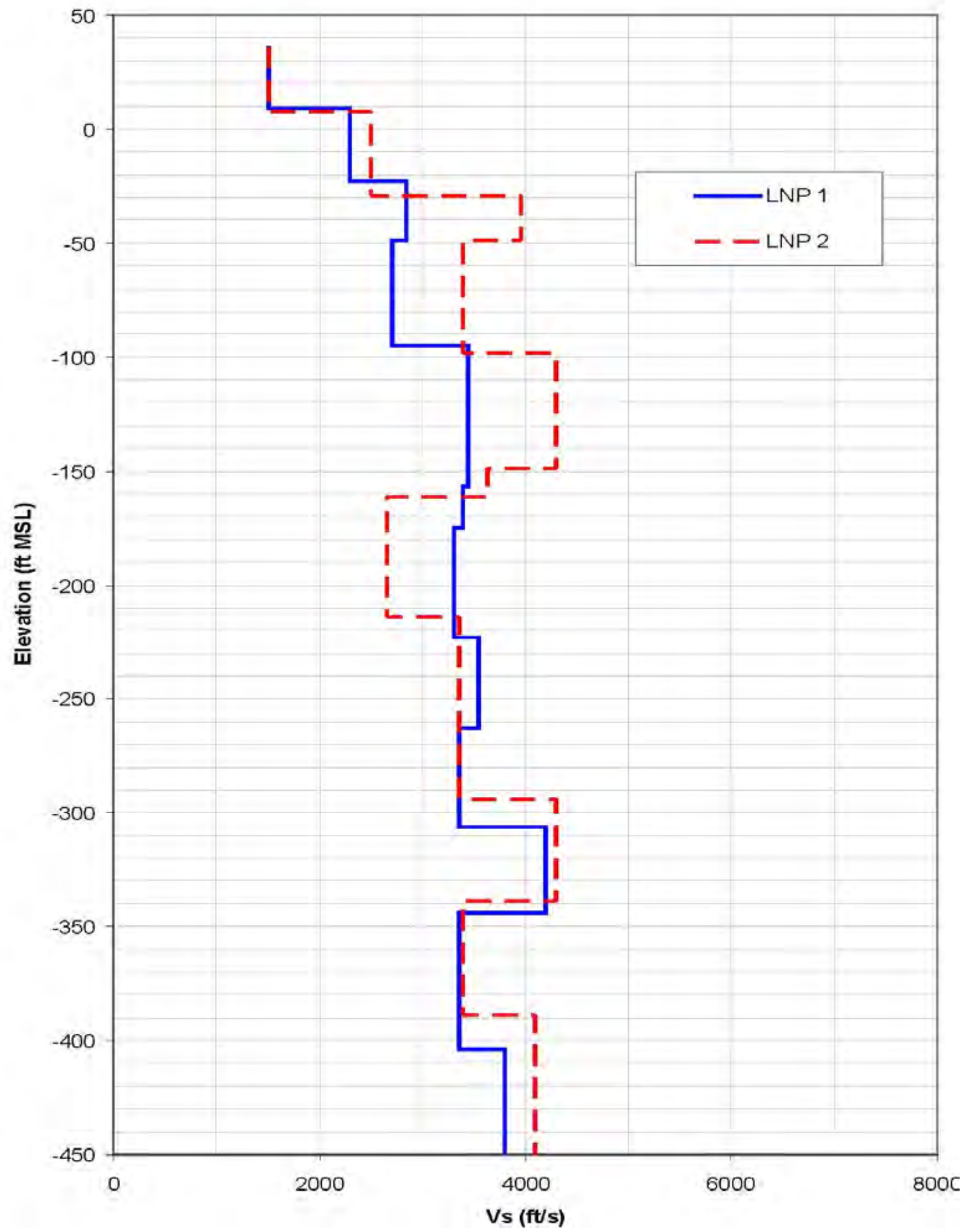


Figure 2.5.2-3. Shear Wave Velocity Profile for the LNP 1 and 2 Site Used in the GMRS Analysis
(FSAR Figure 2.5.2-254)

2.5.2.2.3.2 Site Response Methodology and Results

The applicant followed RG 1.208 and defined the site-specific GMRS at the top of the first competent layer. Since FSAR Section 2.5.4.5 states that the upper Quaternary sands have low velocity and are to be removed during construction, the reference point for the GMRS is taken to be the top of the calcareous silt unit S2, weathered limestone at an average elevation of 11 m (36 ft) using the North American Vertical Datum 1988 (NAVD88).

The applicant stated that once it determined the appropriate soil and rock dynamic properties, it modeled the variability present in the site data by randomizing the soil and rock V_S profiles, shear modulus reduction and damping values. The applicant generated 60 randomized profiles using the V_S correlation model developed by Silva et al. (1996). These artificial profiles represent the soil column from the top of bedrock to the ground surface.

The applicant developed response spectra for each controlling and deaggregation earthquake for two frequency ranges, HF (5 to 10 Hz) and LF (1 to 2.5 Hz), as defined in RG 1.208. The applicant developed 30 time histories from the sets given in NUREG/CR-6728 <u>for each deaggregation earthquake spectrum</u>. The applicant then scaled the selected time histories to match the target earthquake spectrum.

The applicant used the V_S profiles for the LNP Units 1 and 2 site as shown on SER Figure 2.5.2-3 to compute the site amplification functions for each of the spectrally matched time histories. For each hazard level (10^{-3}, 10^{-4}, 10^{-5}, and 10^{-6}) and for each controlling and deaggregation earthquake (HF and LF), the applicant paired the 60 randomized soil velocity profiles and the 60 randomized soil modulus reduction and damping curves with the 30 spectrally matched time histories. To compute the final site amplification effects, the applicant divided each output response spectrum (defined at the base of the nuclear island) by the corresponding hard rock input response spectrum and calculated the arithmetic mean of the 60 response spectral ratios.

The applicant compared the mean site amplification functions for the two GMRS profiles for the four levels of input motion (10^{-3}, 10^{-4}, 10^{-5}, and 10^{-6}). Because the comparison illustrated similarity in site amplification, the applicant used a single envelope amplification function for the LNP Units 1 and 2 sites. Those enveloped amplification functions for the four levels are plotted on SER Figure 2.5.2-4. The applicant again then enveloped and smoothed the amplification functions for the four ground motion levels, which is shown in FSAR Figure 2.5.2-280.

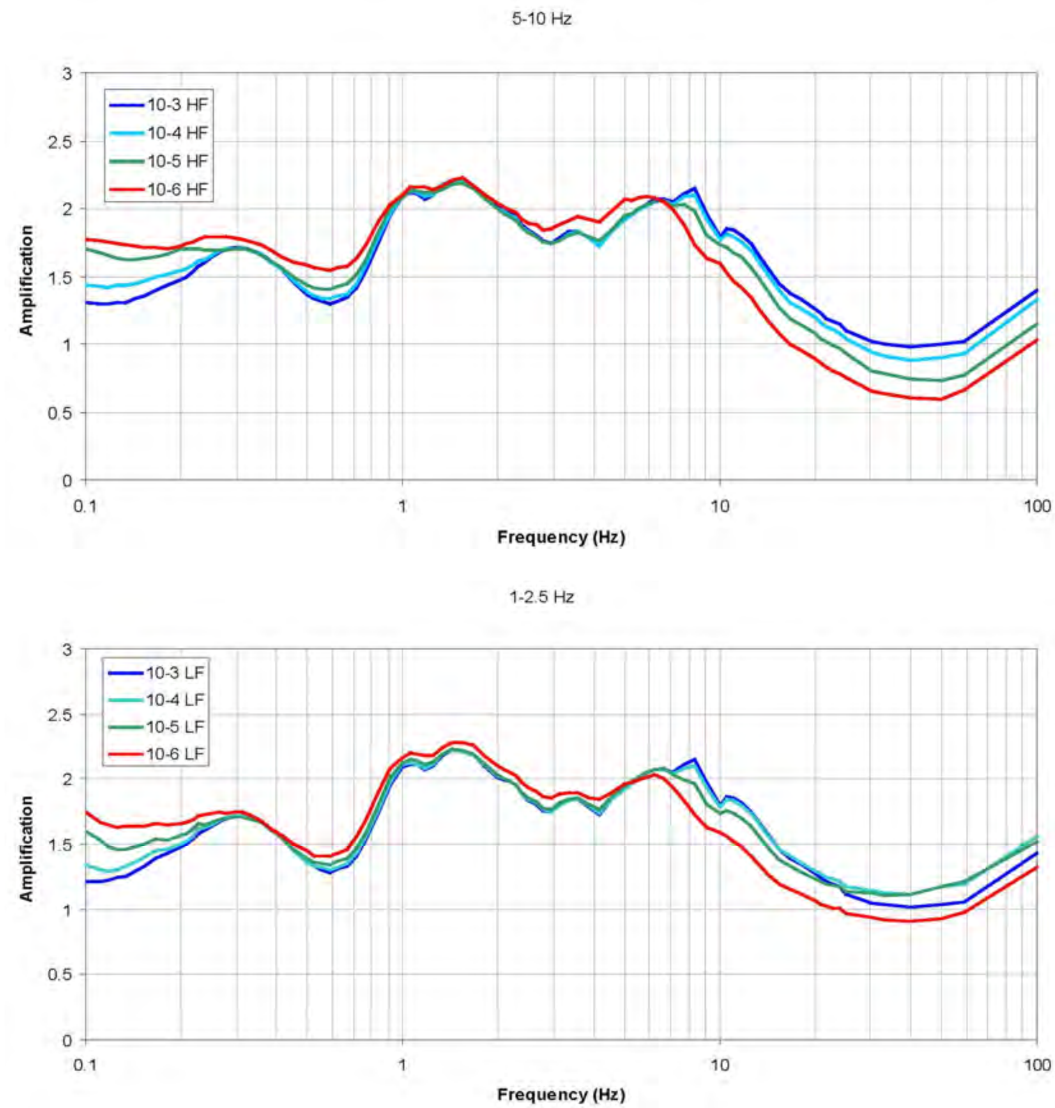


Figure 2.5.2-4. Envelope GMRS Amplification Functions for the LNP Unit 1 and 2 Site (FSAR Figure 2.5.2-278)

The applicant repeated the GMRS profile amplification function process described above for developing amplification functions at the base of the excavation creating the foundation input response spectra (FIRS). The analyses were performed including all material to design grade surface at elevation 15.5 m (51 ft.) NAVD88 and then extracting ground motion at -7.3 m (-24 m).Consistent with the GMRS analysis, a single envelope amplification function was developed for different hazard levels. The resulting FIRS amplification functions are plotted on SER Figure 2.5.2-5.

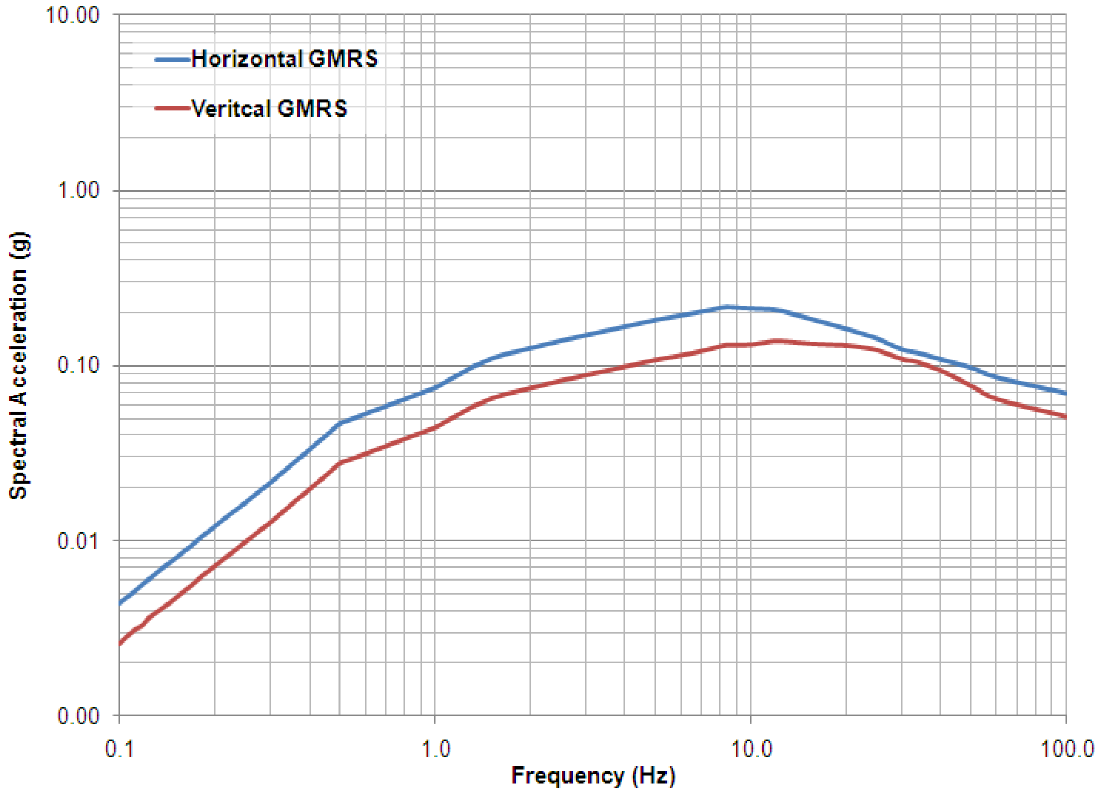


Figure 2.5.2-5. Horizontal and Vertical GMRS for the LNP Unit 1 and 2 Site (Reproduced from data in FSAR Table 2.5.2-226)

2.5.2.2.4 Ground Motion Response Spectra

FSAR Section 2.5.2.6 describes the method the applicant used 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 in American Society of Civil Engineers/Structural Engineering Institute (ASCE/SEI) Standard 43-05, "American Society of Civil Engineers, Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities." The applicant developed the GMRS by scaling the rock controlling and deaggregation earthquakes and UHRS by the site amplification functions. The site-specific GMRS is defined at the top of the first competent layer at the elevation of 11 m (36 ft). The applicant developed the vertical GMRS by applying vertical-to-horizontal (V/H) response spectral ratios, based on NUREG/CR-6728, to the horizontal GMRS.

The applicant implemented the EPRI cumulative absolute velocity (CAV) model (EPRI, 2006) in a second set of PSHA calculations for the LNP Units 1 and 2 site. The method is described in RG 1.208 and is based on the probability that earthquakes of a given magnitude can produce damaging ground motions, where the damaging ground motion is defined as CAV exceeding 0.16 g-second. The EPRI CAV model results indicate that earthquakes of moment magnitude (M) less than 5 have little probability of producing ground motions greater than 0.16 g-second. The 10^{-4} surface UHRS with CAV is zero.

2.5.2.2.4.1 Horizontal GMRS

In FSAR Section 2.5.2.6.3, the applicant developed a horizontal, site-specific, performance-based GMRS using the method described in RG 1.208 and ASCE/SEI Standard 43-05. 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 horizontal GMRS for each spectral frequency, which meets the PF, is obtained by scaling the near-surface 10^{-4} UHRS by the design factor (DF):

$$DF = \max\left(1.0,\ 0.6(A_R)^{0.8}\right) \qquad \text{Equation (2.5.2-1)}$$

In SER Equation 2.5.2-1, the amplitude ratio, A_R, is given by the ratio of the 10^{-5} UHRS and the 10^{-4} UHRS spectral accelerations for each spectral frequency. When A_R exceeds 4.2, RG 1.208 specifies that the value of the GMRS is to be no less than 45 percent of the 10^{-5} UHRS. Since the 10^{-4} UHRS with CAV is 0, this criterion is used to define the horizontal GMRS. Finally, the applicant applied a scale factor to the horizontal GMRS. As described by the applicant in FSAR Section 2.5.2.5 and the staff in SER Section 3.7.1, the applicant developed site-specific scaled FIRS. The applicant calculated a scale factor of 1.212 such that the horizontal FIRS at 100 Hz is equal to 0.1 g as required by 10 CFR Part 50, Appendix S. To be consistent with the scaled FIRS, the applicant also applied the 1.212 scale factor to the horizontal GMRS. The resulting scaled spectrum is the applicant's horizontal GMRS, shown as the blue line in SER Figure 2.5.2-5 and these values are listed in FSAR Table 2.5.2-226.

2.5.2.2.4.2 Vertical GMRS

In FSAR Section 2.5.2.6.4, the applicant obtained the vertical GMRS by deriving V/H ratios and applying them to the applicant's final horizontal GMRS. The applicant calculated rock V/H ratios using spectral ratios from NUREG/CR-6728. NUREG/CR-6728 presents categories of V/H ratios for PGA less than 0.2 g, between 0.2 g and 0.5 g, and greater than 0.5 g. The applicant used ratios for PGA < 0.2 g, for the LNP Units 1 and 2 site. Since the applicant's best estimate of kappa for the LNP Units 1 and 2 site is intermediate between the Western United States (WUS) and CEUS, the applicant developed an intermediate V/H ratio for the LNP Units 1 and 2 site. FSAR Figure 2.5.2-295 shows the V/H spectral ratios for the WUS, CEUS, and the applicant's LNP intermediate values. The LNP vertical GMRS was then computed by multiplying the horizontal GMRS by the intermediate V/H ratio. The resulting vertical GMRS is shown as the red line in SER Figure 2.5.2-5 and values are listed in FSAR Table 2.5.2-226.

2.5.2.2.5 Sensitivity Study of CEUS Seismic Source Characterization Model

In January 2012, the NRC published NUREG-2115, "Central and Eastern United States Seismic Source Characterization for Nuclear Facilities." In FSAR Section 2.5.2.7, the applicant decribes its sensitivity study using the new seismic hazard model presented in NUREG-2115 and a modified CAV filter, as described in SECY-2012-0025 Enclosure 7, Attachment 1 to Seismic Enclosure 1. The staff's summary and evaluation of FSAR Section 2.5.2.7 is located in SER Section 20.1. Based on its sensitivity study, the applicant concluded that the scaled site-specific ground motions developed using the updated EPRI-SOG model with the CAV filter presented in FSAR Section 2.5.2.6 are appropriate for use as the design basis for the LNP site.

2.5.2.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is addressed within the FSER related to the DCD.

In addition, the applicable regulatory requirements for reviewing the applicant's discussion of vibratory ground motion are as follows:

- 10 CFR 100.23 with respect 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.

In addition, the related acceptance criteria from Section 2.5.2 of NUREG-0800 are summarized as follows:

- Seismicity: To meet the requirements in 10 CFR 100.23, this section 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 identified and characterized by the Lawrence Livermore National Laboratory (LLNL) and the Electric Power Research Institute (EPRI) were used for studies in the CEUS in the past.

- Correlation of Earthquake Activity with Seismic Sources: To meet the requirements in 10 CFR 100.23, acceptance of this section 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 LLNL or EPRI 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 section, 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; RG 1.206; and RG 1.208.

2.5.2.4 *Technical Evaluation*

The NRC staff reviewed Section 2.5.2 of the LNP 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 vibratory ground motion. 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 LNP COL application.

The staff reviewed the following information in the LNP COL FSAR:

AP1000 COL Information Items

- LNP COL 2.5-2

The NRC staff reviewed LNP COL 2.5-2 related to COL Information Item 2.5-2 (COL Action Item 2.5.2-1), 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, PSHA, seismic wave transmission characteristics and the SSE ground motion. The COL information item in AP1000 DCD Section 2.5.2.1 states:

> Combined License applicants referencing the AP1000 certified design will address the following site-specific information related to the vibratory ground motion aspects of the site and region: (1) seismicity, (2) geologic and tectonic characteristics of 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; and (6) SSE ground motion.

- LNP COL 2.5-3

The NRC staff reviewed LNP COL 2.5-3 related to COL Information Item 2.5-3 (COL Action Item 2.6-2), which addresses the provision for performing site-specific evaluations, if the site-specific GMRS at foundation level exceeds 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. The COL information item in AP1000 DCD Section 2.5.2.3 states:

> The Combined License applicant may identify site-specific features and parameters that are not clearly within the guidance provided in subsection 2.5.2.1. These features and parameters may be demonstrated to be acceptable by performing site-specific seismic analyses. If the site-specific spectra at foundation level at a hard rock site or at grade for other sites exceed the certified seismic design response spectra in 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, a site-specific evaluation can be performed. These analyses may be either 2D or 3D. Results will be compared to the corresponding 2D or 3D generic analyses.

SER Section 2.5.2.4 provides the NRC staff's evaluation of the seismic, geologic, geophysical and geotechnical investigations carried out by the applicant to determine the site-specific GMRS, and 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 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 LNP COL application. To thoroughly evaluate the geologic, seismic and geophysical information the applicant presented, the staff obtained additional assistance from experts at the USGS. The staff, with its USGS advisors, made visits to the LNP Units 1 and 2 site in April and September 2009 (ML092600064 and ML093280825) to confirm interpretations, assumptions, and conclusions presented by the applicant related to potential geologic and seismic hazards. The staff's evaluation of the information the applicant presented in LNP COL FSAR Section 2.5.2 and of the applicant's responses to RAIs is presented below.

In addition to the RAIs addressing specific technical issues regarding vibratory ground motion at the LNP Units 1 and 2 site and discussed in detail below, the staff also prepared several editorial RAIs to clarify certain descriptive statements made by the applicant in the FSAR and to qualify FSAR figures and tables. These editorial RAIs are not discussed in this technical evaluation. Also, RAIs related to vibratory ground motion resolved in FSARs previously prepared for other sites in the CEUS are not discussed in detail in this technical evaluation for the LNP Units 1 and 2 site, but rather are addressed by a cross-reference to and a summary of the pertinent information used to satisfactorily resolve the issues as presented in those FSARs.

2.5.2.4.1 Seismicity

To characterize the seismic hazard for the LNP Units 1 and 2 site, the applicant followed the methodology provided in RG 1.208 and used the EPRI-SOG seismic hazard models

(EPRI-SOG, 1986), developed in the late1980s, as a starting point. The EPRI-SOG study used an earthquake catalog compiled through 1984 that covers the CEUS. FSAR Section 2.5.2.1 describes the applicant's update of the original EPRI-SOG earthquake catalog to extend it from 1985 through December 2006 and also to extend the coverage to include the portions of the Gulf of Mexico that were not covered in the original EPRI-SOG catalog.

2.5.2.4.1.1 EPRI-SOG Seismicity Catalog Updates

The staff focused its review of FSAR Section 2.5.2.1 on the adequacy of the applicant's description of the historical record of earthquakes. To update the EPRI-SOG earthquake catalog for the region surrounding the LNP Units 1 and 2 site, the applicant evaluated several different earthquake catalogs, including the ANSS, ISC, and NEIC catalogs.

2.5.2.4.1.2 Gulf of Mexico Seismicity

Because the EPRI-SOG earthquake catalog did not include events from the Gulf of Mexico except along its immediate coast, the applicant extended the coverage of its catalog to include seismicity within the Gulf of Mexico between latitude 24°N to 32°N and longitude 100°W to 83°W. The applicant's update was prompted in large part by two recent, moderate-magnitude seismic events in the Gulf. These events were the m_b 4.9 event that occurred on February 10, 2006, offshore of the Louisiana coast and the m_b 6.0 event that occurred on September 10, 2006, offshore of the Florida coast.

In FSAR Section 2.5.2.3, the applicant noted that due to the use of different magnitude conversion relationships its estimated m_b for the September 10, 2006 event, m_b 6.08, differs from that reported in the COL application submitted for STP Units 3 and 4, which gives m_b 6.11. In RAI 2.5.2-5, the staff asked the applicant to clarify the difference in the magnitude for the September 10, 2006, event as well as the different magnitude conversion relationships used in the STP Units 3 and 4 COL application (STPNOC, 2008) in comparison to those used in the LNP COL application. In response to RAI 2.5.2-5, the applicant explained that both LNP and STP averaged the output of three moment magnitude (M) and m_b relationships to calculate estimated m_b. Two relationships used are the same in both the LNP and STP COL applications. The third relationship differs. The applicant explained that STP used an earlier version of this relationship, while LNP utilized the final version of the conversion relationship. The staff reviewed the two different m_b estimates and finds that the difference in estimated m_b of 6.08 for LNP and 6.11 for STP is not significant. Both the 6.08 and 6.11 estimates are conservative since the value of directly measured m_b presented in the ANSS catalog is 5.8. Furthermore, the staff concludes that the slight difference in m_b for the September 10, 2006 Gulf earthquake does not affect the seismic hazard analysis at the LNP site. Therefore, the staff considers RAI 2.5.2-5 resolved.

2.5.2.4.1.3 Staff Conclusions Regarding Seismicity

Based upon its review of FSAR Section 2.5.2.1 and RAI 2.5.2-5, the staff concludes that the applicant developed a complete and accurate earthquake catalog for the region surrounding the LNP Units 1 and 2 site, including the Gulf of Mexico seismicity. The staff concludes that the seismicity catalog as described by the applicant in 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(a)(1)(iii) and 10 CFR 100.23.

2.5.2.4.2 Geologic and Tectonic Characteristics of the Site and Region

This SER section provides the staff's evaluation of the seismic source models the applicant used as part of its PSHA for the LNP Units 1 and 2 site. FSAR Section 2.5.2.2 describes the seismic sources from the original EPRI-SOG seismic source models (EPRI-SOG, 1986) that contribute to 99 percent of the total hazard at the CR3, located 15 km (9 mi) from the LNP Units 1 and 2 site. These seismic source models were developed in 1986 by the six EPRI-SOG ESTs. FSAR Section 2.5.2.4 describes the applicant's sensitivity studies to determine if the 1986 EPRI-SOG seismic source models needed updating based on more recent studies in the geologic and seismic literature cited by the applicant. Consistent with RG 1.208, the applicant evaluated more recent seismic hazard studies and data available for the region surrounding the site for comparison to the 1986 EPRI-SOG seismic source models. As a result of this evaluation, the applicant updated several of the original source models developed by the EPRI-SOG ESTs.

The staff's review of the application of the updated seismic source model to the hazard calculation at the LNP Units 1 and 2 site is discussed in SER Section 2.5.2.4.4.

2.5.2.4.2.1 Original EPRI-SOG Seismic Sources

In FSAR Section 2.5.2.4.3.1, the applicant describes its selection of EPRI-SOG seismic sources and stated that one relationship was used to convert m_b to M. Later in FSAR Section 2.5.2.4.2.3, the applicant presents three relationships used to convert m_b to M to use in its hazard analysis. In RAI 2.5.2-15, the staff asked the applicant to clarify why only one conversion relationship was used to select the EPRI-SOG seismic source zones and not three relationships, like those used in the hazard analysis. In response to RAI 2.5.2-15, the applicant explained that the hazard results are not very sensitive to the use of the alternative m_b to M relationships, as illustrated in FSAR Figure 2.5.2-236. Therefore, the applicant thought it was sufficient to use one relationship for the purpose of identifying the appropriate set of EPRI-SOG seismic sources. After reviewing the issues, the staff concludes that since the single relationship was used for the purpose of identification of EPRI-SOG sources only, and not for final hazard calculation where the applicant used the weighted average of the three formulas, this does not compromise the GMRS and hazard calculations. Therefore, the staff finds the applicant's response acceptable and considers RAI 2.5.2-15 resolved.

The three m_b to M conversion relationships the applicant presented in FSAR Section 2.5.2.4.2.3 are important to the applicant's hazard analyses because the magnitudes in the earthquake catalogs are in m_b whereas the ground motion prediction equations use M. However, the m_b scale saturates at m_b of 7, but the conversion relations go beyond m_b of 7. In RAI 2.5.2-19, the staff asked the applicant to clarify how it dealt with the issue of m_b saturation when performing magnitude conversion to use in its hazard analyses. In response to RAI 2.5.2-19, the applicant explained that the m_b to M conversion saturation has the most impact for the LNP Units 1 and 2 site in the characterization of the Charleston source, since that source is the only source affecting the LNP site with an m_b greater than 7. For the Charleston source zone, the repeated large earthquakes are initially characterized in terms of M. These M estimates were used directly to calculate the ground motions and hazard from this source, so that the m_b to M conversion was not necessary. Based on its review of the applicant's RAI response, the staff concludes that the saturation aspect of the m_b to M conversion relations has no material effect on the hazard analyses at the LNP site. This is due to the applicant's use of direct estimates of M for the Charleston source for which the m_b to M conversion relations were not used. The staff considers RAI 2.5.2-19 resolved.

2.5.2.4.2.2 Update of EPRI-SOG Seismic Source Models

FSAR Section 2.5.2.2.2 describes four PSHA studies that were completed after the 1989 EPRI PSHA and which involved the characterization of seismic sources within the LNP Units 1 and 2 site region. FSAR Sections 2.5.2.4.1 through 2.5.2.4.3 present the applicant's discussion and sensitivity analyses determining whether the 1986 EPRI-SOG seismic source models needed to be updated based on more recent seismic hazard studies or on new seismicity data for the region surrounding the LNP Units 1 and 2 site. The four PSHA studies that were completed after the 1989 EPRI PSHA include the USGS National Seismic Hazard Mapping Project (Frankel et al. 1996, 2002), the South Carolina Department of Transportation (SCDOT) seismic hazard mapping project (SCDOT, 2003), the LLNL Trial Implementation Program study (NUREG/CR-6607, Savy, et al., 2002), and the updated PSHA for the VEGP plant site (SNC, 2006). The applicant provided a description of these four models in FSAR Section 2.5.2.2.2, as well as a comparison of these more recent studies with the EPRI source PSHA models.

2.5.2.4.2.2.1 Update of the Charleston Seismic Source

The applicant updated the EPRI-SOG Charleston seismic source models with a model that was originally presented in the Site Safety Analysis Report (SSAR) for the VEGP ESP site (SNC, 2007). This update was based on the results of several post-EPRI PSHA studies (Frankel et al. 2002; Chapman and Talwani 2002) and the availability of paleoliquefaction data (Talwani and Schaeffer 2001). The applicant updated the EPRI characterization of the Charleston seismic source zone as part of the COL application. The applicant used the UCSS model to update the Charleston seismic source. The SSAR for the VEGP ESP Site (SNC, 2007) provides the details of the UCSS model and the SER for the VEGP ESP (NUREG-1923, 2009) describes the NRC staff's review of the UCSS. The UCSS model development followed the guidelines provided in RG 1.208 and used a Senior Seismic Hazard Analysis Committee (SSHAC) Level 2 (NUREG/CR-6372, "Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and User of Experts") expert elicitation method to incorporate current literature cited by the applicant and data and the understanding of experts into an update of the Charleston seismic source model. The staff reviewed and approved the UCSS model as part of its review of the VEGP ESP application (NUREG-1923).

2.5.2.4.2.2.2 Gulf Coast Source Zones

Based on the geological and seismological data published since the 1986 EPRI-SOG seismic source model, the applicant evaluated whether the maximum magnitudes for the EPRI-SOG sources needed updating. As a result of two 2006 Gulf of Mexico earthquakes, the applicant updated the EPRI-SOG GCSZ.

The applicant updated five of the six EST GCSZ M_{max} distributions due to the occurrence of the February 10, 2006, m_b 4.9 earthquake and the September 10, 2006, m_b 6.0 earthquake in the Gulf of Mexico. The magnitudes of these two earthquakes exceeded, in some cases, the upper- and lower-bounds of the original EPRI GCSZ M_{max} distributions. To perform this update, the applicant implemented the GCSZ updates described in the STP Units 3 and 4 COL application (STPNOC, 2008). To determine what updates to make, STP performed a SSHAC Level 2 expert elicitation study (SSHAC, 1997). The purpose of the SSHAC process was to integrate expert opinion and to capture the center, body, and range of the scientific community's opinion on updating the EST GCSZ.

In RAI 2.5.2-16 and RAI 2.5.2-22, the staff asked the applicant to thoroughly describe the details of its GCSZ update, to provide further justification for the updated parameters, and to explain how the updated source models adequately characterize the seismic hazard of the Gulf Coast. In response, the applicant stated that not all the geometries of the six EST GCSZ encompass the locations of these two 2006 Gulf of Mexico earthquakes and therefore only the M_{max} distributions of sources that do encompass the earthquakes were updated. Additionally, the applicant proposed changes for a later version of FSAR Section 2.5.2, which describes in more detail the applicant's GCSZ update. For the reasons discussed in the following paragraphs, the staff finds these proposed FSAR changes acceptable. The staff is tracking these changes to the FSAR as **Confirmatory Item 2.5.2-1**.

Resolution of Confirmatory Item 2.5.2-1

Confirmatory Item 2.5.2-1 is an applicant commitment to update Section 2.5.2 of its FSAR. The staff verified that LNP COL FSAR Section 2.5.2 was appropriately updated. As a result, Confirmatory Item 2.5.2-1 is now closed.

STPNOC (2008) preformed a SSHAC Level 2 expert elicitation study (NUREG/CR-6372) to determine what updates to make to the GCSZ. The SSHAC Technical Integration (TI) team's original recommendation was for a M_{max} distribution with m_b and weights of 6.1 [0.1], 6.6 [0.4], 6.9 [0.4], and 7.2 [0.1]. However, the SSHAC Peer Review Panel (PRP) did not approve this M_{max} distribution. Instead, the SSHAC PRP recommended that the individual M_{max} distributions for five of the six ESTs GCSZ be updated. The applicant implemented the PRP M_{max} distribution in its update. As part of RAI 2.5.2-22, the staff asked the applicant to provide justification for not adopting the original TI team's M_{max} distribution. To address this, the applicant conducted a sensitivity analysis in which M_{max} distributions for the three GCSZ that encompass the September 10, 2006, earthquake were replaced with the TI team's original M_{max} distribution of: 6.1 [0.1], 6.6 [0.4], 6.9 [0.4], 7.2 [0.1]. SER Table 2.5.2-3 shows the resulting percent change in site ground motions at various spectral frequencies.

Table 2.5.2-3. Percent Change in LNP Site Ground Motions at Finished Grade Elevation Resulting from the Use of a M_{max} Distribution of 6.1 [0.1], 6.6 [0.4], 6.9 [0.4], 7.2 [0.1] (RAI 02.05.02-22 Table 1)

SPECTRUAL FREQUENCY (Hz)	PERCENT CHANGE IN LNP SITE GROUND MOTIONS AT FINISHED GRADE ELEVATION
0.5	+2
1.0	+4
2.5	+4
5.0	+6
10.0	+6
25.0	+7
100.0	+7

After reviewing the applicant's sensitivity study, the staff concludes that the updated M_{max} parameters adequately characterize the seismic hazard of the Gulf Coast region. The percent change results from the sensitivity analysis show that the higher M_{max} distribution originally recommended by the TI team does not greatly increase the seismic hazard at the LNP Units 1 and 2 site relative to the M_{max} distributions used by the applicant. Based on the modest size of the two 2006 Gulf earthquakes (m_b 4.9 and 6.0) and their distances from the site (758 and 498 km (471 mi and 309 mi)) the staff concludes that the applicants' updated GCSZ models adequately characterize the potential hazard.

RAIs 2.5.2-16 and 2.5.2-22 address the applicant's update of the EST GCSZ M_{max} distributions. The one source zone that the applicant did not update is the Woodward-Clyde Consultant background source model (WCC-B43). The staff asked the applicant to justify not updating that particular source and to describe how the source sufficiently characterizes the hazard for the Gulf. In response, the applicant described that the WCC-B43 background source is characterized by a 2-by-2 degree latitude-longitude zone centered near the LNP site and that neither recent Gulf of Mexico earthquake occurred within the zone. Additionally, the applicant compared the tectonic setting and type of crust of the WCC-B43 zone and that of the locations of the recent Gulf of Mexico earthquakes. The applicant demonstrated that the WCC-B43 zone primarily encloses the stable continental crust of the Florida platform, while the recent Gulf earthquakes occurred within transitional or oceanic crust (Johnston et al., 1994; Sawyer et al., 1991). The staff reviewed the tectonic and topographic maps the applicant provided in response to RAI 2.5.2-22 (RAI 2.5.2-22, Figures 2, 3, and 4). The staff concludes that, because of its placement and size about the LNP site, the WCC-B43 source zone was intended to characterize seismicity local to the site and not to characterize the entire Gulf Coast region. Additionally, the staff concludes from review of the tectonic and topographic maps that recent Gulf of Mexico earthquakes occurred in a type of crust different than the WCC-B43 zone characterizes. Finally, because the WCC-B43 zone characterizes seismicity locally about the LNP site in a crustal environment distinct from that of the recent Gulf events, the staff concludes that EPRI-SOG WCC-B43 background source model for the LNP Units 1 and 2 site does not need to be updated due to recent earthquakes in the Gulf of Mexico.

After reviewing the applicant's responses to RAIs 2.5.2-16 and 2.5.2-22, the staff concludes that the applicant justified its M_{max} parameters characterizing the seismic hazard of the Gulf Coast region, and that the EPRI-SOG WCC background source model for the LNP Units 1 and 2 site is not meant to characterize Gulf of Mexico seismicity and, therefore, does not need to be updated due to recent earthquakes in the Gulf of Mexico. The applicant's sensitivity study shows that the updated M_{max} parameters adequately characterize the seismic hazard of the Gulf Coast region. For these reasons, the staff considers RAIs 2.5.2-16 and 2.5.2-22 resolved.

2.5.2.4.2.2.3 Source Zones Outside of the Site Region

In accordance with RG 1.208, the applicant must expand the area of investigation beyond the site region if capable seismic source zones outside the site region are identified that produce large-magnitude earthquakes.

The New Madrid Seismic Zone (NMSZ), which extends from Missouri to Tennessee, is considered a major seismic zone in the CEUS. The NMSZ produced a series of large-magnitude earthquakes between December 1811 and February 1812. Paleoliquefaction studies in the region of the 1811-12 New Madrid earthquakes have identified several sequences of pre-historic earthquakes that have led researchers to estimate a mean recurrence interval for large NMSZ earthquakes of approximately 500 years. The applicant did not provide a discussion of the NMSZ's potential contribution to the seismic hazard at the LNP site. In RAI 2.5.2-18, the staff asked the applicant to discuss the significance of the NMSZ to the LNP Units 1 and 2 site and to provide justification for not including this source in the LNP PSHA.

In response to RAI 2.5.2-18, the applicant provided the staff with its evaluation results of the effect of NMSZ to the hazard at the LNP. SER Figure 2.5.2-6 compares the 2-second mean spectral acceleration hazard of repeated large-magnitude earthquakes for the NMSZ to that computed for earthquakes in the region of Charleston, South Carolina. The Charleston source was included in the LNP PSHA. SER Figure 2.5.2-6 illustrates that the mean hazard from the NMSZ is less than 1 percent of the hazard from the Charleston source for the 2-second spectral acceleration. The NMSZ is a distant source zone (> 1000 km (> 620 mi)) from the LNP Units 1 and 2 site. The effect of a large-magnitude earthquake on the site at such distances would be greatest at low frequencies, for example at 0.5 Hz equivalent to the 2-second period used by the applicant in SER Figure 2.5.2-6. Since the hazard of the NMSZ at the LNP Units 1 and 2 site is less than 1 percent of the Charleston source at low frequencies, the NMSZ contribution to the total hazard at the LNP Units 1 and 2 site will be less than that shown in SER Figure 2.5.2-6. Therefore, the NMSZ is not a significant contributor to the seismic hazard at the LNP Units 1 and 2 site. Based on the results of the applicant's testing of the NMSZ, the staff concludes that the NMSZ does not contribute significantly to the hazard at the LNP Units 1 and 2 site and, therefore, does not need to be included in the LNP PSHA. The staff considers RAI 2.5.2-18 resolved.

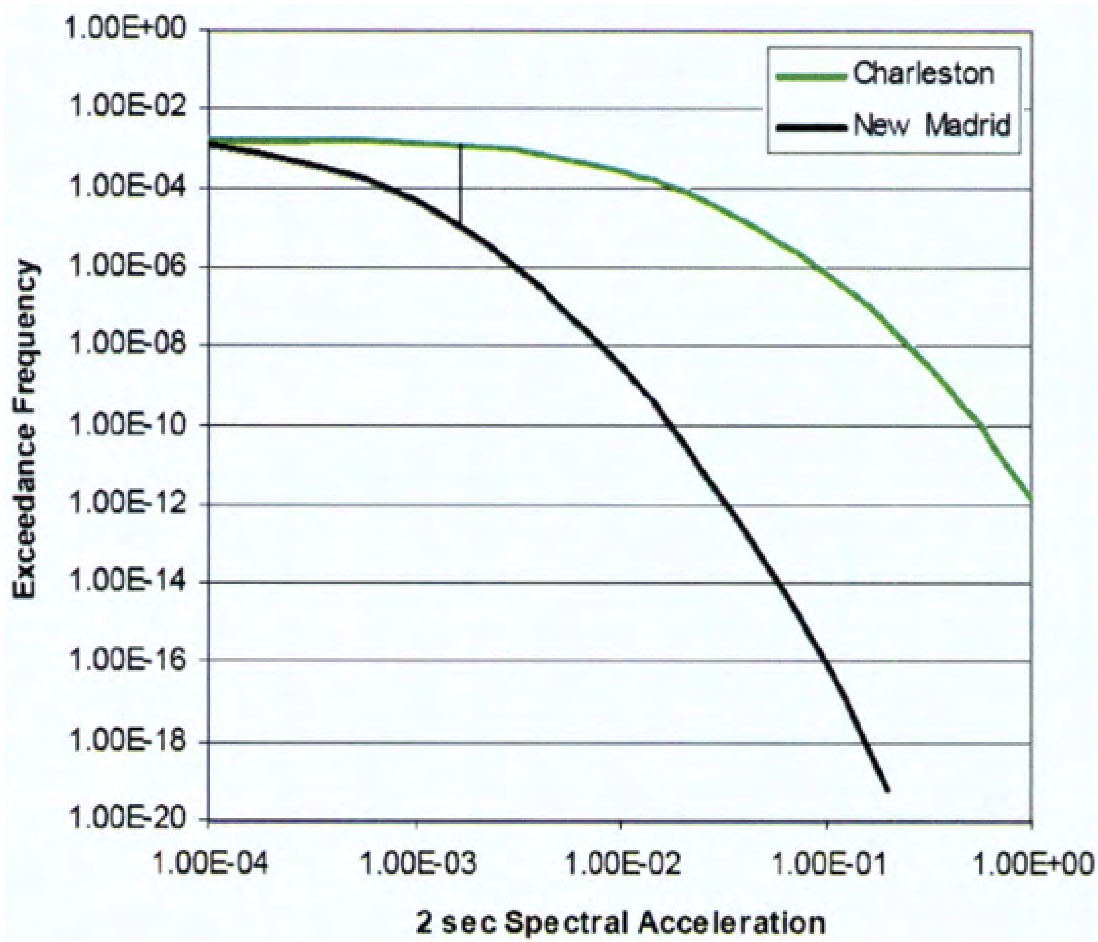


Figure 2.5.2-6. Mean Hazard Curves for the NMSZ (New Madrid) and Charleston Sources of Repeated Large-magnitude Earthquakes (RAI 02.05.02-18 Figure 1)

2.5.2.4.2.3 Staff Conclusions of the Geologic and Tectonic Characteristics of the Site and Region

Based upon its review of LNP COL FSAR Sections 2.5.2.2 and 2.5.2.4, the staff concludes that the applicant adequately updated the original EPRI-SOG seismic source models as the input to its PSHA for the LNP Units 1 and 2 site. In addition, the staff concludes that the applicant adequately considered seismic sources that were not part of the EPRI-SOG sources for the LNP Units 1 and 2 site, such as the NMSZ and the updated GCSZ. The staff concludes that the applicant's use of EPRI-SOG seismic source models in addition to the updates of the model, as described by the applicant in FSAR Section 2.5.2.2 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(a)(1)(iii) and 10 CFR 100.23.

2.5.2.4.3 Correlation of Earthquake Activity with Seismic Sources

FSAR Section 2.5.2.3 describes the correlation of updated seismicity with the EPRI-SOG seismic source model. The applicant compared the distribution of earthquake epicenters from both the original EPRI-SOG historical catalog (1627 to 1984) and the updated earthquake catalog (1985 to 2006) with the seismic sources characterized by the 1986 EPRI-SOG Project. The applicant concluded that there are no new earthquakes within the site region that can be associated with a known geologic structure and that there are no clusters of seismicity suggesting a new seismic source not captured by the EPRI-SOG seismic source model. The applicant also concluded that the updated catalog does not show a pattern of seismicity that would require significant revision to the geometry of any of the EPRI-SOG seismic sources.

In its review, the staff evaluated the completeness of the applicant's updated earthquake catalog and the applicant's subsequent conclusions by comparing the applicant's earthquake catalog to a compilation catalog derived from USGS seismicity catalogs. The USGS seismicity catalog from February 1985 to December 2006 is shown in SER Figure 2.5.2-7 as the yellow circles. The applicant's updated seismicity catalog is illustrated by the red circles, which covers February 1985 to December 2006. The comparison of these datasets illustrates that the applicant's updated earthquake catalog adequately characterizes the seismicity within and around the LNP Units 1 and 2 site region. The blue circles in SER Figure 2.5.2-7 illustrate the seismicity from the USGS catalog covering December 2006 to June 2010. This recent seismicity does not show any significant deviations from the applicant's seismicity catalogs. Based on the spatial distribution of earthquakes in the applicant's updated catalog and the staff's independent review of the USGS seismicity catalog through April 2010, the staff concludes that revisions to the existing EPRI-SOG source geometries are not warranted.

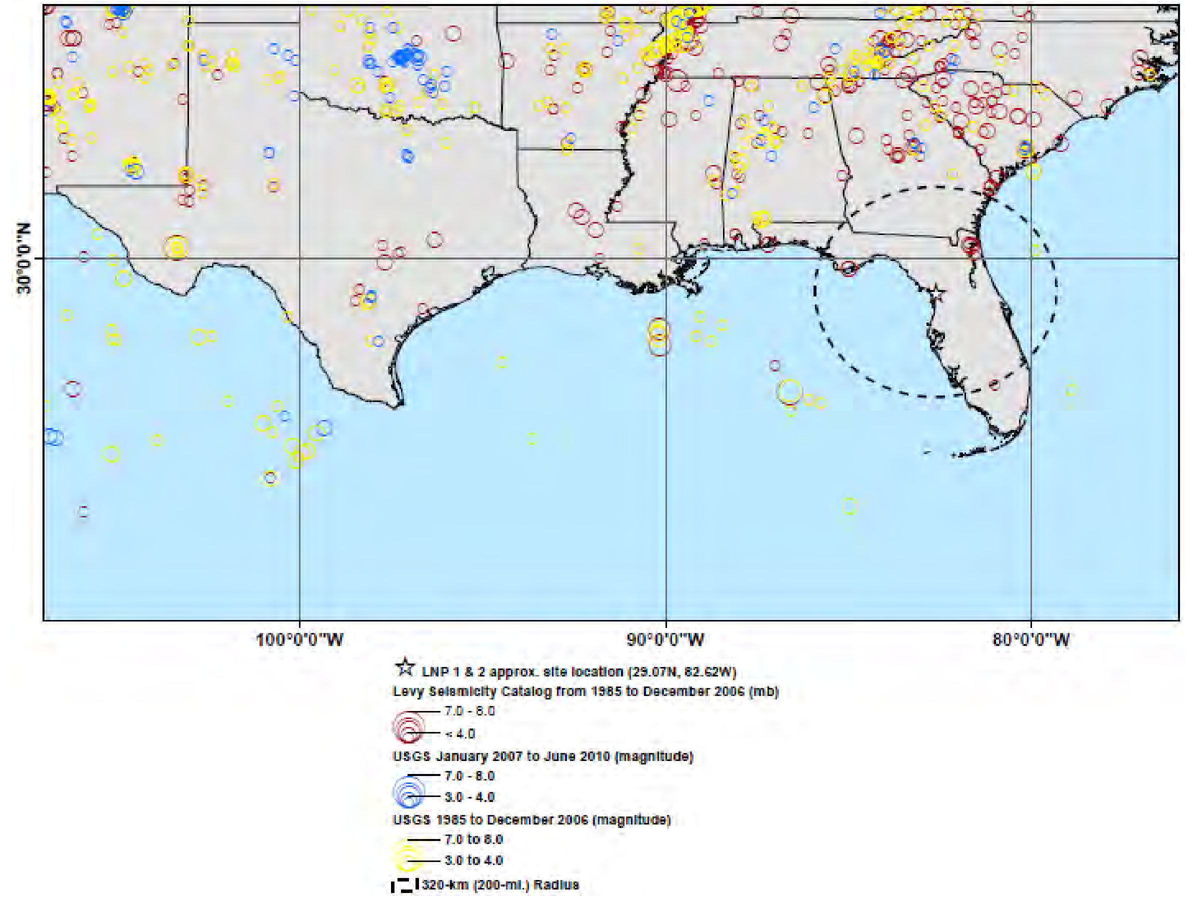


Figure 2.5.2-7. A Comparison of Events ($m_b \geq 3$) from the LNP Unit 1 and 2 Site Earthquake Catalog from 1985 to 2006 (Red Circles), the USGS Earthquake Catalog from 1985 to 2006 (Yellow Circles), and the USGS Earthquake Catalog from 2007 to 2010 (Blue Circles)

The star corresponds to the location of the LNP Unit 1 and 2 site and the dashed black oval corresponds to the 320-km (200-mi) site radius.

2.5.2.4.4 Probabilistic Seismic Hazard Analysis and Controlling Earthquakes

The staff focused its review of FSAR Section 2.5.2.4 on the applicant's updated PSHA and the LNP Units 1 and 2 site controlling earthquakes determined by the applicant after completion of its PSHA. The staff's review of the applicant's update of the EPRI-SOG seismic source model is described in SER Section 2.5.2.4.2, therefore this SER section focuses on the review of the application of the updated seismic source model to the hazard calculation at the LNP Units 1 and 2 site.

2.5.2.4.4.1 PSHA Calculation

The applicant performed PSHA calculations for PGA and spectral acceleration at frequencies of 0.5, 1.0, 2.5, 5, 10, 25, and 100 Hz. Following the guidance provided in RG 1.208, the PSHA calculations were performed assuming generic hard rock site conditions with V_S of 2.8 km/s (9,200 fps). The actual local site characteristics are incorporated in the calculation of the SSE spectrum, which uses the hard rock PSHA hazard results as the starting point.

2.5.2.4.4.2 Controlling and Deaggregation Earthquakes

FSAR Section 2.5.2.4.4.2 describes the deaggregation of final PSHA hazard curves to determine the controlling earthquakes for the LNP Units 1 and 2 site. To determine the LF and HF controlling earthquakes, the applicant followed the procedure outlined in RG 1.208. This procedure specifies that controlling earthquakes are determined from the deaggregation of the PSHA results corresponding to annual frequencies of 10^{-4}, 10^{-5}, and 10^{-6} and are based on the magnitude and distance values that contribute most to the hazard at the average of 1 and 2.5 Hz for LF and the average of 5 and 10 Hz for HF. The LF controlling earthquake for the site often represents a large distant source, while the HF controlling earthquake often corresponds to a smaller, local earthquake.

For the CR3 site, the HF controlling earthquake is M 5.3 at a distance of 17 km (10.5 mi). In RAI 2.5.2-7, the staff asked the applicant to explain the absence of a similar local, moderate-magnitude HF controlling earthquake for the LNP Units 1 and 2 site. In response to RAI 2.5.2-7, the applicant explained that the CR3 site seismic hazard analysis did not include the updated Charleston seismic source that produces large-magnitude earthquakes with a recurrence period of 500 years. Updating the Charleston source changed the contributions to the hazard, such that Charleston-type events are the major contributor to the HF hazard. The applicant also determined a weighted distribution of controlling earthquakes, which are called deaggregation earthquakes. As described in NUREG/CR-6728, deaggregation earthquakes separately address the contribution of nearby, intermediate, and distance events. SER Table 2.5.2-2 lists the LF and HF 10^{-4}, 10^{-5}, and 10^{-6} deaggregation earthquakes for the site and their associated weights. The deaggregation earthquakes include a nearby, or local, moderate-magnitude event as a contributor to the hazard. Since the Charleston source update resulted in changing the seismic source contributors, the staff concludes that a controlling earthquake similar to the CR3 site HF controlling earthquake [M 5.3, distance 17 km (10.5 mi)] is not necessary to characterize the hazard at the LNP Units 1 and 2 site. Additionally, the applicant's calculation of deaggregation earthquakes, following the procedure outlined in Appendix D of RG 1.208, accurately determined the significant contributing events. Therefore, the staff concludes that the applicant adequately determined the LNP Units 1 and 2 site controlling and deaggregation earthquakes.

As described in FSAR Section 2.5.2.4.4.3, the applicant then used the updated ground motions discussed in SER Section 2.5.2.2.4, aleatory variability models, and the spectral shape functions of NUREG/CR-6728's CEUS ground motions to develop response spectra to represent each of the controlling and deaggregation earthquakes. When assessing the uncertainty that arises due to inherent randomness in data, the aleatory variability, for the spectral frequencies between 0.1 and 100 Hz, the applicant used a combination of relationships from a number of references. In RAI 2.5.2-8, the staff asked the applicant to identify the sources of relations it used and to illustrate the dependence between the aleatory variability and frequencies that the applicant adopted. In its response, the applicant provided the justification of relations used as illustrated in SER Figure 2.5.2-8. The figure illustrates the relationship between aleatory variability (increase in Sigma) and frequency (or Period). The applicant provided the requested information; therefore, the staff considers RAI 2.5.2-8 resolved.

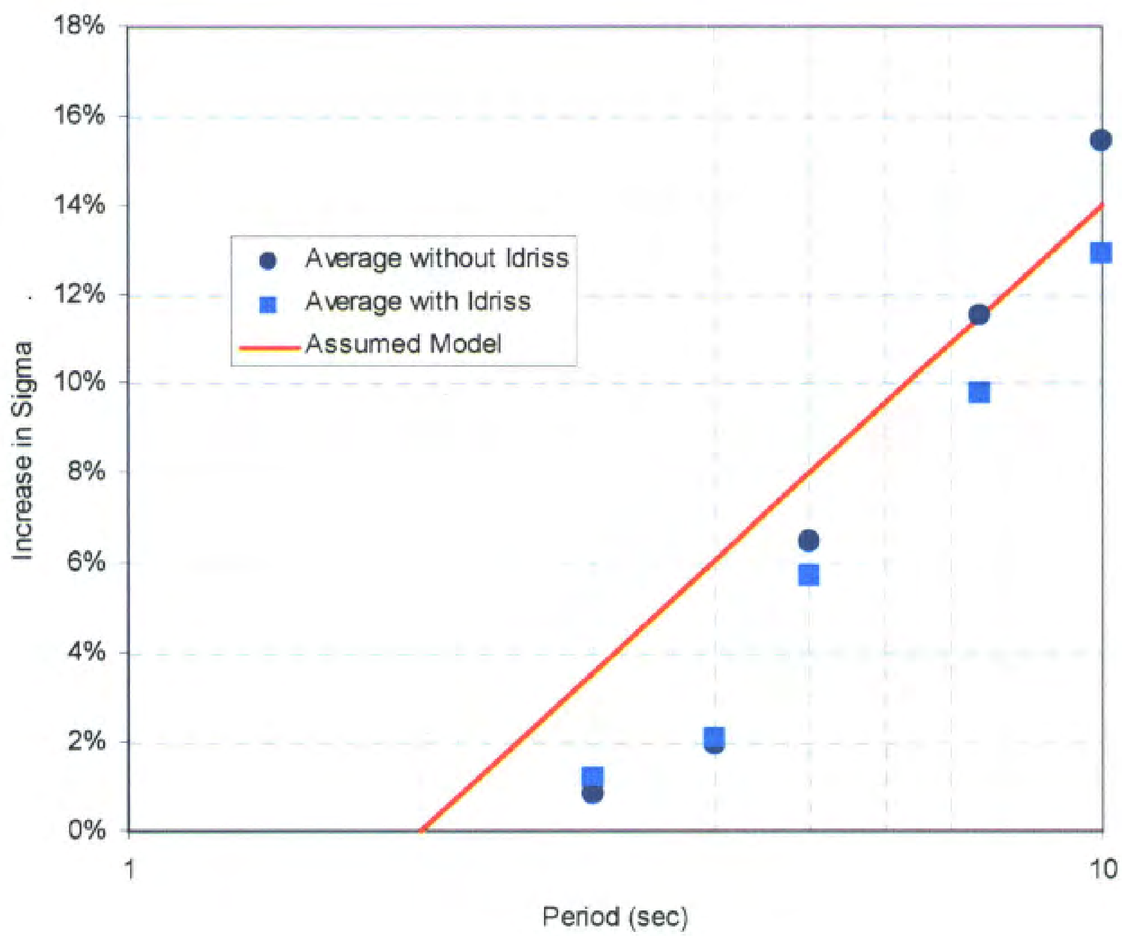


Figure 2.5.2-8. Increase in Aleatory Variability for Periods Longer than 2.0 Seconds Based on the PEER-NGA Ground Motions (RAI 02.05.02-08 Figure 1)

2.5.2.4.4.3 Staff Conclusions Regarding PSHA and Controlling and Deaggregation Earthquakes

After review of the applicant's PSHA and controlling and deaggregation earthquake determination and the applicant's responses to RAIs 2.5.2-7 and 2.5.2-8, the staff concludes that the applicant's PSHA adequately characterizes the seismic hazard for the region surrounding the LNP Units 1 and 2 site, 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 and that the PSHA and controlling and deaggregation earthquakes determination meets the requirements of 10 CFR 52.79(a)(1)(iii) and 10 CFR 100.23.

2.5.2.4.5 Seismic Wave Transmission Characteristics of the Site

FSAR Section 2.5.2.5 describes the method the applicant used to develop the LNP Units 1 and 2 site free-field ground motion spectra. The seismic hazard curves generated by the applicant's PSHA are defined for generic hard rock conditions (characterized by a V_S of 2.8 km/s (9,200 fps). According to the applicant, these hard rock conditions exist at a depth of 1,300 m (4,300 ft) beneath the ground surface at the LNP Units 1 and 2 site. To determine the site free-field ground motion, the applicant performed a site response analysis. The output of the applicant's site response analysis is the site amplification functions, which are used to determine the site-specific soil UHRS for the 10^{-4} and 10^{-5} hazard levels. To determine the soil UHRS, the applicant used Approach 2B outlined in NUREG/CR-6728. The 10^{-4} and 10^{-5} soil UHRS were then used to calculate the GMRS for the LNP Units 1 and 2 site.

2.5.2.4.5.1 Site Response Inputs

An important part of site response analysis is the model of the site subsurface soil and rock properties. Key properties include site stratigraphy, unit thickness, V_S and strain dependent behavior of each of the soil and rock layers underlying the site. The LNP Units 1 and 2 site location, within a 1 km (0.5 mi) radius, stratigraphy is known to a depth of approximately 1,370 m (4,500 ft) from oil exploration that took place around 1949. Stratigraphy to a depth of 150 m (500 ft) beneath the LNP Units 1 and 2 site is known from geotechnical borings that were drilled as part of the applicant's COL application study, which are described in FSAR Section 2.5.4.

2.5.2.4.5.1.1 Shear Wave Velocity

In FSAR Figure 2.5.2-249, the applicant shows four median V_S profiles for the shallow subsurface. In that figure, there are two LNP Unit 1 profiles and two LNP Unit 2 profiles, where the subsurface beneath each unit is described using both suspension logging data and downhole data. That figure demonstrates the differences in V_S measurements obtained by the two different methods. In RAI 2.5.2-11, the staff asked the applicant to clarify which of the two velocity measurements provide more reliable data and why. In response to RAI 2.5.2-11, the applicant explained that it did not assess which approach was more reliable and that it used all four profiles in the initial assessment of site amplification. The applicant then selected the site profiles, one for LNP Unit 1 and one for Unit 2 that produced the largest amount of amplification for use in the final site response analyses. Ultimately, the applicant enveloped the results of the two LNP sites to produce the final GMRS. The staff concludes that the procedure the applicant described is a conservative method to assess site amplification. Therefore, the staff finds the applicant's response adequate and considers RAI 2.5.2-11 resolved.

2.5.2.4.5.1.2 Density

In FSAR Section 2.5.2.5.1.3, the applicant discusses subsurface densities beneath the LNP Units 1 and 2 site. The applicant presented data in FSAR Section 2.5.4.2.3.2 for weathered and unweathered limestone showing densities increasing from 1.92 grams per cubic centimeter (g/cm^3; 120 pounds per cubic foot (pcf)) near the surface to 2.24 g/cm^3 (140 pcf) below elevation -91.5 m (-300 ft) msl. The applicant then described that V_S increase below the elevation of -305 m (-1,000 ft) msl and that it is likely that this velocity increase corresponds to an increase in density. Therefore, the applicant applied density of 2.4 g/cm^3 (150 pcf) for the rock layers below elevation -305 m (-1,000 ft) mean sea level (msl).

In RAI 2.5.2-20, the staff asked the applicant to provide a reference for a functional relationship between limestone velocity and density and then, based on that information; provide justification for the density of 2.4 g/cm^3 (150 pcf) at depths below -305 m (-1,000 ft.) msl. In response to RAI 2.5.2-20, the applicant provided a relationship between P-wave velocity and rock density for sedimentary rocks from Gardner et al. (1974). According to that relationship, a P-wave velocity of 3.66 km/s (12,000 fps) corresponds to a density of approximately 2.4 g/cm^3 (150 pcf). FSAR Figure 2.5.2-250 shows that P-wave velocities below -305 m (-1,000 ft) are in the range of 3.66 to 3.96 km/s (12,000 to 13,000 fps). Therefore, a density of 2.4 g/cm^3 (150 pcf) below -305 m (-1,000 ft) is consistent with Gardner's relationship. The staff concludes that the applicant provided sufficient justification for use of a density of 2.4 g/cm^3 (150 pcf) below -305 m (-1,000 ft) at the LNP Units 1 and 2 site. The staff considers RAI 2.5.2-20 resolved.

2.5.2.4.5.1.3 Karst Feature Characterization and Permeation Grouting Program

In order to understand how thoroughly the subsurface karst features were characterized by geophysical testing and the extent of the applicant's grouting program, in RAI 2.5.2-2, the staff asked the applicant why geophysical tools, such as resistivity, microgravity, and seismic tomography were not used to further characterize the extent of subsurface karst features.

In response to RAI 2.5.2-2, the applicant described that during pre-COL application site selection investigations, surface refraction and microgravity surface geophysical surveys were performed in addition to a series of preliminary boreholes. The applicant found that these investigation methods did not produce reliable results at the site due to subsurface heterogeneities. As a result, the COL application investigation instead included a large number of borehole geophysical loggings and surveys. Seismic tomography was tested at the Savannah River Site in an attempt to characterize "soft zones" at a depth of approximately 44 m (145 ft). The staff reviewed the Savannah River Site Report (Cumbest, et al., 1996). In the report, seismic tomography discerned anomalous layers, but identification of specific cavities, including karst features, was not successful. Also, microgravity and electrical resistivity are insufficiently sensitive to characterize such features and the reliability of these technologies to find subsurface karst features is estimated as poor or fair. Regarding the geophysical tools the applicant used to characterize the extent of potential subsurface karst features, the staff concludes that additional geophysical investigations would not improve characterization of the site's subsurface karst features and that the applicant used adequate methods to characterize the extent of subsurface karst features. The staff considers RAI 2.5.2-2 resolved.

In RAI 2.5.2-1, the staff asked the applicant to describe its plans for ensuring that V_S post-grouting at the site was appropriately represented in the site response analyses. Since the applicant's permeation grouting program will inject grout material permanently into the subsurface beneath the LNP Units 1 and 2 site, in this RAI, the staff questioned whether the applicant's site characterization, including site uniformity and V_S, presented in its COL application will remain accurate after grout injection.

To address the staff's concerns, the applicant conducted a grout test program. The purpose of the grout test program was to validate the applicant's permeation grout program design and grouting techniques, to measure the change in V_S and permeability of the grouted zone, and to determine the amount of grout take in the subsurface. The applicant presented the shear wave test results from its grout test program. During the grout test program, the applicant made pre- and post-grouting measurements of V_S using P-S suspension logging. SER Figure 2.5.2-9 shows the applicant's seismic wave velocity results for pre- and post-grouting measurements. The pre- and post-grouting measurements were performed in cased 10-cm (4-inches (in)) borehole PVC pipe. The applicant additionally addressed a concern of the staff regarding this P-S suspension logging methodology. The staff's concern was whether the casing surrounding the borehole piping affected the applicant's velocity measurements. In response, the applicant provided a figure, which is now SER Figure 2.5.2-10, that illustrates a comparison between velocities obtained using the cased 10-cm (4-in) borehole PVC pipe P-S suspension logging methodology and the downhole layered model methodology.

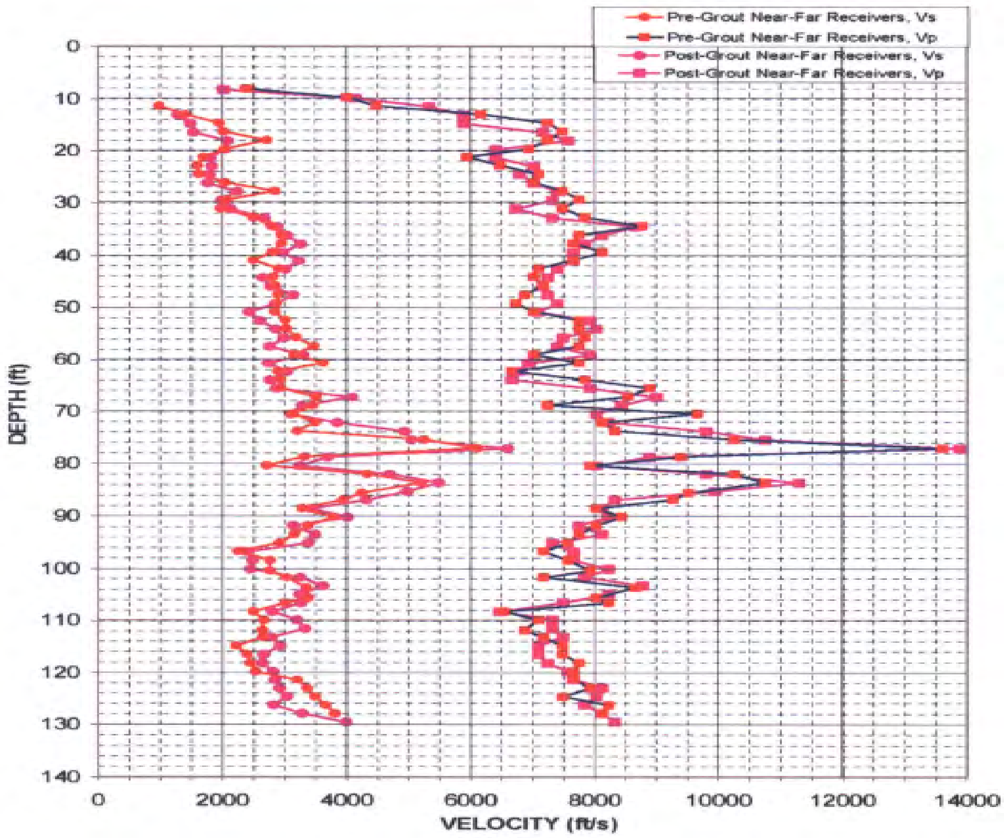


Figure 2.5.2-9. Pre- and Post-grouting Compressional (V_p) and Shear Wave (V_s) Velocities Suspension Logging Measurements (RAI Figure 02.05.02-1-01)

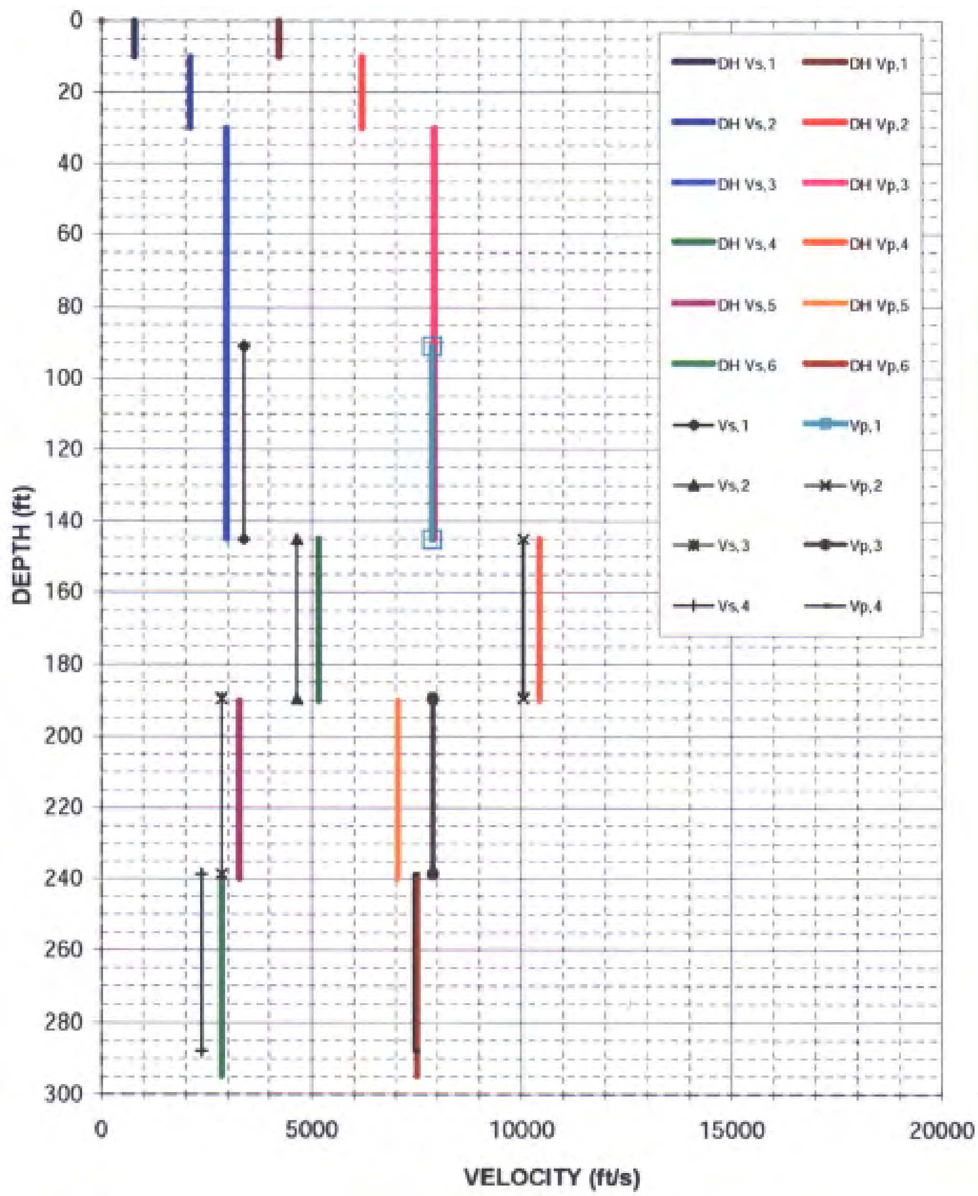


Figure 2.5.2-10. Seismic Wave Velocities Measured Using the Cased P-S Suspension Logging Methodology (Black Lines and Symbols) and Using Downhole Layered Models (Line and Symbols Labeled DH) (RAI Figure 02.05.02-1-02)

Regarding the pre- and post-grouting seismic wave velocities, the staff concludes that after the permeation grouting program is concluded, the LNP Units 1 and 2 site will maintain its site uniformity and V_S characterization as described in the LNP Units 1 and 2 COL application. As shown in SER Figure 2.5.2-9, the pre- and post-grouting measurements are within the expected precision of the P-S suspension logging method, and the change in V_S from pre- to post-grouting is within the standard deviation for the upper layers of the Avon Park Formation. Additionally, SER Figure 2.5.2-10 shows that both the cased and uncased P-S suspension logging methods produce similar seismic velocities versus depth and the comparison illustrates that the casing used in the borehole measurements did not systematically affect seismic wave velocity measurements during P-S suspension logging data collection. These comparative results of the cased P-S suspension logging and downhole layered models, assure the staff that the cased borehole piping did not significantly affect the applicant's seismic wave measurements. The staff considers RAIs 2.5.2-1 resolved.

2.5.2.4.5.1.4 Acceleration Time Histories

In FSAR Section 2.5.2.5.2, the applicant discusses its use of acceleration time histories for input rock motions in the site response analysis. The applicant developed response spectra for each controlling and deaggregation earthquake for the HF and LF ranges and 10^{-4}, 10^{-5}, and 10^{-6} hazard levels. Thirty time histories were chosen from the sets given in NUREG/CR-6728. The applicant scaled the time histories to match the target earthquake spectrum. However, the applicant did not provide the specific information about which acceleration time histories it chose from NUREG/CR-6728 and it provided minimal description of the scaling procedure used to match the spectra to the target earthquake spectra. In RAI 2.5.2-12, the staff asked the applicant to provide a list of the actual time histories used, specifically describing earthquakes and stations, which recorded the motion, and to describe in detail how the records were scaled.

In response to RAI 2.5.2-12, the applicant provided a list of the specific recordings used and recording parameters such as date, time, magnitude, station, and distance from event to station, among others. The applicant chose recordings from active tectonic regions and modified the spectra to have the general characteristics expected for rock site motions in the CEUS. Regarding the scaling of the input response spectra, the applicant explained that it first defined a target spectrum for each controlling and deaggregation earthquake. Second, the applicant scaled the individual input acceleration time histories in the frequency domain to match the target spectra. SER Figure 2.5.2-11 shows the time history from NUREG/CR-6728 and its response spectrum, the target spectrum, and the scaled time history and its response spectrum. After reviewing the specific list of the acceleration time histories used as input motions the staff concludes that the applicant demonstrated appropriate use of the input acceleration time histories and of the scaling process, because the applicant used inputs and scaling consistent with the controlling and deaggregation earthquakes. The staff finds the applicant's response to RAI 2.5.2-12 adequate and considers this RAI 2.5.2-12 resolved.

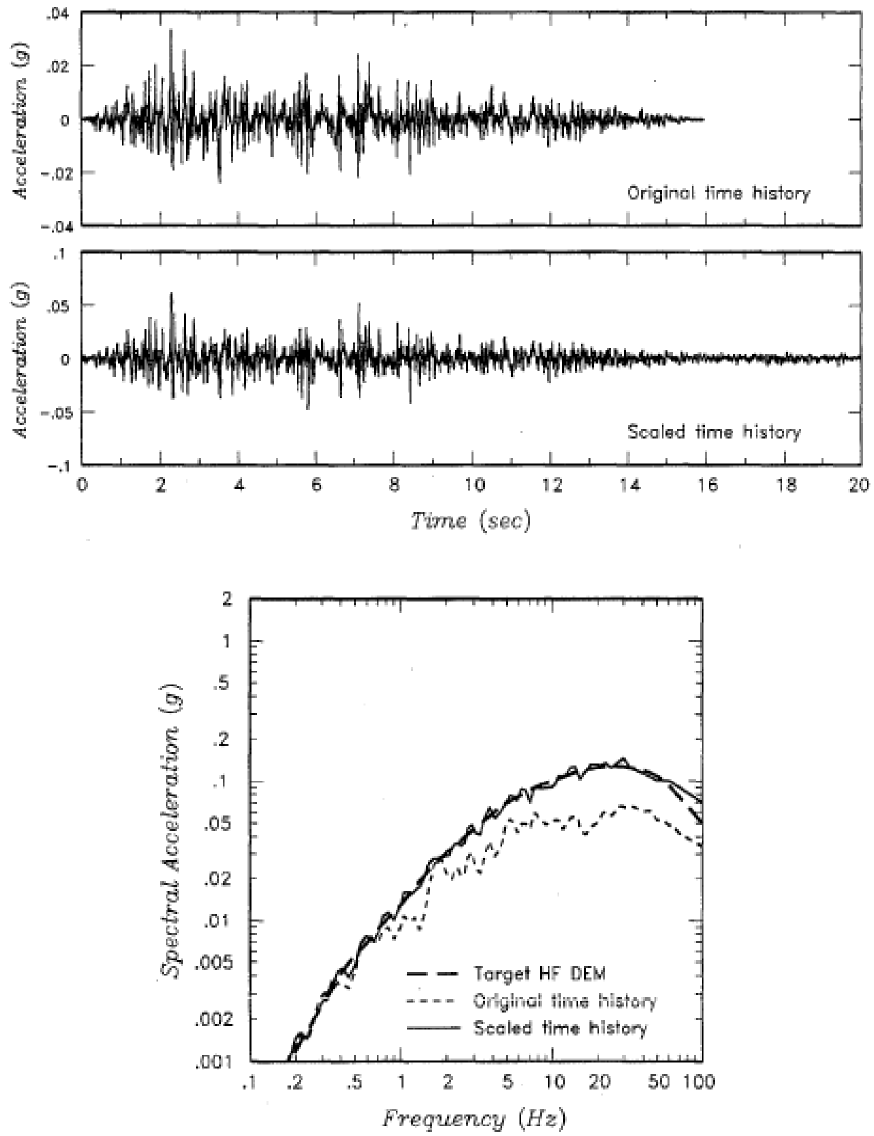


Figure 2.5.2-11. An Example of the Applicant's Scaling of Input Acceleration Time Histories Using the Parkfield Earthquake, San Luis Obispo 234° Component to Match the 10^{-4}, HF, DEM Earthquake Target Spectrum (RAI 02.05.02-12 Figure 2)

2.5.2.4.5.2 Site Response Methodology

In FSAR Section 2.5.2.5, the applicant describes the methodology it used to develop the soil UHRS for the 10^{-4}, 10^{-5}, and 10^{-6} hazard levels. To determine the soil UHRS, the applicant used Approach 2B outlined in NUREG/CR-6728, in which the applicant first developed soil models;

next randomized the soil profiles to account for variability; and lastly performed the final site response analysis.

FSAR Section 2.5.2.5.1 discusses the dynamic properties of the LNP Units 1 and 2 site. Seismological methods of site response calculations, including Approach 2B and analyses using the one-dimensional SHAKE program (Schnabel et al., 1972), used by the applicant are based on the assumption of a uniform (flat) layer structure under the site. In RAI 2.5.2-3 and RAI 2.5.2-21, the staff asked the applicant to justify the assumption of uniformity of layers based on the available boring and shear wave profiles, to clarify how variability was accounted for in the site response analysis, and to justify the use of only one V_S base model.

In response, the applicant described that analysis results indicate rock layer dips of 1 to 2 degrees and velocity data from three deep wells illustrate similar trends at depth. Likewise, the top of the basement rock dips at about 1 degree. To address variability in V_S, the applicant constructed four initial base case velocity profiles, calculated individual site responses for each, and chose the two profiles, one for LNP Unit 1 and one for LNP Unit 2 that resulted in the largest site amplification. The two chosen amplification functions were used to develop a single GMRS for the LNP site.

In order to review the applicant's responses to RAI 2.5.2-3 and 2.5.2-21, the staff evaluated the results of the dip analysis of the rock strata, the velocity data from the three deep wells, and the data regarding dip of the top of the basement rock. Dip analysis and well data indicate that the strata are flat-lying and suitable for use in the one-dimensional site response analysis. Additionally, the variability in layer velocity is accounted for by the use of multiple base-case profiles and then enveloping the site response amplification functions. For these reasons, the staff concludes that the assumption of a uniform (flat) layer structure under the LNP site is appropriate for the applicant to use for its site response analysis. In addition, the staff concludes that the applicant conservatively accounted for variability in V_S by enveloping the site amplification functions. The applicant provided sufficient information to address the staff's RAIs and the staff considers RAIs 2.5.2-3 and 2.5.2-21 resolved.

2.5.2.4.5.3 NRC Site Response Confirmatory Analysis

To determine the adequacy of the applicant's site response calculations, the staff performed its own confirmatory site response analysis. As input to its calculations of GMRS, the staff used the static and dynamic soil properties provided in FSAR Table 2.5.2-222 for LNP Unit 1 and FSAR Table 2.5.2-223 for LNP Unit 2. Those profiles consist of 29 layers on the top of hard rock at the depth of 1,325 m (4,350 ft) for GMRS at the elevation of 11 m (36 ft) NAVD88. The staff performed the site response calculations using the programs SHAKE2000 and STRATA, which are both based on the equivalent linear (EQL) method. To represent the input motions, the staff used 17 time histories of earthquakes similar in size and source-to-site distances to that of controlling earthquakes shown in SER Table 2.5.2-2. The staff weakly scaled the time histories. The staff first calculated site amplification functions for each of the 29-layer V_S profiles of LNP Units 1 and 2. Next, the staff took the maximum of the two site amplifications. Lastly, the staff enveloped the maximum of the two LNP Units 1 and 2 site amplification functions. The staff's resulting amplification curves are compared with the applicant's GRMS amplification functions in SER Figure 2.5.2-12. In the frequency range 0.1 to 30 Hz and 80 to 100 Hz (PGA), the applicant's site amplification functions are equal or exceed the staff's site amplification. The staff's site amplification function exceeds the applicant's in the frequency range of 30 to 75 Hz. This exceedance is not significant because of the limitations of methods used, where the EQL method produces accurate results up to the frequencies of 25 Hz. Furthermore, GMRS calculated using this AF is still much lower than the CSDRS. Therefore, the staff concludes that in the frequency range significant to a reactor's structures, systems, and components, there are no significant differences between the staff's and the applicant's calculated amplification functions for the LF and HF, 10^{-4} and 10^{-5} hazard levels.

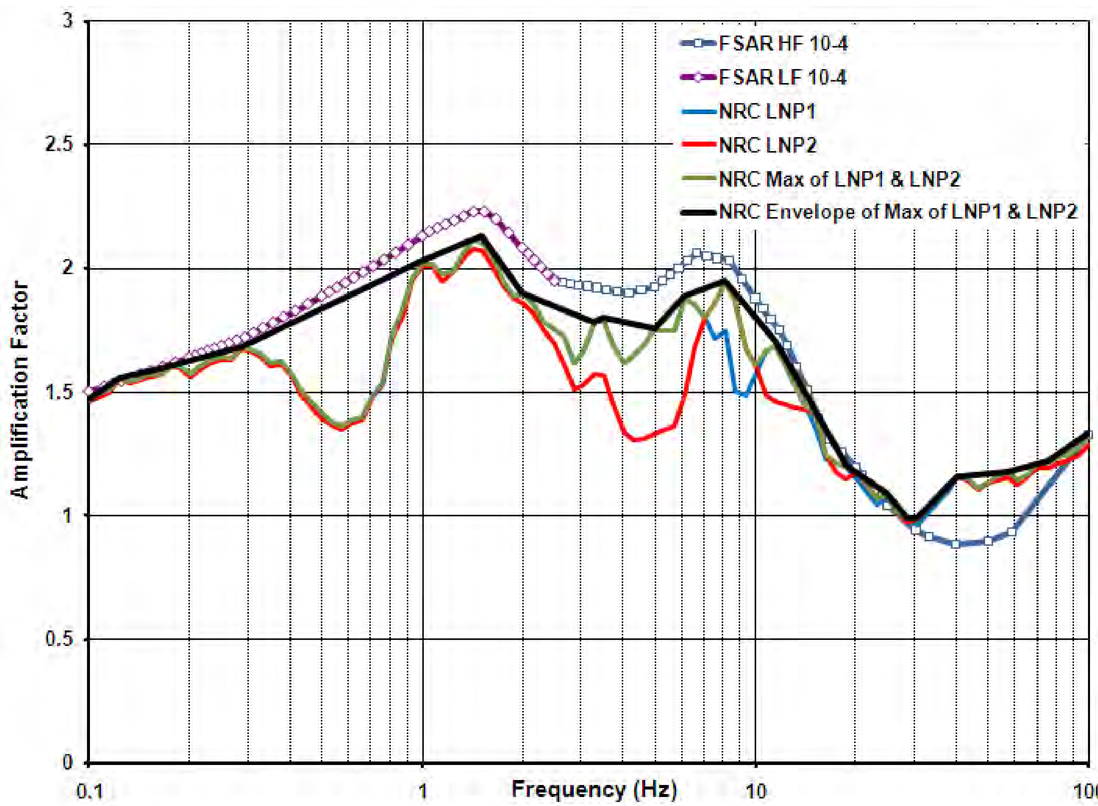


Figure 2.5.2-12. Results of the Staff's Confirmatory Analysis

2.5.2.4.5.4 Staff Conclusions Regarding Seismic Wave Transmission Characteristics of the Site

Based on the results of the staff's confirmatory analysis and the applicant's responses to RAIs 2.5.2-1 through 2.5.2-3, RAI 2.5.2-11, RAI 2.5.2-12, RAI 2.5.2-20, and RAI 2.5.2-21 discussed above, the staff concludes that the applicant's site response inputs, methodology, and results are acceptable. Specifically, the staff concludes that the applicant's site response inputs adequately characterize the site subsurface, that the permeation grouting program will not alter the site uniformity or V_S structure at the site, and that applicant adequately accounted for variability in V_S by enveloping the site amplification functions. The applicant used appropriate approaches to incorporate soil property uncertainties and followed the guidance provided in RG 1.208, which meets the requirements of 10 CFR 52.79(a)(1)(iii) and 10 CFR 100.23. This conclusion is further supported by the results of the confirmatory site response calculations performed by the staff that are similar to the applicant's results.

2.5.2.4.6 Ground Motion Response Spectra

RG 1.208 defines the GMRS as the site-specific SSE to distinguish it from the certified seismic design response spectra (CSDRS), the design ground motion for the AP1000 certified design. FSAR Section 2.5.2.6 describes the method the applicant used to develop the horizontal and vertical, site-specific GRMS. To obtain the horizontal GMRS, the applicant used the performance-based approach described in RG 1.208 and ASCE/SEI Standard 43-05 and additionally multipled the spectrum by a 1.212 scale factor. To develop the vertical GMRS, the applicant used V/H ratios, based on NUREG/CR-6728. The applicant's horizontal and vertical GMRS are shown in SER Figure 2.5.2-5.

In FSAR Section 2.5.2.6.4, the applicant describes its development of the vertical GMRS. The applicant used NUREG/CR-6728 to develop V/H ratios for an intermediate site, where an intermediate site has a subsurface characterized between rocks typical of sites in the WUS and sites in the CEUS. In RAI 2.5.2-13, the staff asked the applicant to clarify why the LNP Units 1 and 2 site was considered as intermediate and to justify the value used for kappa. In response to RAI 2.5.2-13, the applicant explained the kappa site value of 0.022 seconds was calculated using the EPRI (2005) empirical relationship between kappa and V_S. The applicant's kappa value of 0.022 seconds is between the typical value assigned to the WUS rock sites (0.04) and the value used for CEUS (0.006). The applicant stated that based on this kappa value, the peak in the V/H response spectral ratio would be expected to occur at an intermediate frequency between the values near 15 and 63 Hz for WUS and CEUS. Both EPRI (2005) and NUREG/CR-6728 are documents that the NRC supports for the use of seismic hazard analyses. Since the applicant developed its V/H ratios using these documents and the applicant's implementation of these documents was consistent with characterizing the site as intermediate, the staff concludes that the LNP Units 1 and 2 site is appropriately characterized as an intermediate site. The staff concludes that the applicant's calculated kappa values and V/H ratios for the LNP Units 1 and 2 site are acceptable. The staff considers RAI 2.5.2-13 resolved.

Based on the applicant's use of the standard procedure outlined in RG 1.208 to develop both the horizontal and vertical GMRS and the applicant having increased those spectra by a scale factor of 1.212, as well as on the applicant's responses to RAI 2.5.2-13, the staff concludes that the applicant's GMRS adequately represents the LNP Units 1 and 2 site ground motion.

2.5.2.4.7 Sensitivity Study of CEUS Seismic Source Characterization Model

On March 15, 2012, the NRC sent RAI Letter No. 108 (Agencywide Document Access and Management System (ADAMS) No. ML120550146) to the applicant. That letter explained that the staff was implementing some of the Fukushima Near-Term Task Force recommendations, as 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. ML12039A111). Among other topics, RAI Letter No. 108 requested that the applicant evaluate seismic hazards at the LNP site against current NRC requirements and guidance as described in SECY-2012-0025 Enclosure 7, Attachment 1 to Seismic Enclosure 1 (ADAMS Accession No. ML12039A188), and, if necessary, update the design basis and structures systems and components important to safety to protect against the updated hazards. The applicant responded to RAI Letter No. 108 in Progress Energy Letter NPD-NRC-2012-029 (ADAMS Accession No. ML122230155). The staff's evaluation of the applicant's response is located in SER Section 20.1. Based on the evaluation, the staff concludes that the scaled site-specific ground motions developed using the updated EPRI-SOG model with the CAV filter presented in FSAR Section 2.5.2.6 are appropriate for use as the design basis for the LNP site.

2.5.2.5 *Post Combined License Activities*

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

2.5.2.6 *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 vibratory ground motion, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

As set forth above, the staff reviewed the seismic information submitted by the applicant in LNP COL FSAR Section 2.5.2. On the basis of its review of the information in LNP COL 2.5-2 and LNP COL 2.5-3, the staff finds that the applicant has 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 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 LNP Units 1 and 2 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 LNP Units 1 and 2 COL site is acceptable from a geologic and seismologic standpoint and meets the requirements of 10 CFR 52.79 (a)(1)(iii) and 10 CFR 100.23.

2.5.3 Surface Faulting

2.5.3.1 *Introduction*

LNP COL FSAR Section 2.5.3 discusses the potential for tectonic (i.e., due to faulting) and non-tectonic surface and near-surface deformation at the LNP site. The applicant collected information related to both tectonic and non-tectonic surface and near-surface deformation during the LNP site characterization investigations and presented this information in the LNP COL FSAR in regard to the following specific topics: geologic, seismic, and geophysical investigations; geologic evidence, or absence of evidence, for surface deformation, including lineament analysis; correlation of earthquakes with capable tectonic sources; ages of most recent deformations; relationship of tectonic structures in the site area to regional tectonic structures; characterization of capable tectonic sources; designation of zones of Quaternary (2.6 Ma to present) deformation in the site region; and potential for tectonic and non-tectonic surface deformation at the site, including that associated with karst development.

2.5.3.2 *Summary of Application*

Section 2.5 of the LNP COL FSAR, Revision 9, incorporates by reference Section 2.5.3 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Section 2.5.3, the applicant provided site-specific information to address the following:

<u>*AP1000 COL Information Item*</u>

- LNP COL 2.5-4

The applicant provided additional information in LNP COL 2.5-4 to address COL Information Item 2.5-4 (COL Action Item 2.5.3-1). LNP COL 2.5.4 addresses the evaluation of site-specific subsurface geologic, seismic, and geophysical information in regard to the potential for surface or near-surface faulting at the site.

The applicant developed LNP COL FSAR Section 2.5.3 for the LNP site based on information derived from review of existing geologic and seismicity data and published literature; discussions with experts in geology, seismology, tectonics, and karst development who have conducted recent research in and around the site area; geologic field reconnaissance studies in the site vicinity and site area and at the site location; lineament analyses using aerial photographs and remote sensing imagery; and detailed investigations performed for the LNP COL application, including subsurface borings, surface geophysical testing, and downhole geophysical logging and seismic testing. The applicant also incorporated limited information applicable to the LNP site based on the CR3 FSAR (Florida Power Corporation, 1976), particularly in regard to fracture orientations and a lack of data indicative of faulting. The CR3 site is located about 18 km (11 mi) southwest of the LNP site.

Based on the information sources defined above, the applicant concluded in FSAR Section 2.5.3 that no deformational or geomorphic features indicative of potential Quaternary (2.6 Ma to present) tectonic activity at the LNP site have been reported in the literature, and that none were identified either by the site area experts or during the detailed field investigations performed for the LNP COL application. Following SER Sections 2.5.3.2.1 through 2.5.3.2.8 present a summary of the information provided by the applicant in LNP COL FSAR Section 2.5.3 related to tectonic surface deformation due to faulting, as well as non-tectonic surface deformation.

2.5.3.2.1 Geologic, Seismic, and Geophysical Investigations

FSAR Section 2.5.3.1 briefly describes the geologic, seismic, and geophysical investigations the applicant performed at the LNP site and in the site area to evaluate the potential for tectonic surface deformation, including surface fault rupture. The applicant cross-referenced FSAR Sections 2.5.1.2.1.3 and 2.5.1.2.4, which include detailed summaries of the information the applicant used to evaluate karst and site area and site vicinity structural geology, respectively, and concluded that no documented Quaternary (2.6 Ma to present) faults occur within the site region, site vicinity, or site area and that no capable tectonic sources exist therein. The applicant extended this conclusion to the faults postulated by Vernon (1951) to occur within the site vicinity and site area, which were also discussed in the FSAR for the CR3 site (Florida Power Corporation, 1976), based on the fact that no well-documented geologic evidence exists for these faults. The applicant also discussed the faults proposed by Vernon (1951) in FSAR Section 2.5.3.2 as addressed below in SER Section 2.5.3.2.2.

2.5.3.2.2 Geologic Evidence, or Absence of Evidence, for Surface Deformation

FSAR Section 2.5.3.2 discusses the presence or absence of surface deformation within the LNP site area. The applicant stated that recent geologic maps and evaluations of subsurface data do not show any structural features within the LNP site area. However, the applicant indicated that Vernon (1951) postulated seven faults in Citrus and Levy counties, three of which lie within the site area. These three postulated faults, the Inverness fault and two unnamed faults designated as Faults “A” and “B”, are shown in SER Figure 2.5.3-1 (reproduced from FSAR Figure 2.5.3-201). The applicant indicated that the northern end of the postulated Inverness Fault is located approximately 2 km (1.2 mi) east of the LNP site, and postulated Faults A and B are located about 4 km (2.5 mi) southwest and 7 km (4.3 mi) northeast of the site, respectively.

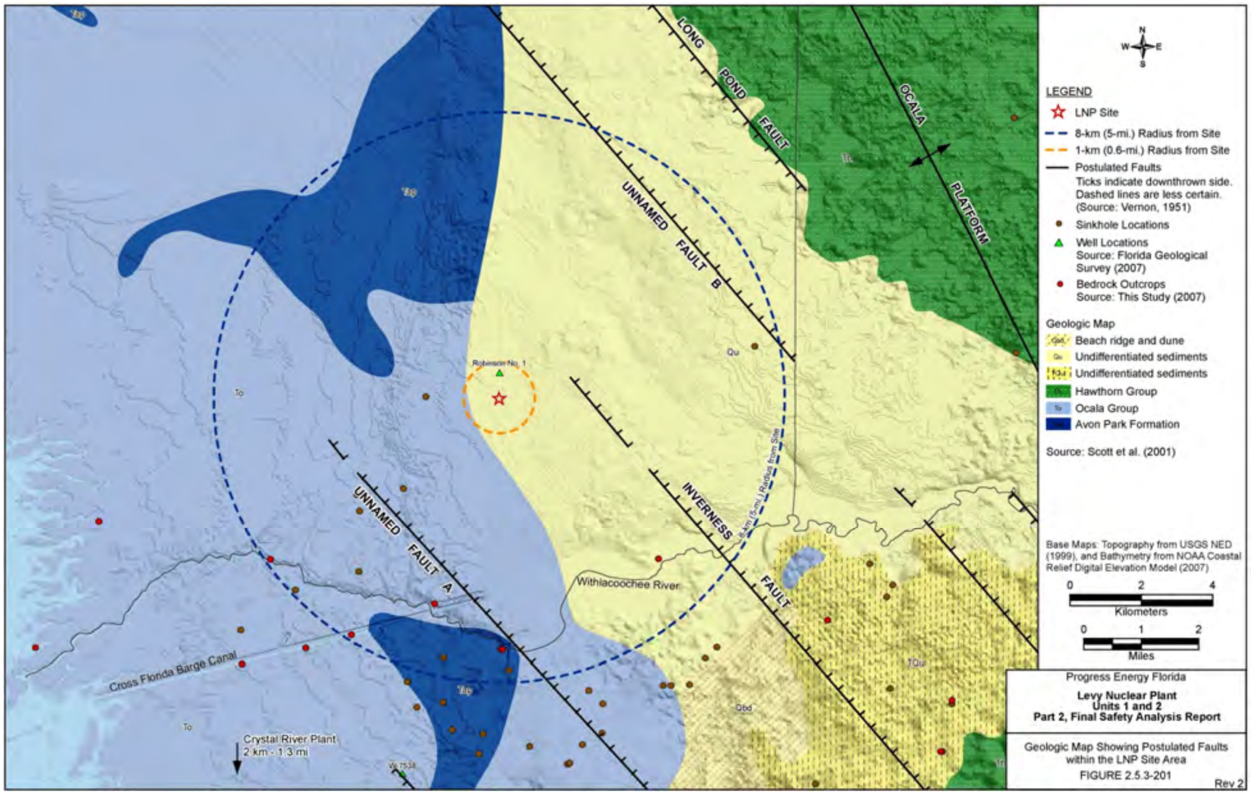


Figure 2.5.3-1. Geologic Map Showing Faults Postulated by Vernon (1951) to Lie Within the LNP Site Area (FSAR Figure 2.5.3-201)

The applicant reported that the faults postulated by Vernon (1951) to occur in the site area, based on his analysis of lineaments and interpretation of sparse geologic data, could not be detected in aerial photographs acquired in 1949; in Landsat images acquired in 2000; in a 10 m (32.8 ft) resolution USGS DEM; or in a DEM developed from 2007 light detection and ranging (LIDAR) data. The applicant also noted that the postulated faults do not disrupt marine terraces in the site area, which are estimated to be Late to Early Pleistocene (2.6 to 0.01 Ma), or possibly Pliocene (5.3 to 2.6 Ma), in age. The applicant further indicated that stratigraphic units used by Vernon (1951) to postulate the faults could not be differentiated, and that he based his interpretations on inferred correlation of stratigraphic units between widely-spaced outcrops and borehole data such that identification of the faults was highly speculative. The applicant cited Scott (1997), who noted that, because Vernon (1951) identified many of his faults based on interpreted offsets of the top of the Ocala Limestone, a surface with as much as 50 m (164 ft) of relief due to karst development, identification of faulting would be difficult at best.

Based on the results of research and geologic mapping as stated above, which post-date the work of Vernon (1951), the applicant concluded that no evidence exists for the three faults Vernon (1951) postulated to occur in the site area (SER Figure 2.5.3-1). In addition, based on analysis of lineaments at the site location scale using the 1949 aerial photographs and the DEM developed from the 2007 LIDAR data, the applicant further concluded that no mapped lineaments intersect the LNP Units 1 and 2 site locations, although the sites are located between zones of northwest-trending lineaments and a zone of northeast-trending lineaments lies between Units 1 and 2. The applicant interpreted these northwest and northeast-trending lineaments to be due to differential carbonate dissolution localized along joints rather than along faults, and recognized that lineaments control sinkhole alignment and stream drainages in the site area.

2.5.3.2.3 Correlation of Earthquakes with Capable Tectonic Sources

FSAR Section 2.5.3.3 discusses correlation of earthquakes with capable tectonic sources within the LNP site vicinity. The applicant cross-referenced FSAR Section 2.5.2.1 for earthquake catalog data, and stated that no recorded earthquakes greater than m_b = 3.0 exist within the LNP site vicinity. The applicant concluded that no historical earthquakes or alignment of earthquakes in the site region can be associated with any mapped fault.

2.5.3.2.4 Ages of Most Recent Deformations

FSAR Section 2.5.3.4 addresses ages of most recent deformations within the LNP site vicinity and at the LNP site. The applicant stated that basement rocks, which occur about 1,330 m (4,377 ft) beneath the LNP site, record Mesozoic (251 to 65.5 Ma) deformation related to rifting associated with development of the present-day Atlantic Ocean and the Gulf of Mexico. The applicant stated that there is no well-documented evidence for faulting of the Late Cretaceous (99.6 to 65.5 Ma) or Cenozoic (6.5 Ma to present) stratigraphic sections in the site vicinity, or for the faults postulated by Vernon (1951) to displace the Middle Eocene age (48.6 to 40.4 Ma) Avon Park Formation in the site area. The applicant indicated that there is no geomorphic evidence to suggest tectonic deformation due to faulting of the bedrock surface (i.e., a marine planation surface interpreted to be older than 340,000 years) underlying Quaternary (2.6 Ma to present) terrace deposits at the site location, and that no pronounced lineaments cut across the site location to suggest a through-going fault or major fracture system.

2.5.3.2.5 Relationship of Tectonic Structures in the Site Area to Regional Tectonic Structures

FSAR Section 2.5.3.5 discusses the relationship of tectonic structures in the site area to regional tectonic structures. The applicant stated that no documented faults occur within the site vicinity, but that the faults and fractures proposed by Vernon (1951) are sub-parallel to regional fracture trends observed throughout Florida. The applicant concluded that trends of fractures inferred from topographic lineaments and alignment of wetlands are consistent with trends of fractures inferred from analysis of regional lineaments and fracture sets observed in the site excavation for the CR3 site (Florida Power Corporation, 1976).

2.5.3.2.6 Characterization of Capable Tectonic Sources

FSAR Section 2.5.3.6 discusses characterization of capable tectonic sources within the LNP site vicinity. Based on review of published geologic data, interviews with technical experts knowledgeable about the site region and vicinity, and investigations performed by the applicant for the LNP COL application, the applicant concluded that no capable tectonic sources exist within the site vicinity. The applicant included the faults postulated by Vernon (1951) in the assessment of potential capable tectonic sources, concluding that no evidence exists for Quaternary deformation associated with these proposed structures.

2.5.3.2.7 Designation of Zones of Quaternary Deformation in the Site Region

FSAR Section 2.5.3.7 addresses zones of Quaternary (2.6 Ma to present) deformation in the site region. Based on review of available data and investigations preformed for the LNP COL application, the applicant concluded that no evidence exists for Quaternary tectonic deformation within the site region and site area or at the site location.

2.5.3.2.8 Potential for Surface Deformation at the Site

FSAR Section 2.5.3.8 discusses the potential for surface tectonic and non-tectonic deformation at the site. Based on review of available data and investigations preformed for the LNP COL application, the applicant stated that no capable tectonic faults or geomorphic features indicative of Quaternary (2.6 Ma to present) surface tectonic deformation occur within the site area. Consequently, the applicant concluded that the potential for surface tectonic deformation at the LNP site is negligible.

The applicant also concluded that the potential for non-tectonic surface deformation from any phenomenon other than karst-related subsidence or collapse is negligible at the site. To make this conclusion, the applicant assessed the potential effects of glacial rebound, intrusive and extrusive igneous activity, salt migration, growth faulting, and subsidence or collapse due to mining activity and gas extraction. The applicant discussed possible natural and human-induced controls on karst development, and stated that any potential for dissolution and formation of karst at the site will be mitigated by appropriate ground remediation and foundation design measures, including site-specific grouting. The applicant discussed the grouting program in detail in LNP COL FSAR Section 2.5.4.5.1, “Diaphragm Walls and Grouting.” The applicant summarized the available information reviewed as part of the karst development evaluation in FSAR Section 2.5.1.2.1.3, “Karst Terrain,” and presented the detailed evaluation of subsurface karst features in the vicinity of safety-related facilities at the LNP Units 1 and 2 site in FSAR Section 2.5.4.2, “Properties of Subsurface Materials.”

2.5.3.3 *Regulatory Basis*

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

The applicable regulatory requirements for 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 deformations at and in the region surrounding the site.

In addition, the related acceptance criteria associated with the relevant requirements of the Commission regulations are given in Section 2.5.3 of NUREG-0800 as follows:

- Geologic, Seismic, and Geophysical Investigations: Requirements of 10 CFR 100.23 are met and guidance in RG 1.132, Revision 2; RG 1.198; and RG 1.208 is 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 well with studies conducted by others in the same area, and are supported by detailed investigations performed by the applicant.

- Geologic Evidence, or Absence of Evidence, for Surface Tectonic Deformation: Requirements of 10 CFR 100.23 are met and guidance in RGs 1.132, Revision 2; RG 1.198; and RG 1.208 is followed for this area of review if sufficient surface and subsurface information is provided by the applicant for the site vicinity, site area, and site location to confirm presence or absence of surface tectonic deformation (i.e., faulting) and, if present, to demonstrate age of most recent fault displacement and ages of previous displacements.

- Correlation of Earthquakes with Capable Tectonic Sources: Requirements of 10 CFR 100.23 are met for this area of review if all reported historical earthquakes within the site vicinity are evaluated with respect to accuracy of hypocenter location and source of origin, and if all capable tectonic sources that could, based on fault orientation and length, extend into the site area or site location are evaluated with respect to potential for causing surface deformation.

- Ages of Most Recent Deformation: Requirements of 10 CFR 100.23 are met for this area of review if every significant surface fault and feature associated with a blind fault, any part of which lies within the site area, is investigated in sufficient detail to demonstrate, or allow relatively accurate estimates of, age of most recent fault displacement and enable identification of geologic evidence for previous displacements (if such evidence exists).

- Relationship of Tectonic Structures in the Site Area to Regional Tectonic Structures: Requirements of 10 CFR 100.23 are satisfied for this area of review by discussion of structural and genetic relationships between site area faulting or other tectonic deformation and the regional tectonic framework.

- Characterization of Capable Tectonic Sources: Requirements of 10 CFR 100.23 are met for this area of review when it has been demonstrated that investigative techniques employed by the applicant are sufficiently sensitive to identify all potential capable tectonic sources, such as faults or structures associated with blind faults, within the site area; and when fault geometry, length, sense of movement, amount of total displacement and displacement per faulting event, age of latest and any previous displacements, recurrence rate, and limits of the fault zone are provided for each capable tectonic source.

- Designation of Zones of Quaternary Deformation in the Site Region: Requirements of 10 CFR 100.23 regarding designation of zones of Quaternary (< 2.6 Ma) deformation in the site region are met if the zone (or zones) designated by the applicant as requiring detailed faulting investigations is of sufficient length and width to include all Quaternary deformation features potentially significant to the site as described in RG 1.208.

- Potential for Surface Tectonic Deformation at the Site Location: To meet requirements of 10 CFR 100.23 for this area of review, information must be presented by the applicant in this section if field investigations reveal that surface or near-surface tectonic deformation along a known capable tectonic structure (i.e., a known capable tectonic feature related to a fault or blind fault) must be taken into account at the site location.

In addition, the geologic characteristics should be consistent with appropriate sections from RG 1.132, Revision 2; RG 1.198; RG 1.206; and RG 1.208.

2.5.3.4 *Technical Evaluation*

The NRC staff reviewed Section 2.5.3 of the LNP 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 tectonic and non-tectonic 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 its supplements document the results of the staff's evaluation of the information incorporated by reference into the LNP COL application.

The staff reviewed the following information in the LNP COL FSAR:

AP1000 COL Information Item

- LNP COL 2.5-4

The NRC staff reviewed LNP COL 2.5-4 included in Section 2.5.3 of the LNP COL FSAR. LNP 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 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 deformation, (5) relationship of tectonic structures in the site area to regional tectonic structures, (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 LNP COL FSAR Section 2.5.3, the staff concludes that the applicant provided the information required to satisfy LNP COL 2.5-4.

The technical information presented in FSAR Section 2.5.3 resulted from the applicant's review of existing geologic and seismicity data and published literature; discussions with individuals who have conducted recent research in and around the site area; field reconnaissance studies in the site vicinity and site area and at the site location; lineament analyses using aerial photographs and remote sensing imagery; and detailed investigations performed for the LNP site, including subsurface borings, surface geophysical testing, and downhole geophysical logging and seismic testing. The applicant also provided limited information applicable to the LNP site as derived from the FSAR prepared by Florida Power Corporation (Florida Power Corporation, 1976) for the CR3, which is located about 18 km (11 mi) southwest of the LNP COL site. Through the review of LNP 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.

NRC staff focused the review of LNP COL FSAR Section 2.5.3 on the applicant's descriptions of previous studies and data collected during those studies, as well as on the results of investigations the applicant conducted to assess the potential for surface and near-surface tectonic and non-tectonic deformation at the site. The staff visited the site in April 2009 (ML092600064), supported by technical experts from the USGS, and interacted with the applicant and its consultants in regard to the geologic, seismic, geophysical, and geotechnical investigations being conducted for the LNP COL application. During this site visit, the staff examined core samples from the initial site characterization boreholes placed at the locations of containment structures and turbine buildings for LNP Units 1 and 2, as well as exposures of the Avon Park Formation along the Waccasassa River about 25 km (16 mi) northwest of the site. Examination of the core allowed staff to assess subsurface stratigraphic relationships at the site, and the outcrops along the river permitted staff to observe and measure spacing and orientation of fractures in the Avon Park Formation. The staff also visited the site in September 2009 (ML093280825) to examine core samples from the test grouting program. The staff noted grout uptake in a single vertical fracture intersected by one of the grout boreholes. Also during the September 2009 site audit, the staff examined exposures of the Avon Park Formation at the abandoned Gulf Hammock quarry about 19 km (12 mi) north-northwest of the LNP site, which again permitted staff to observe and measure spacing and orientation of fractures in the Avon

Park Formation. In addition, in February 2010 at the applicant's records facility in Virginia, the staff examined boring logs, core photographs, and written core sample descriptions for six additional boreholes, located to be offset approximately 1.5 m (5 ft) from the position of the initial site characterization boreholes. These "offset" boreholes were drilled using controlled coring techniques to improve core recovery and further characterize soft zones postulated to mark horizons of low recovery in the initial site characterization boreholes for LNP Units 1 and 2. The two site visits and the examination of boring logs, core photographs, and core descriptions enabled the staff to assess and confirm the interpretations, assumptions, and conclusions the applicant made regarding the potential for surface and near-surface tectonic and non-tectonic deformation at the LNP site, including features related to karst development.

The following SER Sections 2.5.3.4.1 through 2.5.3.5.8 present the staff's evaluation of the information the applicant provided in LNP COL FSAR Section 2.5.3 and in responses to RAIs on that FSAR section. In addition to the RAIs addressing specific technical issues related to tectonic and non-tectonic surface deformation at the site, discussed in detail below, the staff also prepared editorial RAIs to further clarify certain descriptive statements the applicant made in the FSAR and to qualify geologic features illustrated in FSAR figures. These editorial RAIs are not discussed in this detailed technical evaluation. Also, RAIs related to geologic issues resolved in FSARs previously prepared for other sites in the CEUS are not discussed in detail in this technical evaluation for the LNP site, but rather addressed by a cross-reference to and a summary of the pertinent information used to satisfactorily resolve the issues as presented in those FSARs.

2.5.3.4.1 Geologic, Seismic, and Geophysical Investigations

LNP COL FSAR Section 2.5.3.1 summarizes the geologic, seismic, and geophysical investigations the applicant performed to assess the potential for tectonic surface deformation due to faulting within 8 km (5 mi) and 40 km (25 mi) of the site (i.e., the site area and site vicinity, respectively), as well as the potential for surface fault rupture at the LNP Units 1 and 2 site. Based on the results of these investigations, the applicant concluded that no documented tectonic faults of Quaternary age (2.6 Ma to present) occur within the site region, site vicinity, or site area, and no evidence exists for any capable (i.e., Quaternary) surface faults at the site location.

The staff focused the review of FSAR Section 2.5.3.1 on documentation of the sources used by the applicant to conclude that no capable tectonic sources occur within the site area and site vicinity, and that no evidence exists for surface faulting at the site location. In RAI 2.5.3-1, the staff asked the applicant: (a) to identify the research workers contacted; and (b) to summarize the information they provided supporting the conclusions that no capable tectonic sources occur within the site vicinity and site area and no evidence for surface faulting exists at the site location. In the response to RAI 2.5.3-1, the applicant supplied names and affiliations of the research workers who were contacted and summarized the information used to support the conclusions that no capable tectonic sources occur within the site area and site vicinity and that no evidence exists for surface faulting at the site location. The applicant emphasized the following key and current interpretations by geologists at the FGS, which strongly support these two conclusions:

- The Ocala Platform, (also referred to by some researchers as the Ocala Arch), which occurs about 14 km (8.5 mi) east of the LNP site as shown in SER Figure 2.5.3-1, is the result of sedimentary downwarping and not faulting.

- The faults postulated by Vernon (1951) to occur in the site area (i.e., unnamed Faults “A” and “B” and the Inverness fault as shown in SER Figure 2.5.3-1) based on his lineament analysis are not confirmed by more recent field data. The lineaments he associated with faulting are interpreted to be due to localized dissolution of carbonate rocks along joints.

- No known surface faults occur in the site area and none are indicated in the subsurface based on well logs, which penetrate the Avon Park Formation, the proposed foundation unit at the LNP site.

Based on the review of the applicant’s response to RAI 2.5.3-1, in particular the current key interpretations provided by FGS geologists as summarized above, the staff concludes that the applicant documented the research workers contacted and summarized the pertinent information those workers provided to support the statements that no capable tectonic features occur within the site area and site vicinity and that no evidence exists for surface faulting at the site location. The staff makes this conclusion because the FGS geologists the applicant contacted are highly knowledgeable in regard to the geology and tectonic setting of Florida, and their interpretations are based on the most current data available. Furthermore, based on independent review of the technical publications provided by the applicant related to the geology and tectonic setting of Florida that support the statements made by the applicant, as well as the response to RAI 2.5.3-1, the staff further concludes that there is no reported evidence from current geologic, seismic, and geophysical investigations to indicate that capable tectonic features occur within the site area and site vicinity or that surface faulting exists at the site location. Consequently, the staff considers RAI 2.5.3-1 to be resolved.

Based on the review of LNP COL FSAR Section 2.5.3.1 and the applicant’s response to RAI 2.5.3-1, the staff finds that the applicant provided a thorough and accurate description of geologic, seismic, and geophysical investigations performed to assess the potential for tectonic surface deformation due to faulting within the site area and site vicinity, as well as the potential for surface fault rupture at the LNP site, in support of the LNP COL application.

2.5.3.4.2 Geologic Evidence, or Absence of Evidence, for Surface Deformation

FSAR Section 2.5.3.2 summarizes the information the applicant presented related to the geologic evidence, or absence of evidence, for surface deformation at the site. In regard to three faults postulated by Vernon (1951) to occur in the site area (i.e., Unnamed Faults “A” and “B” and the Inverness fault, located in SER Figure 2.5.3-1), the applicant documented that no studies performed more recently than those of Vernon (1951) provide any evidence for these three faults. In addition, based on information provided by FGS geologists, the applicant indicated that the features Vernon (1951) interpreted to show evidence of surface faulting outside the site area (i.e., slickensides, which are lineations indicating direction of slip along a failure surface, and tilted bedding) are most likely the result of non-tectonic surface deformation related to karst-induced collapse. The applicant concluded that no evidence exists to suggest that these postulated features are faults, or that the features exhibit any Quaternary (2.6 Ma to present) deformation. Based on the information derived from the lineament analyses discussed in FSAR Section 2.5.3.2.1, the applicant concluded further that linear features mapped at the site location are due to localized dissolution of carbonate rocks along joints, rather than surface faulting, and that no evidence exists for tectonic surface deformation at the site. The staff focused the review of FSAR Section 2.5.3.2 on the slickensides and tilted bedding ascribed by Vernon (1951) to surface faulting; the mechanism for propagating lineaments upward through unconsolidated sediments; subsurface cross section data that may show one of the faults Vernon (1951) postulated to occur in the site area; and an inferred tectonic basin located within the site region based on FSAR Figure 2.5.3-202, but which the applicant did not discuss in the FSAR.

2.5.3.4.2.1 Slickensides and Tilted Bedding

In RAI 2.5.3-2, the staff asked the applicant to summarize the logic for stating that slickensides and tilted bedding resulted from dissolution collapse to ensure that these features do not indicate the presence of capable tectonic structures in the site area. In response to RAI 2.5.3-2, the applicant documented that FGS geologists who have extensive experience in mapping karst features interpret the slickensides and tilted bedding observed by Vernon (1951) as non-tectonic features related to karst development. Based on information provided by those geologists, the applicant indicated that the slickensides were observed to have a limited lateral extent, to be clearly associated with dissolution collapse sinkholes, and to exhibit random orientations. Therefore, the applicant concluded that these features are non-tectonic in origin and specifically related to karst development rather than faulting.

Based on review of the applicant's response to RAI 2.5.3-2, the original discussion by Vernon (1951), and the field data disclosed by FGS geologists documenting that the slickensides have a limited lateral extent, are clearly associated with dissolution collapse sinkholes, and exhibit random orientations, the staff concludes that the slickensides, and by association the tilted bedding, ascribed by Vernon (1951) to faulting are non-tectonic features related to karst development. The staff makes this conclusion because the preponderance of field evidence strongly supports a non-tectonic origin for these features. Consequently, the staff considers RAI 2.5.3-2 to be resolved.

2.5.3.4.2.2 Lineament Propagation

In RAI 2.5.3-3, the staff asked the applicant to discuss the possible non-tectonic mechanisms for propagating a lineament upward through unconsolidated sediments. This information is important to ensure that lineaments occurring in unconsolidated sediments in the site area are not related to active faulting. In the response to RAI 2.5.3-3, based on Upchurch (2008), the applicant identified the following non-tectonic mechanisms, which can cause upward propagation of fractures in competent bedrock through overlying unconsolidated sediments, without requiring the presence of faulting, which produces lineaments visible at the ground surface. The applicant incorporated changes in FSAR Sections 2.5.3.2.1 and 2.5.3.2.1.1 to include a discussion of these and other non-tectonic mechanisms for propagation of bedrock fractures upward through overlying sediments.

- Settlement of unconsolidated sediments into solution-enlarged fractures in underlying consolidated strata.
- Differential weathering or erosion caused by groundwater movement across karst surfaces.
- Differential consolidation of sediments into relict erosional features preserved in underlying unconformity surfaces.
- Growth of vegetation in clay-rich or silt-rich, moisture-holding soils located over deeper bedrock features associated with fractures.

Based on the review of the applicant's response to RAI 2.5.3-3 and the associated changes in LNP COL FSAR Sections 2.5.3.2.1 and 2.5.3.2.1.1, as well as an independent examination of the information from Upchurch (2008), the staff concludes that the applicant documented potential non-tectonic mechanisms for propagating fractures upward through unconsolidated sediments, resulting in lineaments at the ground surface that are not associated with faulting. The staff draws this conclusion based on the independent review of the information from Upchurch (2008), who is a highly credible expert on fractures, photolineaments, and mechanisms for upward propagation of fractures in bedrock through overlying unconsolidated sediments. Consequently, the staff considers RAI 2.5.3-3 to be resolved.

2.5.3.4.2.3 Postulated Subsurface Tectonic Structures

In RAI 2.5.3-4, the staff asked the applicant to discuss the cross-section data from Arthur and others (2001), illustrated in FSAR Figure 2.5.1-245, in regard to whether subsurface faulting related to a fault postulated by Vernon (1951) could be responsible for the missing subsurface limestone unit in that section. This information is important for determining if subsurface evidence exists to suggest the presence of any of the faults Vernon (1951) postulated to occur in the site area. In response to RAI 2.5.3-4, the applicant stated that Arthur and others (2001) did not interpret or discuss faulting in relation to the cross-section shown in FSAR Figure 2.5.1-245. In addition, the applicant noted that Arthur and others (2008) did not identify any faults in the LNP site area, which offset the top of the Avon Park Formation based on the isopach and structural contour maps they constructed. Given the erosional and karstic nature of the top of the Avon Park Formation, which creates a very irregular surface, the applicant concluded that there is little stratigraphic control for defining subsurface faults in the site area and that no information provided by Arthur and others (2001 and 2008) suggests the presence of faults such as those postulated by Vernon (1951).

Based on the review of the applicant's response to RAI 2.5.3-4, examination of the cross-section shown in FSAR Figure 2.5.1-245, and independent appraisal of the isopach and structural contour maps prepared by Arthur and others (2008), the staff concludes that no information provided by Arthur and others (2001 and 2008) suggests the presence of subsurface faulting in the site area, which is young enough to offset the Middle Eocene (48.6 to 40.7 Ma) age Avon Park Formation. The staff makes this conclusion because none of the data presented by Arthur and others (2001 and 2008) indicate the existence of subsurface faults, such as those postulated by Vernon (1951), in the site area. Consequently, the staff considers RAI 2.5.3-4 to be resolved.

In RAI 2.5.3-9, the staff asked the applicant to describe an inferred basin-bounding fault labeled as “D/U” in FSAR Figure 2.5.3-202, which was not discussed in the FSAR, although it occurs within the site region. This information is important to determine whether this inferred feature may be a capable tectonic structure. In response to RAI 2.5.3-9, the applicant indicated that the inferred northeast-trending fault, labeled as “D/U” is based on subsurface data, which are not definitive. Applin, who initially proposed the structure, stated that this feature occurs beneath rocks of Mesozoic (251 to 65.5 Ma) age and does not affect either Mesozoic units or younger Cenozoic (65.5 Ma to present) sediments (Applin, 1951). Based on this information from Applin (1951), the applicant concluded that this structure, if it exists, is a basement feature that does not affect rocks younger than Mesozoic. The applicant made changes to FSAR Section 2.5.1.1.4.3.1, which discuss and qualify the age of this inferred structure.

Based on the review of the applicant’s response to RAI 2.5.3-9 and the associated changes in LNP COL FSAR Section 2.5.1.1.4.3.1, the staff concludes that the inferred structure, if it exists, is a basement feature that does not affect rock units younger than Mesozoic in age. The staff draws this conclusion based on the strong field evidence cited by the applicant, which provides a Mesozoic age constraint for the feature and marks it as a structure that is older than Quaternary (2.6 Ma to present) and, therefore, not a capable tectonic structure. Consequently, the staff considers RAI 2.5.3-9 to be resolved.

Based on the review of FSAR Section 2.5.3.2, the applicant’s responses to RAIs 2.5.3-2, 2.5.3-3, 2.5.3-4, and 2.5.3-9, and the associated changes in LNP COL FSAR Sections 2.5.3.2.1, 2.5.3.2.1.1 and 2.5.1.1.4.3.1, the staff finds that the applicant provided a thorough and accurate description of the geologic evidence, or absence of evidence, for surface deformation at the site in support of the LNP COL application.

2.5.3.4.3 Correlation of Earthquakes with Capable Tectonic Sources

FSAR Section 2.5.3.3 discusses the correlation of earthquakes with capable tectonic sources within the site region and site vicinity. Based on analysis of seismic events within 320 km (200 mi) and 40 km (25 mi) of the site using an updated earthquake catalog that spanned the time frame from 1826 through December 2006, the applicant concluded that no historically-reported earthquakes or earthquake alignments can be associated with any mapped fault in the site region or site vicinity.

The staff focused the review of FSAR Section 2.5.3.3 on completeness of the seismic and tectonic information used to assess the correlation of earthquakes with capable tectonic structures in the site vicinity and site region. Based on an independent review of the updated earthquake catalog data used by the applicant, including the discussion of these data presented in FSAR Section 2.5.2.1.1, and tectonic maps showing the locations of known faults and shear zones in the site region and site vicinity, the staff concludes that no evidence exists for any correlation between earthquakes and capable tectonic structures in the site region or site vicinity.

Based on the review of FSAR Section 2.5.3.3 and the discussion in FSAR Section 2.5.2.1.1 regarding the updated earthquake data, the staff finds that the applicant provided a thorough and accurate description of the correlation of earthquakes with capable tectonic sources in support of the LNP COL application.

2.5.3.4.4 Ages of Most Recent Deformations

FSAR Section 2.5.3.4 discusses data related to the ages of most recent deformation in the site vicinity and at the LNP site. The applicant stated that there is no documented evidence for faulting of Late Cretaceous (99.6 to 65.5 Ma) or Cenozoic (65.5 Ma to present) rocks in the site vicinity, or for the faults postulated by Vernon (1951) to displace the Middle Eocene (48.6 to 40.4 Ma) Avon Park Formation in the site area. The applicant did not present information to constrain the age of the faults postulated by Vernon (1951). The applicant also stated that there is no geomorphic evidence to suggest faulting of bedrock underlying Quaternary (2.6 Ma to present) terrace deposits at the site location, and that no pronounced lineaments indicate a through-going fault or major fracture system crosscutting the site location. However, FSAR Figures 2.5.3-216, 2.5.3-218, and 2.5.3-220 show lineaments within the LNP site location based on 2007 LIDAR data, 1949 aerial photographs, and 2007 aerial photographs, respectively.

The staff focused the review of FSAR Section 2.5.3.4 on age of the faults postulated by Vernon (1951), and whether lineaments occurring at the site location may be segments of regional fracture patterns that represent geologic structures that control dissolution. In RAI 2.5.3-5, the staff asked the applicant to summarize existing information, which constrains the age of the faults postulated by Vernon (1951), particularly in regard to data indicating they are older than Quaternary, if they exist. In response to RAI 2.5.3-5, the applicant stated that the recognized experts on deformation history of the site region at the FGS do not believe the faults postulated by Vernon (1951) exist based on current field data. The applicant indicated that Arthur and others (2008) used the most current data from surface geologic mapping and water and petroleum wells to develop structure contour maps that show no faults cutting the Avon Park Formation or the overlying Ocala Limestone. The applicant reported that lineaments identified in remote sensing imagery at both a regional and site-specific scale correlate with fracture trends observed in bedrock within the site vicinity, rather than with faults. Therefore, the applicant concluded that there is no evidence to support the existence of the faults postulated by Vernon (1951) in the LNP site vicinity or site area, or that the postulated faults, if they exist, are associated with Quaternary (2.6 Ma to present) tectonic deformation. The applicant noted that this conclusion rendered it unnecessary to summarize information constraining the age of the faults postulated by Vernon (1951). The applicant provided changes in FSAR Section 2.5.3.2 to document that recent data do not support the existence of the faults postulated by Vernon (1951) to occur in the site vicinity and site area.

Based on the review of the applicant’s response to RAI 2.5.3-5, which includes data provided by FGS geologists that post-date the work of Vernon (1951) and that the staff independently reviewed, as well as the associated changes in FSAR Section 2.5.3.2, the staff concludes that the more recent data do not support the existence of the faults postulated by Vernon (1951) to occur in the site vicinity and site area. The staff also concludes that no evidence exists to indicate that the lineaments identified by Vernon (1951) are indicative of Quaternary tectonic deformation. The staff makes these two conclusions because the recent field evidence provided to the applicant by FGS geologists, including the structure contour maps of Arthur and others (2008) that show no faults cutting the Avon Park Formation or the overlying Ocala Limestone as Vernon (1951) had suggested, strongly supports the interpretations that the faults postulated by Vernon (1951) do not exist and that the identified lineaments do not indicate Quaternary tectonic deformation. Consequently, the staff considers RAI 2.5.3-5 to be resolved.

In RAI 2.5.3-6, the staff asked the applicant: (a) to assess whether regional fractures may cross-cut the site location, even if discontinuously, as possibly suggested by lineaments shown in FSAR Figures 2.5.3-216, 2.5.3-218, and 2.5.3-220; and (b) whether these linear features represent geologic structures that exercise control on dissolution and sinkhole development. This information is important because fractures are known to exercise strong control on dissolution pathways in carbonate rocks. In response to RAI 2.5.3-6, the applicant cross-referenced FSAR Section 2.5.3.2.1.3 and reiterated that lineaments mapped at the site location likely reflect structurally controlled joints that have been enhanced by dissolution of carbonate and erosion. The applicant stated that the prominent northwest-trending alignment of shallow depressions located approximately 300 m (1,000 ft) west of the LNP Units 1 and 2 footprints in FSAR Figures 2.5.3-216, 2.5.3-218, and 2.5.3-220 is consistent with the strike direction of the predominant regional fracture set mapped by Vernon (1951), and with one of the predominant orthogonal outcrop-scale fracture sets mapped in exposures of the Avon Park Formation at the Gulf Hammock quarry and along the Wacasassa River, located 19 km (12 mi) and 25 km (16 mi) northwest of the site, respectively. The staff examined and measured fractures at the quarry and along the river during site visits in April and September 2009 (ML092600064 and ML093280825), and documented that outcrop-scale fractures do reflect regional fracture trends. The applicant concluded that the discontinuous character of the lineaments, the low relief exhibited by the marine terrace surface, and the absence of faulting in boreholes at the site location indicate there is no evidence to suggest that capable tectonic surface faults occur at the site. The applicant incorporated changes in FSAR Sections 2.5.3.2.1.3 and 2.5.3.4 to clarify that predominant trends of fracture sets at the site, as inferred from mapped lineaments, are consistent with regional fracture trends, stream drainage patterns, and sinkhole alignments.

Based on review of the applicant's response to RAI 2.5.3-6 and the associated changes in FSAR Sections 2.5.3.2.1.3 and 2.5.3.4, and the field observations made by staff during the April and September 2009 visits to the LNP site in regard to fracture patterns in the site vicinity, the staff concludes that lineaments mapped at the site location likely reflect structurally-controlled joints enhanced by dissolution and erosion, and that the prominent northwest-trending alignment of shallow depressions located approximately 300 m (1,000 ft) west of the LNP Units 1 and 2 footprints is consistent with the strike direction of the predominant regional fracture set mapped by Vernon (1951) and with one of the predominant outcrop-scale fracture sets. The staff makes this conclusion based on field observations made during the April and September 2009 site visits (ML092600064 and ML093280825), as well as an independent review of pertinent references the applicant cited which document the relationships between fractures and lineaments stated above. The staff also concludes that no capable tectonic surface faults occur at the site because of the field evidence cited by the applicant and directly observed by the staff related to the discontinuous expression of lineaments, the low relief of the marine terrace surface, and the absence of faulting in boreholes at the site location. Consequently, the staff considers RAI 2.5.3-6 resolved.

Based on review of FSAR Section 2.5.3.4, the applicant's responses to RAIs 2.5.3-5 and 2.5.3-6 and associated changes in LNP COL FSAR Sections 2.5.3.2 and 2.5.3.4, coupled with the observations made by staff during the April and September 2009 site visits (ML092600064 and ML093280825) in regard to regional and local-scale fracture patterns, which exist in the site vicinity, the staff finds that the applicant provided a thorough and accurate description of the ages of most recent deformation in support of the LNP COL application.

2.5.3.4.5 Relationship of Tectonic Structures in the Site Area to Regional Tectonic Structures

FSAR Section 2.5.3.5 discusses the relationship of tectonic structures in the site area to regional tectonic structures. The applicant stated that no documented bedrock faults occur within the site vicinity or site area, and that fracture trends inferred from topographic lineaments and alignment of shallow depressions and wetlands at the site location are consistent with trends of regional fractures inferred from lineament analyses.

The staff focused the review of FSAR Section 2.5.3.5 on completeness of the information used by the applicant to assess the relationship between tectonic features in the site area and regional tectonic structures. Based on the detailed up-to-date information presented in various parts of FSAR Section 2.5.3, which documents a lack of geologic evidence for tectonic faulting in the site area, as well as an independent review of that information and other published literature cited by the applicant, the staff concludes that small-scale topographic lineaments and alignment of shallow depressions and wetlands at the site location reflect the trends of regional fractures inferred from lineament analyses, rather than regional tectonic faults. The staff draws this conclusion because a preponderance of published data supports the interpretation that topographic lineaments and aligned shallow depressions and wetlands in the site area are not related to regional faults, but rather to regional fractures.

Based on the review of FSAR Section 2.5.3.5 and other FSAR sections, which document the lack of evidence for tectonic faulting in the site vicinity and site area, the staff finds that the applicant provided a thorough and accurate description of the relationship of tectonic structures in the site area to regional tectonic structures in support of the LNP COL application.

2.5.3.4.6 Characterization of Capable Tectonic Sources

FSAR Section 2.5.3.6 addresses the characterization of capable tectonic sources within the site vicinity. The applicant specifically addressed the faults postulated by Vernon (1951) to occur in the site vicinity, and documented that available data do not support the existence of these faults and that there is no evidence for Quaternary (2.6 Ma to present) deformation associated with any of these postulated structures. Therefore, the applicant concluded that no capable tectonic sources exist within the site vicinity requiring characterization.

NRC staff focused the review of FSAR Section 2.5.3.6 on completeness of the information used by the applicant to state that no capable tectonic sources requiring characterization exist with the site vicinity. Based on the detailed up-to-date information presented in various parts of FSAR Section 2.5.3, which documents a lack of geologic evidence for tectonic faulting or capable tectonic structures in the site vicinity, as well as an independent review of that information and other published literature cited by the applicant, the staff concludes that no capable tectonic sources requiring characterization exist within the site vicinity. The staff draws this conclusion because a preponderance of published data strongly supports the interpretation that no capable tectonic sources exist within the site vicinity.

Based on the review of FSAR Section 2.5.3.6 and other FSAR sections, which document the lack of evidence for capable tectonic sources at the site, the staff finds that the applicant provided a thorough and accurate description regarding characterization of capable tectonic sources within the site vicinity in support of the LNP COL application.

2.5.3.4.7 Designation of Zones of Quaternary Deformation in the Site Region

FSAR Section 2.5.3.7 addresses the designation of zones of Quaternary (2.6 Ma to present) deformation in the site region, which may require detailed investigations. The applicant cross-referenced the detailed information on site geology presented in FSAR Section 2.5.1.2 and concluded that, based on both surface and subsurface data, no zones of Quaternary deformation requiring further investigation occur within the site region, site area, or at the LNP site location. However, the applicant did not summarize the pertinent results from the subsurface investigations, which supported this conclusion.

NRC staff focused the review of FSAR Section 2.5.3.7 on documentation of subsurface data sources used by the applicant to support the conclusion that no zones of Quaternary deformation requiring further investigation occur within the site region, site area, or at the LNP site location. In RAI 2.5.3-7, the staff asked the applicant to summarize the data derived from subsurface investigations that support this conclusion. In the response to RAI 2.5.3-7, the applicant stated that FSAR Section 2.5.4.2 presents the results of the extensive geotechnical boring program conducted at the LNP site to investigate subsurface rock conditions, and that no faults or other tectonic structures were revealed in geologic logs for more than 100 borings. The applicant cross-referenced the response to RAI 2.5.1-10, which documented that there was no evidence for faults or associated tectonic structures in televiewer logs from eight geotechnical borings drilled to a maximum depth of 152 m (500 ft) below the ground surface within the nuclear island footprint. The applicant also referred to FSAR Section 2.5.1.2.5.2.1, which describes subsurface organic-rich marker beds in the Avon Park Formation at the LNP site, detected in geophysical logs and core samples from four boreholes, and stated that these beds do not display abrupt vertical offsets as would be expected if significant tectonic deformation had occurred. The applicant provided changes in FSAR Section 2.5.3 to include the additional information about site-specific subsurface observations discussed in the response to RAI 2.5.3-7.

During the site visits in April and September 2009 (ML092600064 and ML093280825), staff examined core samples from geotechnical boreholes drilled at the LNP site and noted that no evidence existed for subsurface faults at the site. In addition, in February 2010, the staff examined boring logs, core photographs, and written core sample descriptions for six additional boreholes located to further characterize zones of low recovery observed in boreholes drilled during the initial site characterization phase for LNP Units 1 and 2. Examination of these core logs, photographs, and descriptions also did not reveal the presence of subsurface faults at the site. Therefore, based on the review of the applicant's response to RAI 2.5.3-7, including the revisions in FSAR Section 2.5.3, as well as the direct observations made by staff during the April and September 2009 site visits and the results of the examination of core logs, core photographs, and core sample descriptions in February 2010, the staff concludes that the applicant properly summarized the subsurface information used to determine that no zones of Quaternary deformation, which would require further investigation occur within the site region, site area, or at the LNP site location.

Based on the review of FSAR Section 2.5.3.7, the applicant's response to RAI 2.5.3-7 and the associated changes in FSAR Section 2.5.3, as well as the direct examination by staff of core samples from the LNP site during the April and September 2009 site visits and of core logs, core photographs, and core sample descriptions in February 2010, the staff finds that the applicant provided a thorough and accurate description in regard the designation of Quaternary deformation zones in the site region in support of the LNP COL application.

2.5.3.4.8 Potential for Surface Deformation at the Site

LNP COL FSAR Section 2.5.3.8 discusses the potential for surface tectonic deformation, as well as non-tectonic surface deformation related to karst development and phenomena other than karst-induced collapse or subsidence, at the site. In FSAR Section 2.5.3.8.1, the applicant concluded that the potential for surface tectonic deformation at the site is negligible because no capable tectonic structures or geomorphic features indicative of Quaternary (2.6 Ma to present) deformation exist within the LNP site area. Also in FSAR Section 2.5.3.8.1, the applicant indicated that excavations for all safety-related structures for LNP Units 1 and 2 would be mapped in detail, and the NRC notified immediately if previously unrecognized geologic features that may represent a hazard to the facilities were identified. In addition, the applicant stated that any deformation features observed in the excavations would be characterized to assess the potential for surface deformation and ground motion following guidance in RG 1.208. These actions are identified as License Condition 2-1 under SER Section 2.5.3.5. In FSAR Section 2.5.3.8.2.1, the applicant stated that the potential for non-tectonic surface deformation at the site is negligible, except for phenomena related to karst-induced collapse or subsidence. In FSAR Section 2.5.3.8.2.2, the applicant specifically addressed the potential for karst-related non-tectonic surface deformation and concluded that karst-induced collapse and subsidence pose a potential geologic hazard at the LNP site.

NRC staff focused the review of FSAR Section 2.5.3.8 on completeness of the information the applicant used to assess the potential for surface tectonic and non-tectonic deformation at the site. In regard to tectonic surface deformation, based on the detailed up-to-date information presented in various parts of FSAR Section 2.5.3, which documents a lack of geologic evidence for surface or subsurface tectonic faulting in the site area, as well as an independent review of that information and other published literature cited by the applicant, the staff concludes that the potential for surface tectonic deformation at the site is negligible. For non-tectonic surface deformation, based on detailed up-to-date information presented in various parts of FSAR Section 2.5.3, which documents a lack of geologic evidence for non-tectonic surface deformation except for phenomena associated with karst-related collapse and subsidence, as well as an independent review of that information and other published literature cited by the applicant, the staff concludes that the potential for non-tectonic surface deformation exists only in connection with karst-related collapse and subsidence. The staff draws these two conclusions because a preponderance of published data strongly supports the interpretations that the potential for surface tectonic deformation at the site is negligible, and that phenomena associated with karst-related collapse and subsidence provide the only potential for non-tectonic surface deformation. In addition, detailed examination by staff during the April and September 2009 site visits of core samples taken from boreholes at the LNP site revealed only a few fractures and no extensive dissolution features or faults in support of the applicant's conclusions regarding tectonic and non-tectonic surface deformation. Furthermore, in February 2010, the staff examined boring logs, core photographs, and written core sample descriptions for six additional boreholes located to further characterize postulated soft zones, which were noted in boreholes drilled during the initial site characterization studies for LNP Units 1 and 2. Examination of these core logs, photographs, and descriptions likewise did not reveal the presence of either subsurface faulting or extensive dissolution cavities at the site.

Based on the review of FSAR Section 2.5.3.8 and other FSAR sections, which document the lack of evidence for surface tectonic faulting and the possibility of non-tectonic surface deformation related to karst development at the site, as well as the examination by staff during the April and September 2009 site visits of core samples from the LNP site and examination of core logs, photographs, and descriptions in February 2010, the staff finds that the applicant provided a thorough and accurate description of the potential for tectonic and non-tectonic surface deformation at the site in support of the LNP COL application.

2.5.3.5 *Post Combined License Activities*

Staff identified the following License Condition as the responsibility of the COL licensee in SER Section 2.5.3.4.8 ("Potential for Surface Deformation at the Site"). This License Condition relates to geologic mapping of both tectonic and non-tectonic (i.e., karst-induced collapse and subsidence) surface deformation features at the site.

- License Condition (2-1) – The licensee shall perform detailed geologic mapping of the excavations for LNP Units 1 and 2 nuclear island structures; examine and evaluate geologic features discovered in excavations for safety-related structures other than those for the Units 1 and 2 nuclear islands; and notify the Director of the Office of New Reactors, or the Director's designee, once excavations for LNP Units 1 and 2 safety-related structures are open for examination by NRC staff.

2.5.3.6 *Conclusion*

The NRC staff reviewed the application and checked the referenced DCD. The staff confirmed that the applicant addressed the required information related to tectonic and non-tectonic surface deformation, and that there is no outstanding information expected to be addressed in the LNP COL FSAR related to this topic. The results of the staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

As set forth above, the staff has reviewed the information in LNP COL 2.5-4 and finds that the applicant provided a thorough characterization of the potential for tectonic and non-tectonic surface and near-surface deformation, including faulting, at the LNP site, as required by 10 CFR 100.23 and 10 CFR 52.79(a)(1)(iii). Based on the review of the geologic and seismic information gathered by the applicant during the regional and site-specific investigations and presented in LNP COL FSAR Section 2.5.3, the staff concludes that the applicant performed its investigations in accordance with 10 CFR 100.23 and 10 CFR 52.79(a)(1)(iii) and followed guidance provided in RG 1.208. The staff also concludes that the applicant established an adequate basis to state that no known capable tectonic sources exist in the site vicinity, which would cause surface or near-surface deformation in the site area, and that the potential for surface or near-surface non-tectonic deformation in the site area is negligible, with the exception of karst-induced collapse and subsidence. Therefore, the staff concludes that the proposed LNP Units 1 and 2 COL site is acceptable from the perspective of surface and near-surface tectonic deformation and meets the requirements of 10 CFR 100.23 and 10 CFR 52.79(a)(1)(iii).

2.5.4 Stability of Subsurface Materials and Foundations

2.5.4.1 *Introduction*

Section 2.5.4 of this SER presents information on the static and dynamic stability of subsurface materials and foundations for the LNP Units 1 and 2 COL 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 related to the stability of subsurface materials and foundations covers the following specific areas: (1) geologic features in the vicinity of the site; (2) static and dynamic engineering properties of soil and rock strata underlying the site; (3) the relationship of the foundations for safety-related facilities and the engineering properties of underlying materials; (4) results of seismic surveys, including in-hole explorations; (5) safety-related excavation and backfill plans and engineered earthwork analysis and criteria; (6) groundwater conditions and piezometric pressure in all critical strata as to affect the loading and stability of foundation materials; (7) responses of site soils or rocks to dynamic loading; (8) liquefaction potential and consequences of liquefaction of all subsurface soils, including the settlement of foundations; (9) earthquake design bases; (10) results of investigations and analyses conducted to determine foundation material stability, deformation and settlement under static conditions; (11) criteria, references, and design methods used in static and seismic analyses of foundation materials; and (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*

Section 2.5 of the LNP COL FSAR, Revision 9, incorporates by reference Section 2.5.4 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Section 2.5.4, the applicant provided site-specific information to address the following:

AP1000 COL Information Items

- LNP COL 2.5-5

The applicant provided additional information in LNP COL 2.5-5 to resolve COL Information Item 2.5-5 (COL Action Item 2.5.1-1). LNP COL 2.5-5 addresses the provision of site-specific information regarding the underlying site conditions and geologic features, including site topographical features and the locations of seismic Category I structures.

- LNP COL 2.5-6

The applicant provided additional information in LNP COL 2.5-6 to resolve COL Information Item 2.5-6 (COL Action Item 2.6-3). LNP COL 2.5-6 addresses the properties of the foundation rock to be within the range considered for the design of the nuclear island basemat.

- LNP COL 2.5-7

The applicant provided additional information in LNP COL 2.5-7 to resolve COL Information Item 2.5-7 (COL Action Item 2.5.4-1). LNP COL 2.5-7 addresses the information concerning the extent (horizontal and vertical) of seismic Category I excavations, fills, and slopes.

- LNP COL 2.5-8

The applicant provided additional information in LNP COL 2.5-8 to resolve COL Information Item 2.5-8 (COL Action Item 2.4.1-1). LNP COL 2.5-8 addresses the ground water conditions relative to the foundation stability of the safety-related structures at the site.

- LNP COL 2.5-9

The applicant provided additional information in LNP COL 2.5-9 to resolve COL Information Item 2.5-9 (COL Action Item 2.5.4.3-1). LNP COL 2.5-9 addresses the provision of demonstrating that the potential for liquefaction is negligible.

- LNP COL 2.5-10

The applicant provided additional information in LNP COL 2.5-10 to resolve COL Information Item 2.5-10 (COL Action Item 2.6-4). LNP COL 2.5-10 addresses the verification that the minimum allowable bearing capacity of the site is greater than that specified in the AP1000 DCD with an adequate factor of safety.

- LNP COL 2.5-11

The applicant provided additional information in LNP COL 2.5-11 to resolve COL Information Item 2.5-11 (COL Action Item 2.5.2-2). LNP COL 2.5-11 addresses the methodology used in determination of static and dynamic lateral earth pressures and hydrostatic groundwater pressures acting on plant safety-related facilities using soil parameters as evaluated in previous sections.

- LNP COL 2.5-12

The applicant provided additional information in LNP COL 2.5-12 to resolve COL Information Item 2.5-12 (COL Action Item 2.5.5-1). LNP COL 2.5-12 addresses the rock characteristics affecting the stability of the nuclear island including foundation rebound, settlement, and differential settlement.

- LNP COL 2.5-13

The applicant provided additional information in LNP COL 2.5-13 to resolve COL Information Item 2.5-13 (COL Action Item 2.6-5). LNP COL 2.5-13 addresses the provision for instrumentation for monitoring the performance of the foundations of the nuclear island, along with the location for benchmarks and markers for monitoring the settlement.

- LNP COL 2.5-16

The applicant provided additional information in LNP COL 2.5-16 to resolve COL Information Item 2.5-16. LNP COL 2.5-16 addresses the verification that both total and differential settlements of the nuclear island, and the differential settlements between the nuclear island and other buildings do not exceed the AP1000 standard design.

- LNP COL 2.5-17

In a letter dated July 21, 2009, Westinghouse proposed COL Information Item 2.5-17 to provide a waterproofing system used for the below grade, exterior walls exposed to flood and groundwater under seismic Category I structures. COL Information Item 2.5-17 states that:

> The Combined License applicant will provide a waterproofing system used for the below grade, exterior walls exposed to flood and groundwater under seismic Category I structures. Waterproofing membrane should be placed immediately beneath the upper Mud Mat, and on top of the lower Mud Mat. The performance requirements to be met by the COL applicant for the waterproofing system are described in subsection 3.4.1.1.1.1.

Evaluation of the waterproofing capability of the system presented in LNP COL 2.5-17 occurs in Section 3.8 of this SER. The evaluation of the system's ability to meet the seismic requirements outlined in DCD Section 3.4.1.1.1.1 is located in Section 3.8 of this SER. The inspections, tests, analyses, and acceptance criteria (ITAAC) for the waterproof membrane is evaluated in Section 14.3 of this SER.

In addition, this LNP COL FSAR section addresses Interface Item 2.12, related to V_S, and Interface Item 2.13, related to the required bearing capacity of foundation materials.

In LNP COL FSAR Section 2.5.4, the applicant described the geotechnical explorations performed at the LNP site to determine the in-situ soil and rock properties and obtain samples for laboratory testing, the laboratory tests conducted to confirm the soil and rock properties, and the analyses made to determine the acceptability of the LNP Units 1 and 2 site as compared to the AP1000 DCD site requirements.

2.5.4.2.1 Geologic Features

LNP COL FSAR Section 2.5.4.1 summarizes the geologic features present at the LNP Units 1 and 2 sites, including those features that could relate to permanent ground deformations or foundation instability; areas of potential or actual subsurface subsidence, solution activity, uplift, or collapse; zones of alteration, irregular weathering, or structural weakness; unrelieved stresses in bedrock; rocks or soils that may be unstable; and the history of deposition and erosion. The applicant referred to FSAR Sections 2.5.1 and 2.5.3 for additional details of the geology and potential for surface faulting, respectively.

2.5.4.2.2 Properties of Subsurface Materials

FSAR Section 2.5.4.2 describes the subsurface investigation activities performed to characterize the soil and rock underlying the safety-related structures at the LNP site. All elevations given are with respect to the North American Vertical Datum of 1988.

2.5.4.2.2.1 Description of Investigation Activities

FSAR Section 2.5.4.2.1 describes the combination of field activities and laboratory tests performed and the engineering standards used to obtain the engineering properties of soils and rock at the LNP site.

2.5.4.2.2.2 Soil Boring and Rock Coring

The applicant described the initial, main and supplemental phases of the site investigations. During the initial phase, the applicant used sonic drilling to drill ten boreholes to characterize the subsurface conditions and conduct geophysical logging. As part of the main phase, the applicant drilled ninety boreholes to obtain soil and rock samples for laboratory testing. Based on the results of the initial and main phases, the applicant concluded that the subsurface conditions were potentially non-uniform. The applicant conducted a supplemental phase to drill eighteen boreholes to better characterize the subsurface conditions and the potential non-uniformity of the LNP site. An additional supplemental phase of drilling referred to as the "Offset Borings" (O-series) was drilled during the COL application review in response to requests for additional information. These borings are discussed in detail in Section 2.5.4.4 of this SER. The A-, B-, AD- and O-series borings were drilled within or in close proximity to the footprint of the nuclear islands and were relied on by the staff in the evaluation of the foundation conditions. SER Figure 2.5.4-1 shows, in plan, the relationship between LNP Units 1 and 2 and the boring locations.

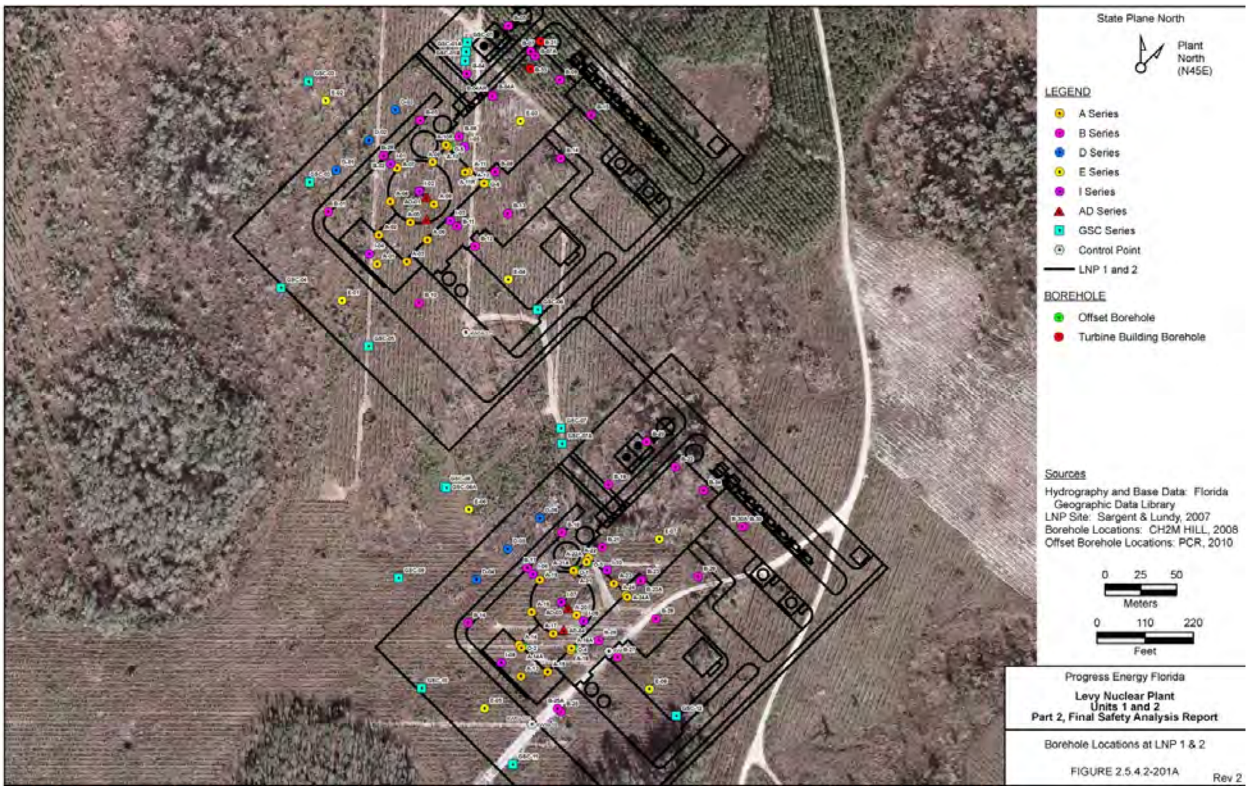


Figure 2.5.4-1. LNP 1 and 2 Boring Location Plan
(FSAR Figure 2.5.4.2-201A)

Criteria for Selection of Borehole Locations and Depths

The applicant selected borehole locations and depths for the initial and main phases following the criteria provided in RG 1.132. The applicant selected the location of supplemental phase borings based on the final orientation of the buildings and the need to obtain additional information for engineering analysis purposes. The applicant advanced borings in the supplemental phase to depths exceeding the maximum dimension of the nuclear island of 78 m (256 ft).

2.5.4.2.2.3 Drilling and Sampling Method

For the initial phase, the applicant used a Rotosonic (sonic) drilling method to continuously sample the soil and rock for visual classification purposes. During the main phase, the applicant used the mud rotary drilling method to advance the boring to collect representative disturbed soil samples using standard penetration tests (SPT) methods in accordance with American Society for Testing and Materials (ASTM) D 1586-99 and to obtain rock core samples using NQ- and HQ-sized, double tube diamond-tipped rock coring tools, in accordance with ASTM D2113. The applicant was unable to obtain undisturbed samples of soil due to the granular nature of the soil.

2.5.4.2.2.4 Field Observations, Logs, and Field Tests

The applicant conducted field investigation activities to characterize the types of soil and rock, soil consistency, rock strength and stiffness. The applicant recorded observations on boring logs, including visual descriptions of soil samples and rock cores, SPT N-values, field measurements of rock soundness and strength, rock core recovery, rock quality designations (RQDs), and R-values. Field tests such as field point load tests (PLTs) on rock cores, and in-situ rock pressuremeter tests (PMTs) in uncased boreholes are summarized in the FSAR.

2.5.4.2.2.5 Basis for Selection of Field Rock Hardness and Strength Tests

The applicant estimated the rock consistency at the LNP site using various field and laboratory tests, including unconfined compressive strength (UCS) tests, field R-scale tests, field PLTs, and downhole PMTs. The UCS tests provided the primary intact rock strength, while the R-scale tests and PLTs provided a check of the UCS data. The applicant performed PMTs in two boreholes at various depths to estimate the in-situ elastic modulus of the rock.

2.5.4.2.2.6 Geophysical Surveys

The applicant performed a series of seismic and non-seismic surveys, including suspension P-S velocity logging, downhole velocity logging, acoustic televiewer surveys, natural gamma measurements, gamma-gamma measurements, neutron-neutron measurements, and induction measurements. The applicant used the V_s profiles from the seismic surveys to determine the GMRS and estimate the stiffness of the Avon Park limestone. The non-seismic geophysical survey data was used to evaluate the stratigraphy at the site.

2.5.4.2.2.7 Management of Soil and Rock Core Samples

FSAR Section 2.5.4.2.1.4 describes the management of soil and rock samples. The applicant stored soil samples recovered by SPT sampling in watertight jars and routine-care rock core samples in core boxes kept at onsite long-term storage facilities. The applicant shipped special-care rock core samples to laboratory facilities for testing.

2.5.4.2.2.8 Laboratory Testing of Soil and Rock

In FSAR Section 2.5.4.2.1.5, the applicant described the laboratory testing of soil and rock at the LNP site, including a summary of the laboratory tests performed and the criteria for the selection of soil and rock samples.

Laboratory Tests Performed

The applicant presented the results of the laboratory tests performed on the special-care intact rock cores, which included UCS tests with axial and radial strain measurement, triaxial compression tests and split-tensile strength tests, petrographic examinations, and x-ray fluorescence examinations. The applicant also performed index tests, resistivity tests, pH tests, and organic content tests on SPT soil overburden samples. The applicant performed additional soil tests on two non-lithified and highly organic soil-like samples sandwiched within the Avon Park limestone at depths significantly below the foundations of the nuclear islands at the LNP site.

2.5.4.2.2.9 Criteria for Selection of Soil Samples for Laboratory Testing

The applicant classified any material that could be penetrated and sampled using SPT methods as “soil” or “soil-like.” The applicant plans to excavate these materials within the nuclear island footprint to the top of rock designated at an elevation (El.) of -7.3 m (-24 ft), prepare the rock surface with dental concrete, and overlay the Avon Park limestone with a 10.7 m (35 ft) thick roller compacted concrete (RCC) bridging mat. The applicant concluded that the laboratory tests on these materials are only relevant for existing soils outside the limits of the nuclear island where they are not excavated and replaced by more stable materials.

2.5.4.2.2.10 Criteria for Selection of Rock Core Samples for Laboratory Testing

The applicant collected special-care rock core samples in order to target specific elevation ranges, characterize different rock types, span the range of apparent rock core soundness, and obtain information on identified rock layers.

2.5.4.2.2.11 Results of Soil and Rock Tests Obtained from Field Investigations

The applicant recorded SPT blow counts (N) in the soil overburden and obtained disturbed samples from the split-spoon sampler for identification of soil and soil-like materials. Beginning at the top of the Avon Park limestone the applicant used a double-tube core barrel to recover rock cores. The applicant noted core recovery, RQD, R-scale values, PLT indices, time of drilling, water circulation loss, rod drops and descriptions of the recovered core on the core logs. Field PMT data in rock was obtained in two uncased boreholes during the field exploration.

2.5.4.2.2.12 Standard Penetration Test Blow Counts (N)

The applicant recorded SPT blow counts (N), the penetration resistance of the soil measured in blows per foot (bpf), at 0.76 to 1.5 m (2.5 to 5.0 ft) intervals from the existing ground surface to the depth of the top of rock in accordance with ASTM D1586 (1999). The applicant used the N-value to characterize three distinct soil layers at LNP Units 1 and 2: top layer S-1 with N-values of less than 30 bpf, intermediate soil S-2 with N-values between 30 and 50 bpf, and bottom soil S-3 with N-values greater than 50 bpf.

2.5.4.2.2.13 Rock Quality Designation, Rock Mass Quality, and Karst Features

The applicant determined the RQD, which is a rock soundness index, based on the length of recovered core pieces greater than 10 cm (4 in) compared to the total length of recovered core. The applicant used RQD values in combination with other data to delineate distinct rock layers. The applicant determined that the karst features identified in the core borings were either voids or soil-infilled based on drilling criteria, such as time of drilling, water circulation loss, and driller comments regarding rig behavior. Subsequent to the offset boring program, the applicant concluded that postulated infilled features are severely weathered or degraded dolomite with properties consistent with the Avon Park Formation.

2.5.4.2.2.14 R-Scale Strength Values

The applicant stated that the R-scale values provide a qualitative indication of rock strength and rated the rock at the site as R2 (weak rock) or stronger. FSAR Appendix 2BB reports the R-values recorded in the rock core logs.

2.5.4.2.2.15 Rock Pressuremeter Test (PMT) Modulus (Epmt)

LNP COL FSAR Section 2.5.4.2.2.5 states that the rock PMTs were performed at various depths in two boreholes at LNP Units 1 and 2. LNP COL FSAR Table 2.5.4.2-206 presents the results, which show that the Young's modulus values range from 6.9 to 1,689 megaPascal (MPa) (1 to 245 kips per square inch (ksi)) and 213 to 2,171 MPa (31 to 315 ksi) at LNP Units 1 and 2, respectively. Because the nature of the soft rock prevented the applicant from completing a sufficient number of pressure stages to provide complete and accurate results, the applicant concluded that the Young's modulus values obtained from the PMT were "worst case" estimates and, therefore, were not used in the engineering analyses.

2.5.4.2.2.16 Hydraulic Conductivity Tests

The applicant installed monitoring wells at the LNP site to monitor the seasonal fluctuation in ground water elevations and observation wells to assess the hydraulic conductivity of soil and rock. The applicant also measured the hydraulic gradients from the onsite ground water monitoring wells and referred to FSAR Section 2.4.12.2 for a more detailed description of the ground water hydrology at LNP Units 1 and 2.

2.5.4.2.2.17 Criteria for Soil Depth and Top of Rock

Because the top of rock was not distinct, the applicant defined the "top of rock" as the first occurrence of rock core and subsequent rock core runs recovering at least 50 percent of the core and having a minimum RQD of 25 percent in each core run.

2.5.4.2.2.18 Results of Soil Laboratory Tests

Based on the results of the petrographic analyses, the applicant concluded that the Avon Park limestone was dolomitized making it more resistant to future development of karst features.

2.5.4.2.2.19 Rock and Soil Properties for Use in Engineering Analyses

FSAR Section 2.5.4.2.4 summarizes rock and soil properties obtained from the field and laboratory tests. SER Table 2.5.4-1 compiles the elastic modulus, Poisson's ratio, rock mass shear strength parameters developed using the Hoek-Brown criteria, V_s and compression wave (V_p) velocity obtained from the suspension P-S velocity logging and the rock mass modulus derived from three data sources. The applicant noted that the rock mass elastic modulus (E_{rm}) values based on UCS data were 40 to 90 percent lower than those estimated using the small strain seismic data. SER Table 2.5.4-2 presents the soil properties and strength parameters derived from empirical relationships.

Table 2.5.4-1. Summary of Rock Samples
(Data Compiled from FSAR Tables 2.5.4.2-211 through 2.5.4.2-215)

	LNP 1			LNP 2			
	SAV*-1	SAV-2	SAV-3	NAV**-1	NAV-2	NAV-3	NAV-4
UCS, Elastic Moduli, Poisson's Ratio and Index Test Results							
Average UCS, MPa (psi)	25.9 (3,760)	5.07 (736)	25.4 (3,690)	16.6 (2,414)	20.2 (2,938)	16.9 (711)	17.4 (2,526)
Poisson's Ratio – Secant	0.29	0.50	0.22	0.34	0.30	0.36	0.16
Bulk Density, kg/m^3 (pcf)	2,210 (138)	2,002 (125)	2,306 (144)	2,146 (134)	2,178 (136)	1,890 (118)	2,162 (135)
Moisture Content, %	10	23	13	14	11	23	20
Poisson's Ratio – Tangent	0.36	0.51	0.32	0.44	0.37	0.53	0.16
Tensile Strength Test Results							
Tensile Strength, kPa (psi)	4,840 (702)	n/a	4,536 (658)	1,640 (238)	3,874 (562)	158.5 (23)	1,130 (164)
Bulk Density, kg/m^3 (pcf)	2,290 (143)	n/a	2,418 (151)	2,098 (131)	2,194 (137)	1,954 (122)	1,938 (121)
Moisture Content, %	9	n/a	10	17	12	27	21
Hoek-Brown Rock Mass Strength Parameters							
Unit Weight, kg/m^3 (pcf)	2,210 (138)	2,002 (125)	2,306 (144)	2,146 (134)	2,178 (136)	1,890 (118)	2,162 (135)
Representative UCS of Intact Rock, MPa (psi)	25.5 (3,700)	4.82 (700)	24.8 (3,600)	16.5 (2,400)	19.9 (2,900)	4.82 (700)	17.2 (2,500)
GSI	31	21	27	37	38	22	31
Rock Mass Cohesion, kPa (psi)	186 (27)	144 (21)	565 (82)	179 (26)	365 (53)	137 (20)	496 (72)
Rock Mass Friction Angle	24	15	22	24	25	16	21

Table 2.5.4-1. Summary of Rock Samples (Data Compiled from FSAR Tables 2.5.4.2-211 through 2.5.4.2-215)

	LNP 1			LNP 2			
	SAV*-1	SAV-2	SAV-3	NAV**-1	NAV-2	NAV-3	NAV-4
Suspension Logging							
V_s, m/s (fps)	1,198 (3,932)	893 (2,932)	1,170 (3,839)	1,115 (3,660)	1,406 (4,614)	943 (3,097)	1,207 (3,963)
V_p, m/s (fps)	893 (9601)	2,366 (7,763)	2,756 (9,045)	2,549 (8,365)	3,022 (9,916)	2,440 (8,008)	2,775 (9,105)
Poisson's Ratio	0.39	0.41	0.38	0.38	0.35	0.41	0.38
Young's Modulus, kPa (ksi)	9,507 (1,379)	4,660 (676)	8,990 (1,304)	7,535 (1,093)	11,948 (1,733)	4,881 (708)	8,928 (1,295)
E_{rm} Values by Method Used, kPa (ksi)							
V_s, m/s (fps)	690 (4,757)	338 (2,330)	652 (4,495)	547 (3,771)	867 (5,977)	354 (2,440)	647 (4,460)
Rock PMT	834 (121)	-	-	427 (62)	-	-	-
UCS Testing [a]	1,048 (152)	268 (39)	1,640 (238)	875 (127)	1,758 (255)	351 (51)	2,868 (416)
UCS Testing [b]	1,172 (170)	179 (26)	1,075 (156)	979 (142)	1,489 (216)	234 (34)	1,799 (261)

* SAV is a rock unit of the Avon Park formation at the south reactor site
** NAV is a rock unit of the Avon Park formation at the north reactor site
[a] Hoek and Diederichs (2006)
[b] Yang (2006)

Table 2.5.4-2. **Estimated Properties of Soils above the Top of Rock (Modified from FSAR Tables 2.5.4.2-216 and 2.5.4.2-217)**

	North Reactor LNP 2			South Reactor LNP 1		
	S-1	S-2	S-3	S-1	S-2	S-3
Based on Laboratory Index Properties						
Avg. f'_{cv}, deg.	31	30	29	31	n/a	-
OCR	1.7	1.0	2.0	4.4	n/a	-
s_u kPa (psf)	21.4 (449)	30.4 (636)	70.4 (1,471)	36.8 (769)	n/a	-
C_c	0.31	0.34	0.38	0.30	n/a	-
C_r	0.05	0.07	0.08	0.06	n/a	-
C_{ea}	0.003	0.004	0.004	0.002	n/a	-
Based on SPT N-values						
Mean SPT N-value, bpf	10	43	85	9	43	82
N_{60}, bpf	11	45	86	11	52	86
Moist Unit Weight, kg/m^3 (pcf)	1,762 (110)	1,922 (120)	2,082 (130)	1,762 (110)	1,922 (120)	2,082 (130)
Relative Density, %	25	50	90	25	50	90
Effective Friction Angle	28	31	36	28	31	36
Effective Cohesion	0					
Elastic Modulus[a], MPa (psi)	5.57 (808)	22.8 (3,307)	43.5 (6,319)	5.57 (808)	26.3 (3,821)	43.5 (6,319)
Elastic Modulus[b], MPa (psi)	11.9 (1,736)	27.7 (4,028)	47.8 (6,944)	11.4 (1,667)	27.7 (4,028)	46.4 (6,736)
Elastic Modulus[c], MPa (psi)	4.70 (683)	14.8 (2,148)	27.1 (3,940)	4.41 (640)	14.8 (2,148)	26.2 (3,812)
Shear Modulus, MPa (psi)	2.43 (353)	9.57 (1,389)	17.2 (2,498)	2.43 (353)	11.0 (1,605)	17.2 (2,498)

[a] Kulhawy and Mayne (1990)
[b] Webb (1969)
[c] Begemann (1974)

2.5.4.2.3 Foundation Interfaces

FSAR Section 2.5.4.3 describes the site layout, plant orientation, surface conditions, and other site details. The applicant located LNP Units 1 and 2 in previously underdeveloped, vegetated areas. The soil profile overlying the Avon Park limestone formation consists of 3 distinct soil layers, S-1, S-2 and S-3. The soil layers were differentiated based on the SPT N-value results, which measure the penetration resistance of the soil over the sampling interval, typically 0.5 m (1.5 ft). Penetration resistance is an index of the compactness of the layer, when other factors such as overburden pressure hammer energy are taken into account. The results of the SPT indicate layer S-1 is very loose to loose, S-2 is intermediate in compactness to S-1 and S-3, and S-3 is dense to very dense. The lower two layers, S-2 and S-3, are thought to be more compact partially due to cementation. Competent Avon Park limestone underlies the soil layers and was identified by the refusal of the SPT to penetrate the limestone. The depth to the competent

Avon Park limestone is variable across the site as shown in the cross-sections in the LNP COL FSAR.

The applicant stated that it will raise the existing ground surface at an El. of 12.2 to 13.4 m (40 to 44 ft) to a final site grade at an El. of 15.5 m (51 ft). The applicant included provisions for drainage in the site grading plan. SER Figure 2.5.4-1 shows the boring locations within the LNP Units 1 and 2 power blocks. SER Figure 2.5.4-2 shows the geologic interpretation at the LNP Unit 1 Plant North-South cross-section with the soil layers, S-1, S-2 and S-3 overlying the Avon Park limestone layers, SAV-1, SAV-2 and SAV-3, at the south reactor site. Similar figures were provided in the LNP COL FSAR for LNP Unit 2, the north reactor site, where the Avon Park limestone was subdivided into 4 layers, NAV-1, NAV-2, NAV-3 and NAV-4. The limestone subdivision was based on the results of the geophysical testing, primarily the results of the suspension P-S velocity logging survey.

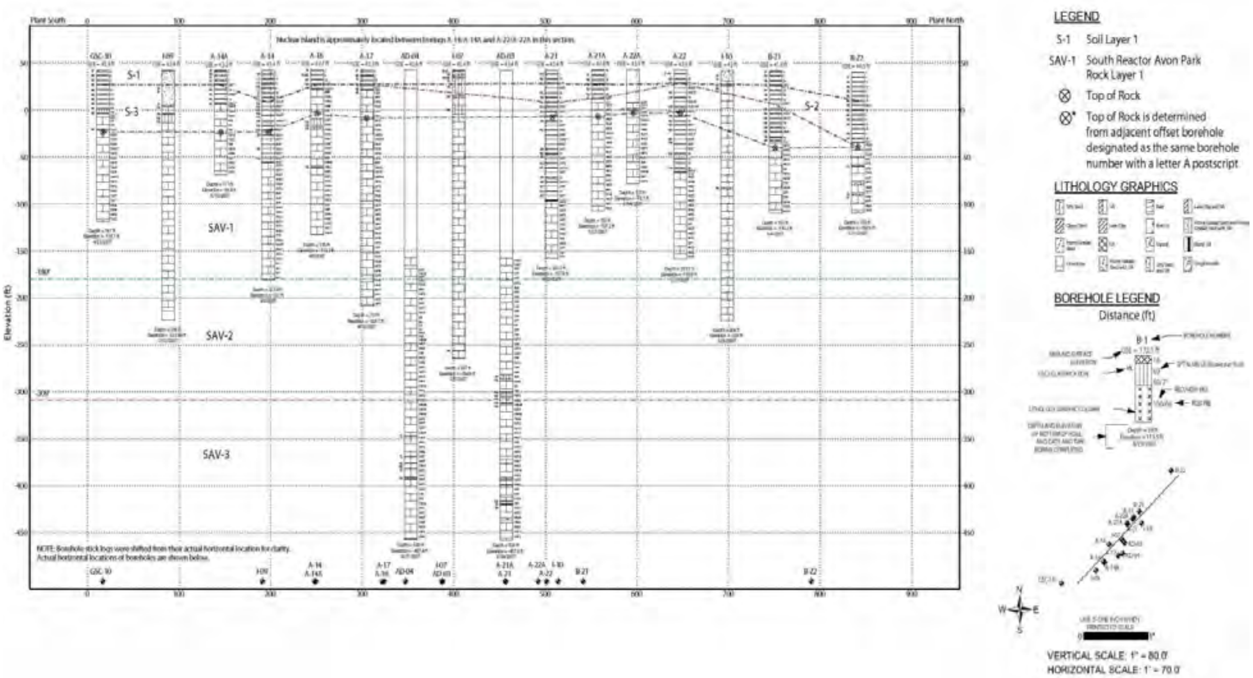


Figure 2.5.4-2. LNP 1 Plant North-South Cross-Section
(FSAR Figure 2.5.4.2-202A)

2.5.4.2.4 Geophysical Surveys

FSAR Section 2.5.4.4 discusses the scope, objectives, and results of the borehole geophysical survey methods performed at the LNP site.

2.5.4.2.4.1 Descriptions of Borehole Geophysical Surveys

FSAR Section 2.5.4.4.1 describes the seismic and non-seismic geophysical survey methods used to characterize the soil and rock properties. The applicant used two phases of suspension P-S velocity logging surveys as the primary data source to characterize the dynamic properties of soil and rock at the LNP site. The applicant also used the acoustic televiewer surveys to assess the verticality of all boreholes, obtain acoustic images of the borehole walls, and identify the dip and orientation of bedding planes and fractures. Downhole V_s surveys were completed to confirm the suspension P-S velocity logging survey results. The applicant also performed natural gamma, gamma-gamma, neutron-neutron, and induction surveys to acquire additional data to assist in the characterization of the subsurface. From this data, the applicant observed differences in soil and rock type, density, porosity, permeability, and pore fluid composition along the boring depth and between borings.

2.5.4.2.4.2 Geophysical Survey Investigation Results

FSAR Section 2.5.4.4.2 summarizes the results obtained from the various borehole geophysical surveys performed, including V_s and V_p profiles, lithological interpretations, and material property assessments.

2.5.4.2.4.3 Suspension P-S Velocity Logging Surveys

The following sections summarize FSAR Section 2.5.4.4.2.1, including the results obtained from the suspension P-S velocity logging surveys at the South, LNP Unit 1, and North, LNP Unit 2, sites.

LNP Unit 1 (South Reactor)

In the soil overburden above the top of rock, soil layers S-1, S-2 and S-3, the applicant measured V_s values between 380 to 1,410 meters per second (m/s) (1,250 fps to 4,630 fps) and observed a gradual transition from low V_s soil to high V_s rock at depths of 16.7 to 24.4 m (55 to 80 ft). The applicant identified three rock layers at LNP Unit 1: SAV-1 from top of rock down to an El. of -54.9 m (-180 ft); SAV-2 from an El. of -54.9 to -94.2 m (-180 to -309 ft); and SAV-3 from an El. of -94.2 to -139.6 m (-309 to -458 ft). SER Figure 2.5.4-3 shows the LNP Unit 1 East–West seismic profile obtained from the suspension P-S velocity logging surveys, seismic data interpreted from downhole seismic surveys, and other geotechnical data.

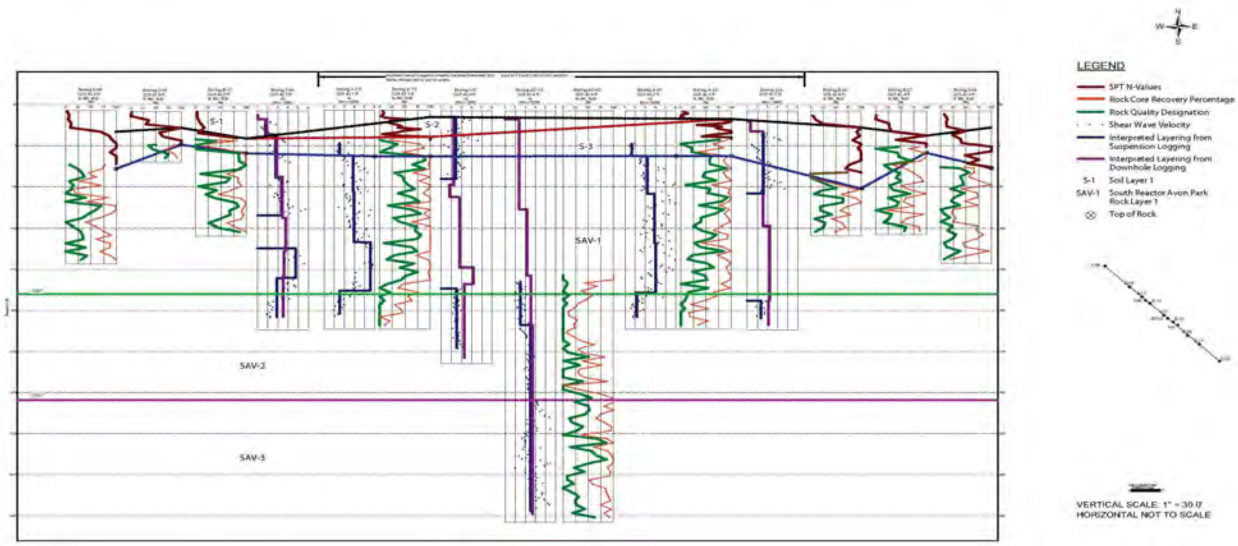


Figure 2.5.4-3. LNP1 East-West Shear Wave Velocity Profile
(FSAR Figure 2.5.4.2-204B)

LNP Unit 2 (North Reactor)

The applicant stated that for soils above top of rock the V_s ranges from 190 to 1,311 m/s (620 to 4,300 fps) with the transition from low V_s soil to high V_s rock at an approximate depth of 12 m (39.4 ft). The applicant identified four rock layers at LNP Unit 2: NAV-1 from top of rock down to an El. of -29.6 m (-97 ft); NAV-2 from an El. of -29.6 to -45.1 m (-97 to -148 ft); NAV-3 from an El. of -45.1 to -92.3 m (-148 to -303 ft); and NAV-4 from an El. of -92.3 m to -139.6 m (-303 to -458 ft). The applicant concluded that the suspension P-S velocity logging surveys in uncased boreholes below depths of 15.2 m (50 ft) produced good quality velocity profiles. However, the results obtained at shallower depths are inconsistent due to the presence of the borehole casing, and erosion and collapse of the borehole walls during drilling. The applicant observed that the rock V_s measured at LNP Unit 1 is lower than at LNP Unit 2, and noted a greater variation in V_s measurements in SAV-1 than in NAV-1 and NAV-2. SER Figure 2.5.4-4 shows an east to west geophysical and geological cross-section underlying Unit 2. The various soil and limestone layers are designated in this figure along with the measured V_s profiles.

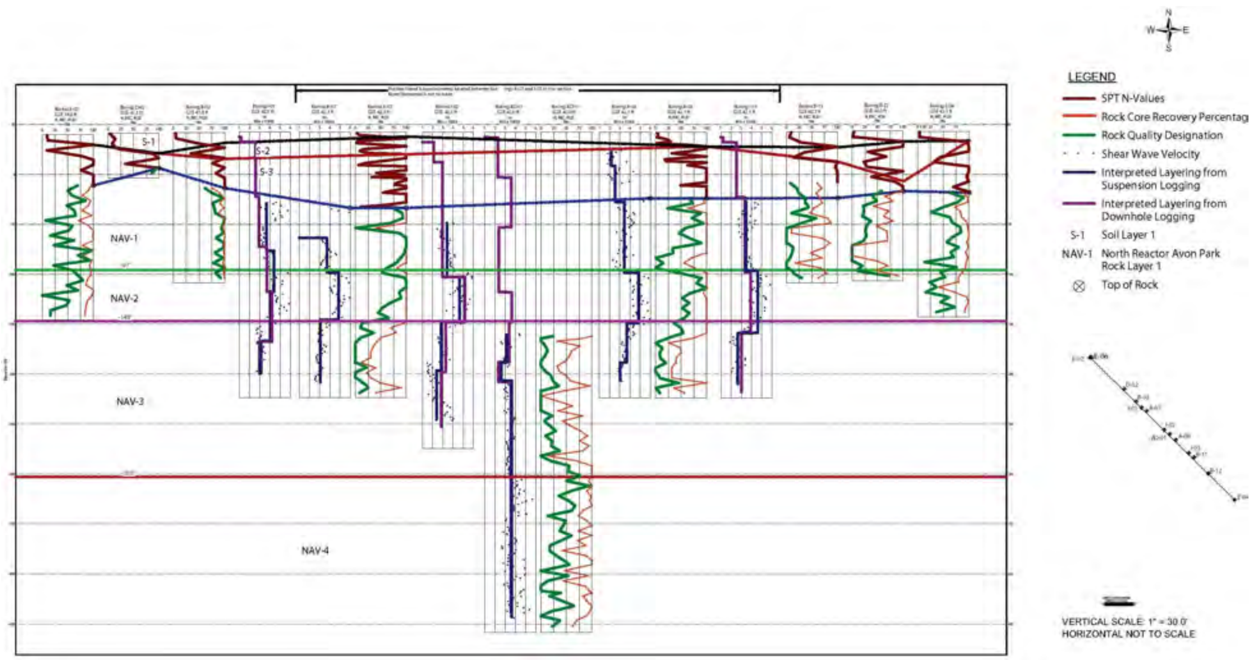


Figure 2.5.4-4. Subsurface Fence Diagram with V_s Results at LNP 2 Plant East to West (FSAR Figure 2.5.4.2-205b)

2.5.4.2.4.4 Acoustic Televiewer Surveys

The applicant used acoustic televiewer surveys to measure borehole deviation and image fractures, bedding planes, and eroded areas along the borehole walls. The applicant identified two vertical open fractures, which it considered a significant occurrence because of the rarity of intersecting vertically-oriented joints while drilling a vertically-oriented borehole.

2.5.4.2.4.5 Downhole Velocity Surveys

The applicant stated that the suspension P-S velocity logging method is more precise than the downhole method, and therefore used the results of the suspension P-S velocity logging in the engineering analyses and the downhole results for confirmation of the suspension P-S velocity logging results.

2.5.4.2.4.6 Natural Gamma Log

The applicant noted the increased clay content in the soil deposits above the top of rock at LNP Units 1 and 2, which was used as one marker in delineating the top of rock. The applicant indicated that a shallow, more weathered clayey profile exists at LNP Unit 1 than at LNP Unit 2. At LNP Unit 2, the applicant observed that the borings generally show a uniform higher natural gamma response with the exception of one borehole, which exhibited a response 1.5 times larger than the response found in the other borings. The applicant postulated the presence of more clay in this boring.

2.5.4.2.4.7 Gamma-Gamma (Density) Log

The applicant used these results to determine the presence of soil infill, which can be correlated to poor rock quality. However, the applicant was unable to correlate the gamma-gamma logs to the drilling logs, which may indicate that the low density zones identified in the gamma-gamma logs and the karst features reported in the core logs are limited in extent. The applicant concluded that all of the significant low density zones occur no deeper than 61 m (200 ft) below the existing ground surface.

2.5.4.2.4.8 Neutron-Neutron (Porosity) Log

The applicant stated that low neutron-neutron values indicate an increase in porosity and lower density, while higher values indicate a decrease in porosity and higher density. The applicant stated that the porosity is lower at LNP Unit 1 than at LNP Unit 2. The applicant identified a relatively lower porosity zone at depths between 42.6 and 57.9 m (140 and 190 ft) below the existing ground surface that is broader at LNP Unit 1 and more distinct at LNP Unit 2.

2.5.4.2.4.9 Induction (Conductivity) Log

The applicant related higher conductivity readings to increased clay content or pore fluids having an increased specific conductance. At LNP Unit 1, the applicant measured high conductivities between depths of 27.4 and 56.3 m (90 to 185 ft) below the existing ground surface, which it concluded were randomly distributed localized thin features. At LNP Unit 2, the applicant found that the conductivities in the upper 30.4 m (100 ft) are more uniform than those occurring at LNP Unit 1. A thin, high conductivity zone occurs in the LNP Unit 2 logs between depths of 27.4 to 28.9 m (90 to 95 ft).

2.5.4.2.4.10 Criteria for Use of Geophysical Survey Results as Design Parameters

The applicant used the suspension P-S velocity logging data as the primary source of V_s and V_p data for the engineering analyses. The applicant used acoustic televiewer, caliper and deviation survey data for borehole verticality checks, lithologic and stratigraphy determinations, and examinations of fractures. The non-seismic geophysical tools provided data that the applicant used to rule out continuity of voids from boring to boring.

2.5.4.2.5 Excavations and Backfill

FSAR Section 2.5.4.5 describes the applicant's plans for the excavation and backfill of the nuclear islands, including the planned diaphragm wall, excavation extents, and assumed properties of concrete backfill to be placed underneath and adjacent to the safety-related structures.

2.5.4.2.5.1 Diaphragm Walls and Grouting

FSAR Section 2.5.4.5.1 discusses the purpose of the diaphragm walls and grouting. The applicant stated that the diaphragm walls will serve as a temporary excavation support system to facilitate the excavation from the existing ground surface down to an El. of -7.3 m (-24 ft). The applicant noted that the diaphragm walls in combination with the grouted portion of the foundation, will aid construction dewatering by reducing seepage rates into the excavation. SER Figure 2.5.4-5 shows the extent of the excavation and diaphragm wall in the plan for LNP Unit 1. The excavation limits and diaphragm wall are coincident. SER Figure 2.5.4-6 shows a cross-section of the LNP Unit 1 diaphragm wall and grouting limits. Also shown on this figure is the RCC bridging mat and pier-supported seismic Category II and nonsafety-related structures that surround the nuclear island.

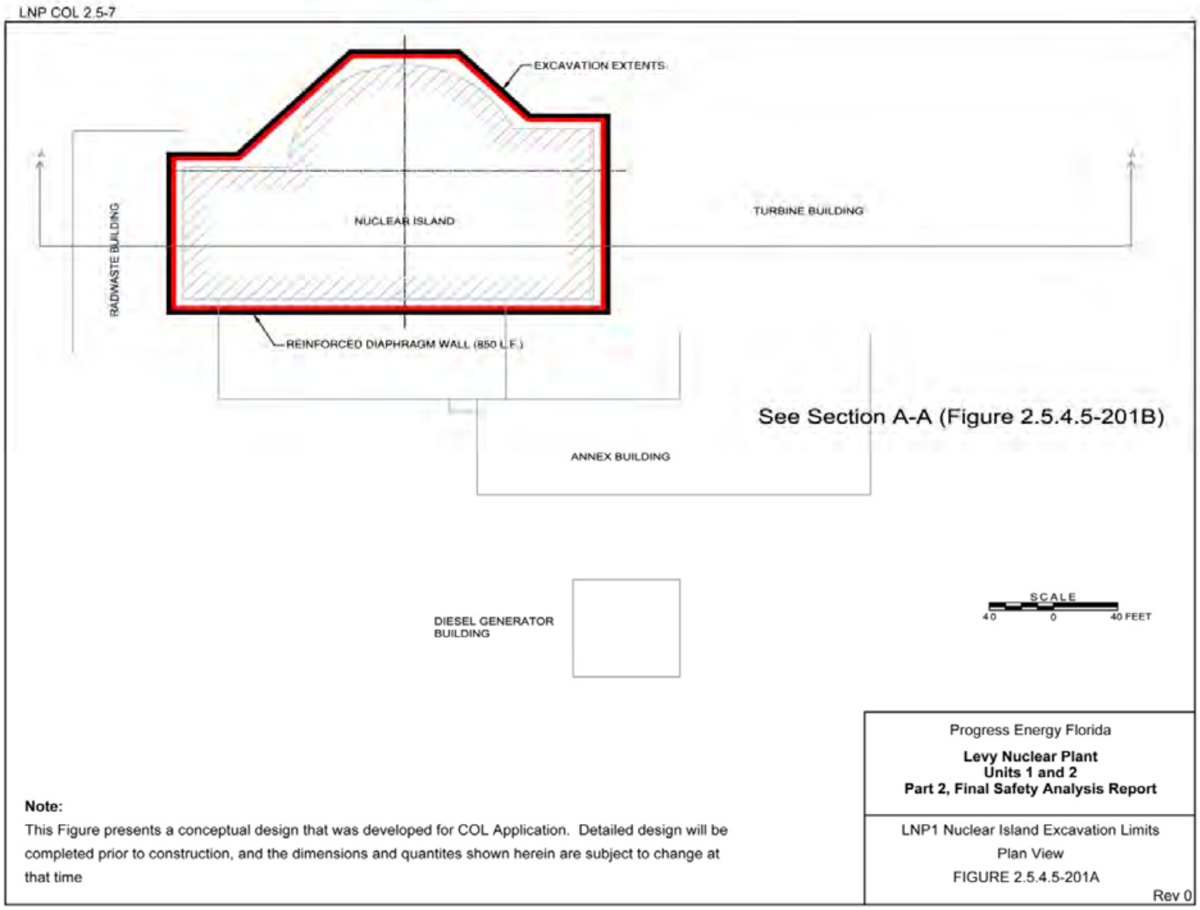


Figure 2.5.4-5. Plan View of LNP Unit 1
(FSAR Figure 2.5.4.5-201A)

2.5.4.2.5.1.1 Perimeter Diaphragm Wall

The diaphragm wall will be constructed prior to commencing excavation. The applicant stated that it will use hydrofraise equipment to excavate and key the diaphragm walls approximately 9.1 m (30 ft) into competent rock. The hydrofraise equipment consists of a crane hoisted drilling machine that cuts a vertical slot down to the desired depth through soil and rock. The wall is excavated in alternating panels that are initially supported by the drilling fluid and subsequently backfilled with tremie concrete. The wall will be tied-back by rows of pre-stressed anchors spaced 3 m (10 ft) on center. The anchor pull out resistance will be developed by grouting the anchors into the Avon Park limestone. The wall will be constructed of 1.06 m (3.5 ft) thick of concrete with compressive strength of 27.6 MPa (4,000 psi) and reinforced with a steel reinforcement cage. The diaphragm wall will serve as an excavation support system and vertical seepage barrier.

2.5.4.2.5.1.2 Permeation Grouting

In order to decrease the permeability of the uppermost layer of the Avon Park limestone, the applicant plans to inject grout from an El. of -7.3 to -30.1 m (-24 to -99 ft) within the limits of the diaphragm wall (see SER Figure 2.5.4-6) to fill voids associated with joint sets and bedding planes. Acting together, the diaphragm wall and the grouted limestone formation will form a "bathtub" and slow ground water seepage into the excavation for the foundation. The grouted section will also reduce the potential for future solution activity by cutting off flow paths. The applicant worked out the details of the methodology and materials planned for the production grouting during a grout test program, which is discussed below.

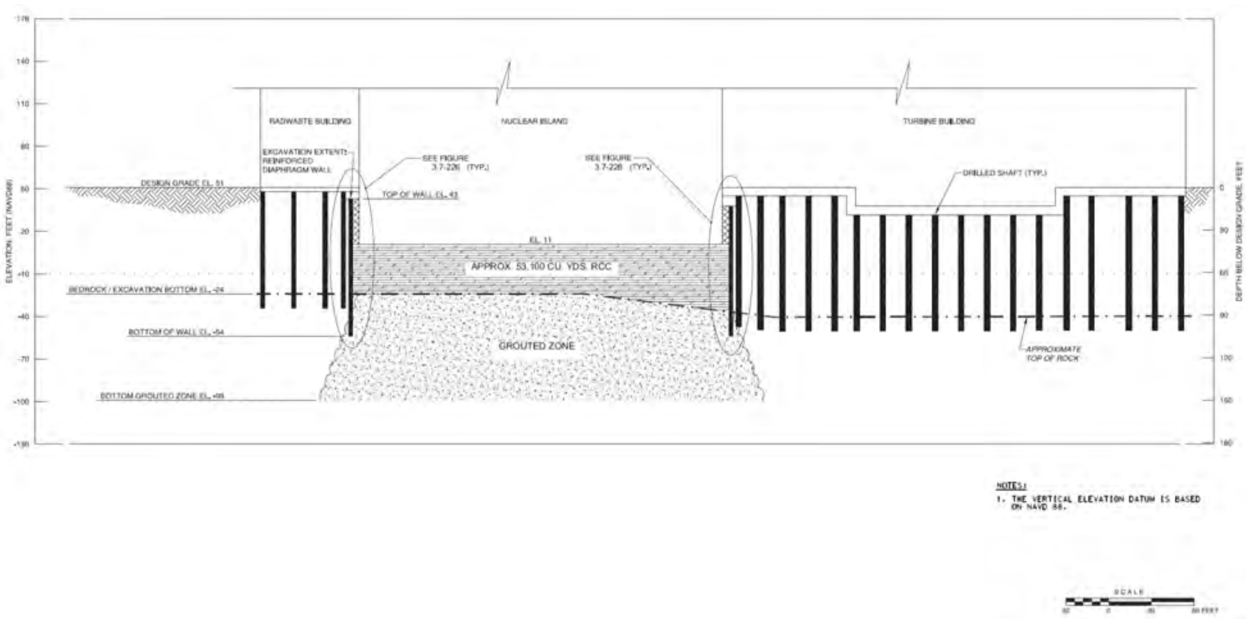


Figure 2.5.4-6. Profile View of LNP 1
(FSAR Figure 2.5.4.5-201B)

2.5.4.2.5.1.3 Grouting Method

The applicant stated that the primary method for the grouting operation will consist of grouting through boreholes, including angled boreholes, to intercept vertical joints. The applicant will perform the grouting using the upstage method with pneumatic packers when possible, and a suite of grout mixes of varying viscosities. Where necessary, the applicant will use downstage grouting methods to prevent borehole collapse. The applicant plans to space primary grout holes on 4.8 m (16 ft) centers and split space to 2.4 m (8 ft) centers. Decisions regarding the use of tertiary boreholes on 1.2 m (4 ft) centers will be determined during the production grouting program. Based on the results of the grout test program, the applicant established grouting pressures of 11.3 kilopascals (kPa) per meter (0.5 psi per foot) of depth during production grouting.

2.5.4.2.5.1.4 Grout Test Program

The applicant performed a grout test program to validate the grout mix design, grouting pressures and grouting techniques; measure any change in VS due to grouting; evaluate the permeability within the grouted zone; and determine the grout take prior to construction. The applicant grouted outside the footprint of the nuclear island, between 42.9 and 20.1 m (141 and 66 ft) below the surface primarily using vertical holes. This interval coincides with the intended grout zone during production. Using state-of-the-art monitoring equipment, the applicant determined best practice grouting pressures, grout mixes, and other grouting criteria. The results of the grout test program demonstrated that a reduction in rock mass permeability to reduce seepage into the excavation to acceptable limits was achieved and the V_S of the limestone was not appreciably affected by the presence of the grout.

2.5.4.2.5.2 Excavation Extents

FSAR Section 2.5.4.5.2 discusses the extent of the excavations, which are within the limits of the diaphragm walls. Outside the excavation for the nuclear island, the nonsafety-related structures will be supported on drilled shaft foundations socketed into rock (see SER Figure 2.5.4-6). The applicant plans a pilot hole at each drilled shaft location to determine the bearing depth at the base of the drilled shaft. The applicant also plans to excavate and replace the very loose to loose near surface soils to a depth of 2.13 m (7 ft) underlying the auxiliary buildings with engineered fill. SER Figures 2.5.4-5 and 2.5.4-6 show the conceptual plans for the excavation, diaphragm wall, grouting limits and seismic Category II and nonsafety-related structures surrounding the nuclear island.

2.5.4.2.5.3 Excavation Methods and Subgrade Improvement

FSAR Section 2.5.4.5.3 describes the methods for excavation and subgrade improvement. The applicant identified an El. of -7.3 m (-24 ft) as the depth at which materials with the most desirable properties for foundation stability were encountered. The in-situ rock at this elevation needs to be moderately to highly cemented, without solution features, loose rock or open or soil-filled joints or fractures. The applicant plans to remove and replace, or improve, foundation conditions that do not meet the design criteria. Excavation will be by ordinary means using earth moving equipment within the diaphragm walled area. The excavation will be incremental to allow geologic mapping and installation of anchors in the reinforced concrete diaphragm wall. Once the excavation reaches an El. of -7.3 m (-24 ft), the applicant will prepare the surface of the Avon Park limestone by removing loose rock from the surface and excavating soil from open joints. The applicant plans to use dental concrete as backfill in all open joints and as a leveling course for the RCC placement.

2.5.4.2.5.4 Properties of Backfill Beneath and Adjacent to the Nuclear Island

FSAR Section 2.5.4.5.4 discusses the backfill properties beneath and adjacent to the nuclear islands. The applicant plans to replace unsatisfactory soils with a 10.7 m (35 ft) thick RCC bridging mat bearing on the surface of the prepared Avon Park Formation. The applicant stated that the RCC bridging mat provided a uniform subgrade for the nuclear island mat foundation and the capability to bridge potential karst features. Between the diaphragm wall and nuclear island basemat, the applicant plans to use a concrete-like controlled low strength material (CLSM) as backfill. SER Figure 2.5.4-7 shows the location of the CLSM, and SER Table 2.5.4-3 summarizes the characteristics assumed for both the RCC and the CLSM. The applicant plans to develop further specifications for each backfill type and quality tests prior to construction.

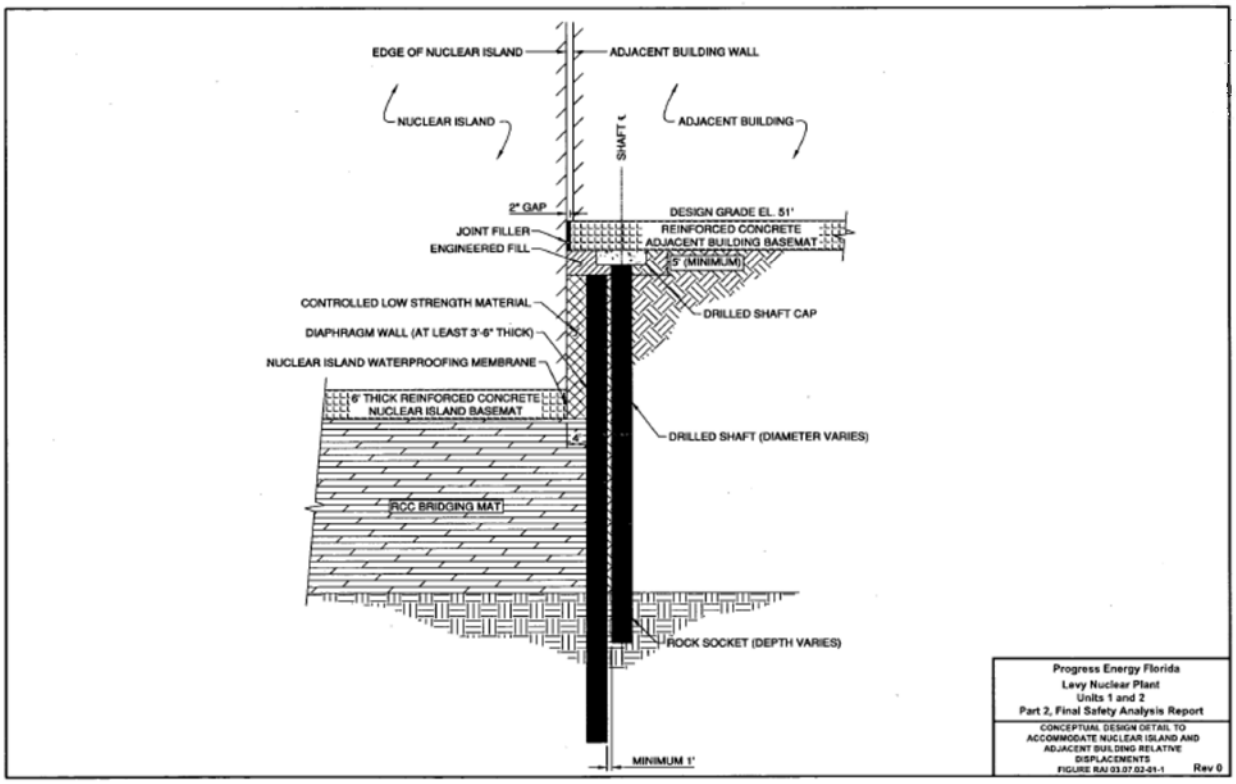


Figure 2.5.4-7. Detail Showing Location of CLSM Between Nuclear Island and Diaphragm Wall (RAI Figure 03.07.02-01-1)

Table 2.5.4-3. As-Built Engineering Properties of Backfill and Structural Fill (FSAR Table 2.5.4.5-201)

Backfill Type	Strength Parameters, MPa (psi)	V_s, m/s (fps)
RCC Bridging Mat	1-Year Compressive Strength: 17.2 (2,500)	1,066 (3,500)
CLSM Backfill	28-Day Compressive Strength: 3.4 (500)	304 ,000)

2.5.4.2.5.4.1 Roller Compacted Concrete Mat Test Pad

The applicant plans to construct a RCC test pad in order to define the material properties and develop the quality control requirements. The applicant stated that among the properties to be tested are mix design, material control testing, strength testing, concrete placement, and field-testing. The applicant also stated that the results of these tests will ensure that all material property specifications are met and the RCC test pad has the same specifications as in FSAR Section 2.5.4.5.4.

2.5.4.2.6 Ground Water Conditions

FSAR Section 2.5.4.6 summarizes the pre- and post-construction ground water elevations and the preliminary plans for construction dewatering. Also in this section, the applicant summarized the existing groundwater table, which ranges from 0.3 to 1.5 m (1 to 5 ft) below the existing ground surface. The applicant concluded that the post-construction ground water elevation at the LNP site is not expected to rise above an El. of 14.6 m (48 ft) below the final site grade at an El. of 15.5 m (51 ft). The applicant referred to FSAR Section 2.4.12.5 for additional details on the groundwater conditions at the site.

2.5.4.2.6.1 Construction Dewatering

FSAR Section 2.5.4.6.2 discusses the estimated construction dewatering flow rates and dewatering methods for LNP Units 1 and 2. The applicant determined that the diaphragm walls will minimize the lateral ground water flow while grouting of the Avon Park Formation will minimize upward seepage and resist uplift pressure. The applicant used MODFLOW 2000 to model the proposed excavation and observe the expected upward gradients and ground water flow rates into the excavation.

To account for variations in the effectiveness of the grout, the applicant varied the gross permeability of the grouted sections in the model. Permeability of the ungrouted sections was based on hydraulic conductivity field tests. The applicant plans to evaluate the exposed subgrade rock and eliminate any significant leakage through a second round of grouting. The applicant also plans to employ a ground water monitoring program during construction to measure the head differential inside and outside of the diaphragm walls and the uplift pressure across the bottom of the excavation.

2.5.4.2.7 Response of Soil and Rock to Dynamic Loading

FSAR Section 2.5.4.7 summarizes the response to dynamic loading for both soil and rock at the LNP site. Because ground motions at the site are low, the applicant did not perform dynamic triaxial shear tests or resonant column torsional shear tests but instead accounted for any uncertainty in modulus and damping relationships by assuming a range of behaviors for the softer layers using two sets of EPRI curves for the site response analysis. The applicant also stated that the potential for non-tectonic deformation is negligible.

2.5.4.2.8 Liquefaction Potential

FSAR Section 2.5.4.8 discusses the potential for liquefaction at the LNP site. The applicant computed the factor of safety (FS) against liquefaction generated by the SSE following the guidance provided in RG 1.198, which recommends using the method of analysis described by Youd et al. (2001).

2.5.4.2.8.1 Soil and Ground Water Conditions

FSAR Section 2.5.4.8.1 discusses the soil conditions at LNP Units 1 and 2 at the time of exploration and employed in the liquefaction analysis. The soil profile consists of loose to very dense sands and silts and some clay overlying the Avon Park Formation. The applicant noted that ground water is typically within 0.9 m (3 ft) of the existing ground surface; the existing ground surface being at approximately an El. of 13.1 m (43 ft). The applicant noted that liquefaction below the nuclear island is not possible as the nuclear island will be founded on RCC overlying the Avon Park limestone. Because the soils outside the diaphragm wall are potentially liquefiable, the applicant included them in the liquefaction analysis, with the exception of the top 2.1 m (7 ft) of soils, which will either be removed or improved as described previously in Section 2.5.4.2.5.3 of this SER.

2.5.4.2.8.2 Liquefaction Analysis Procedure

FSAR Section 2.5.4.8.2 describes the liquefaction analysis procedure, specifically the calculation of the factor of safety against liquefaction, which is a function of cyclic stress generated by the SSE compared to the dynamic strength of the soils. In accordance with RG 1.198, the applicant considered cohesionless soils with FS less than or equal to 1.1 liquefiable, and soils with FS greater than 1.4 to be non-liquefiable. For soils with FS in the intermediate range, greater than 1.1 but less than 1.4, the applicant considered the deleterious effect of increased dynamic pore pressures on the strength of the soil.

2.5.4.2.8.3 Results of Liquefaction Analysis

FSAR Section 2.5.4.8.5 discusses the results of the liquefaction analyses, which show that some random near surface soils beyond the limits of the nuclear island may experience liquefaction. The applicant stated that the presence of random liquefied zones outside of the nuclear island would not interfere with the AP1000's basemat stability with regard to sliding. The applicant based this conclusion on the fact that the liquefied zones are either isolated, negligible, outside the zone that provides resistance to sliding, or will be excavated and replaced with non-liquefiable material. In addition, the applicant stated that the earthwork design incorporates vertical and horizontal drains to prevent buildup of excess pore pressures that cause liquefaction. The applicant also stated that the design of the drilled piers will account for the random liquefied zones such that the lateral stability of the drilled piers will not be affected. The drilled piers support the seismic Category II and nonsafety-related structures and are reviewed in LNP SER Sections 3.7 and 3.8.5.

2.5.4.2.9 Earthquake Site Characteristics

In FSAR Section 2.5.4.9, the applicant referred to FSAR Sections 2.5.2.5 and 2.5.2.6 for a discussion of the methods used to calculate the site amplification at the GMRS elevation and the FIRS.

2.5.4.2.10 Static Stability

FSAR Section 2.5.4.10 discusses the analyses performed to assess the foundation bearing capacity, sliding, foundation settlement, and lateral pressures against below-grade walls.

2.5.4.2.10.1 Bearing Capacity

FSAR Section 2.5.4.10.1 states that the bearing capacities obtained under static and dynamic loading conditions satisfy the safety requirements set forth in the AP1000 DCD.

2.5.4.2.10.1.1 Bearing Capacity Analysis Methodology

The applicant stated that the critical subsurface bearing material beneath each nuclear island is the RCC bridging mat. The applicant used the permissible service load stress equation from American Concrete Institute (ACI) 318-89 to determine the ultimate bearing stresses in concrete, and compared the bearing capacity of the RCC bridging mat to the bearing demand. The applicant determined FSs of 12.1 and 4.5 for the static and dynamic cases, respectively. The applicant noted that the factor of safety for the dynamic case was based on the dynamic bearing demand of 1.15 MPa (24 ksf) which envelops maximum bearing pressure of 0.97 MPa (20.29 ksf) from site-specify SSI analysis with the LNP site-specific SSE of 0.1g. The applicant concluded that the factors of safety are greater than 2.8 for the dynamic case. The applicant also indicated that the factors of safety are greater than the industry accepted factors of safety of 3 for the static case and 2 for the dynamic case.

The applicant used two procedures to determine the bearing capacity of the Avon Park limestone that supports the RCC bridging mat, the simplified American Association of State Highway and Transportation Officials (AASHTO, 2002) formulation for footings on broken or jointed rock, and the U.S. Army Corps of Engineers (USACE) formulation (USACE, 1992) for two different failure modes of rock. Additionally, the applicant used a 3D finite element method (FEM) analysis to compute the FS against bearing capacity considering the presence of postulated voids of different sizes at varied elevations and locations within the Avon Park limestone. The applicant determined that the FS against bearing capacity in the Avon Park limestone was at least 3 for the static case and 2.0 for the dynamic case.

2.5.4.2.10.2 Resistance to Sliding

FSAR Section 2.5.4.10.2 discusses the resistance of the nuclear islands to sliding. The applicant stated that it will found the RCC on Avon Park limestone that meets the design criteria and is clean of any loose material in order to achieve interlocking between the RCC bridging mat and the underlying rock. The applicant assumed zero adhesion and a friction angle of 48 to

60 degrees between the RCC bridging mat and underlying limestone, which is greater than the 35 degrees required by the AP1000 DCD.

2.5.4.2.10.3 Settlement

FSAR Section 2.5.4.10.3 discusses the settlement analyses performed for the LNP site. The applicant calculated small total and differential settlements that fall within the limits specified in the AP1000 DCD. Based on the settlement analyses, the applicant concluded that it satisfied all design criteria for foundation settlement at LNP Units 1 and 2.

2.5.4.2.10.3.1 Elastic (Total) Settlement under Foundation Loads

The applicant calculated the elastic settlement of the nuclear islands at LNP Units 1 and 2 based on the elastic properties of the Avon Park rock mass and obtained results from three methods: a 3D FEM analysis, AASHTO (2002), and elastic theory. The applicant stated that the average settlements obtained from the FEM analysis as measured at the base of the RCC bridging mat were 0.53 and 0.45 cm (0.21 and 0.18 in) at LNP Units 1 and 2, respectively. The other methods used were in agreement with the FEM analysis. The applicant stated that total settlements are likely to occur during construction, and noted that the AP1000 DCD settlement criterion is 7.6 cm (3 in).

2.5.4.2.10.3.2 Differential Settlement

Based on the settlement analysis results, the applicant determined that the maximum settlement occurs at the center of the nuclear island, and calculated a tilt of less than 1:1,200. The applicant concluded that the tilt was within the permissible differential settlement requirements of 1:1200 (1.27 cm in 15.24 m (0.5 inch in 50 ft)) allowed by the AP1000 DCD. Because the nonsafety-related buildings will be founded on drilled shafts socketed into competent rock, the applicant stated that the differential settlements between the nuclear island and the adjacent nonsafety-related buildings are negligible. The applicant planned to perform detailed settlement analyses for the surrounding nonsafety-related buildings prior to construction.

2.5.4.2.10.3.3 Subsurface Instrumentation

The applicant stated that it would monitor water levels and settlement (heave) during construction. As part of this monitoring program, the applicant stated that it will install piezometers outside the perimeter of the diaphragm walls at an El. of -7.3 m (-24 ft); and within the excavation at an El. of 0 and -8.8 m (0 and -29 ft); and below the grouted zone at an El. of -30.1 m (-99 ft).

The applicant stated that it will place settlement monitoring points at the four corners of each nuclear island and at the northernmost point of the containment building, and monitor these benchmarks before and during construction of the nuclear island basemat and sidewalls. The applicant also committed to install and monitor additional settlement points connected to the sidewalls of the nuclear islands 0.9 m (3 ft) above site grade during backfilling operations. Additionally, the applicant committed to monitor settlement after construction of the nuclear -

island until 90 percent of the expected settlement occurred. The applicant committed to establish a post-construction long-term settlement monitoring program using the settlement points established during construction.

2.5.4.2.10.4 Lateral Earth Pressures

FSAR Section 2.5.4.10.3.5 discusses the static and dynamic lateral earth pressures acting on the below-grade nuclear island sidewalls. The applicant considered the ground surface live load, crane load, pseudostatic earthquake load, hydrostatic pressure due to the water table, soil and CLSM backfill loads, and the strength of the backfill in its analysis of the lateral pressures on the nuclear island sidewalls. To minimize the soil stresses against the wall, the applicant plans to use hand-operated compaction equipment in areas adjacent to the nuclear island sidewalls. The applicant did not include the loads from adjacent structures in the lateral pressure calculation because these structures are supported by drilled piers socketed into rock.

2.5.4.2.11 Design Criteria

FSAR Section 2.5.4.11 summarizes the design criteria and methods used in the different analyses, including assumptions, and FS. The applicant compared the site-specific characteristics of bearing capacity, V_S, lateral variability and liquefaction potential to AP1000 DCD site criteria. Based on this comparison, the applicant concluded that the LNP site meets the AP1000 DCD site criteria.

2.5.4.2.12 Techniques to Improve Subsurface Conditions

FSAR Section 2.5.4.12 summarizes techniques the applicant proposed to improve subsurface conditions. To reduce the rock mass porosity and control ground water during excavation for the foundation, the applicant plans to grout the Avon Park limestone from an El. of -7.3 m (-24 ft) down to -32 m (-99 ft). The subsequent placement of a diaphragm wall penetrating 9.1 m (30 ft) into the Avon Park limestone will create a semi-impervious barrier around and below the area to be excavated for the placement of the RCC bridging mat. After dewatering the site, the applicant plans to incrementally excavate down to the Avon Park limestone at an El. of -7.3 m (-24 ft). The bottom surface of the excavation will be prepared for RCC placement by removing any loose rock or unsuitable foundation materials, and backfilling voids in the subgrade with dental concrete to level the surface. The prepared surface will receive the 10.7 m (35 ft) thick RCC bridging mat, which tops out at an El. of 3.3 m (11 ft). SER Figure 2.5.4-6 shows the East-West cross-section of LNP Unit 1 with the location of the diaphragm walls, RCC bridging mat and grouting limits.

2.5.4.3 *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 stability of subsurface materials and foundations are as follows:

- 10 CFR Part 50, Appendix A, GDC 2, "Design Bases for Protection Against Natural Phenomena," relates to the 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," applies to the design of nuclear power plant SSCs important to safety to withstand the effects of earthquakes.

- 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.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the stability of subsurface materials and foundations are given in Section 2.5.4 of NUREG-0800.

- Geologic Features: In meeting the requirements of 10 CFR Parts 50 and 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 that are sufficiently detailed to obtain an unambiguous representation of the geology.

- Properties of Subsurface Materials: In meeting the requirements of 10 CFR Parts 50 and 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.

- Foundation Interfaces: In meeting the requirements of 10 CFR Parts 50 and 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, geologic 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 COL application.

- Geophysical Surveys. In meeting 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 are presented in detail.

- Excavation and Backfill: In meeting 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) 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; (4) the impact of compaction methods are incorporated into the structural design of the plant facilities; (5) quality control methods are discussed and the QA program described and referenced; (6) control of ground water during excavation to preclude degradation of foundation materials and properties is described and referenced.

- Ground Water Conditions: In meeting the requirements of 10 CFR Parts 50 and 100, the analysis of ground water conditions is acceptable if the following are included in this section or cross-referenced to the appropriate sections in Section 2.4 of the FSAR: (1) discussion of critical cases of ground water 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; (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 ground water fluctuations as determined by periodic monitoring of 16 local wells and piezometers.

- Response of Soil and Rock to Dynamic Loading: In meeting the requirements of 10 CFR Parts 50 and 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 in the vicinity of the site; (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; (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.

- Liquefaction Potential: In meeting the requirements of 10 CFR Parts 50 and 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 Stability. In meeting the requirements of 10 CFR Parts 50 and 100, the discussions of static analyses are acceptable if the stability of all safety-related facilities has been analyzed from a static stability standpoint including bearing capacity, rebound, settlement, and differential settlements under deadloads of fills and plant facilities, and lateral loading conditions.

- Design Criteria: In meeting 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: In meeting 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, dental work, rock bolting, or anchors).

In addition, the geologic characteristics should be consistent with appropriate sections from: RG 1.28, "Quality Assurance Program Requirements (Design and Construction)" Revision 4; RG 1.132, Revision 2; RG 1.138, Revision 2; RG 1.198; RG 1.206; and RG 1.208.

2.5.4.4 *Technical Evaluation*

The NRC staff reviewed Section 2.5.4 of the LNP 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 subsurface materials and foundations. 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 LNP COL application.

This SER section presents the staff's evaluation of the geologic and geotechnical engineering information the applicant submitted in LNP COL FSAR Section 2.5.4 to address the stability of the subsurface materials and foundations at the LNP site and to resolve LNP COL Information Items 2.5-5 through 2.5-13, LNP COL 2.5-16. The staff's evaluation of LNP COL 2.5-17 is addressed in Sections 3.8 and 14.3 of this SER. The technical information presented in LNP COL FSAR Section 2.5.4 resulted from the applicant's surface and subsurface geologic and geophysical investigations performed within the site area. Through its review of LNP COL FSAR Section 2.5.4, the staff determined whether the applicant complied with the applicable regulations and conducted its investigations at an appropriate level of detail in accordance with RG 1.132, Revision 2, and RG 1.138, Revision 2.

To thoroughly evaluate the geologic, seismic and geophysical information the applicant presented, the staff obtained the assistance of geotechnical engineers at Information Systems Laboratory, Inc. (ISL) and the USACE. The staff, and its ISL and USACE contractors, visited the LNP site to review and confirm the interpretations, assumptions, calculations and conclusions the applicant presented related to the stability of subsurface materials and foundations at the LNP site.

In addition to the RAIs discussed below, which address specific technical issues related to the stability of subsurface materials and foundations of the LNP site, the staff asked several RAIs requesting clarifications and editorial corrections of figures and text associated with FSAR Section 2.5.4. The staff does not discuss these RAIs as part of its technical evaluation.

AP1000 COL Information Items

- LNP COL 2.5-5, LNP COL 2.5-6, LNP COL 2.5-7, LNP COL 2.5-8, LNP COL 2.5-9, LNP COL 2.5-10, LNP COL 2.5-11, LNP COL 2.5-12, LNP COL 2.5-13, and LNP COL 2.5-16

The staff's review of the information in LNP COL FSAR Section 2.5.4 to ensure that the COL information items were addressed satisfactorily is discussed below.

2.5.4.4.1 Geologic Features

The staff reviewed the summary of the regional and site geologic conditions, particularly the hazards that may affect the LNP site, provided in FSAR Section 2.5.4.1 as well as the description and characterization of the regional and site geology in FSAR Section 2.5.1. Section 2.5.1.4 of this SER includes the staff's technical evaluation of the regional and site geologic information. Based on the information and findings provided in FSAR Sections 2.5.4.1, 2.5.1 and 2.5.3, the staff concludes that the applicant provided adequate information regarding the geologic features at the LNP site. The detailed evaluation and staff findings with respect to the geologic features are provided in Sections 2.5.1.4 and 2.5.3.4 of this SER.

2.5.4.4.2 Properties of Subsurface Materials

The staff focused its review of LNP COL FSAR Section 2.5.4.2 on the applicant's description of the static and dynamic engineering properties of the soil and rock strata underlying the LNP site, and the methods used to determine the site engineering properties. The staff reviewed the applicant's field investigation methods and laboratory testing program used to determine the properties of the subsurface materials. The review was carried out with respect to the guidance of RG 1.132, Revision 2; RG 1.138, Revision 2; RG 1.208; and NUREG-0800 Section 2.5.4.

As stated in FSAR Section 2.5.4.1.2.1.4, both LNP nuclear islands will be supported by a 10.6 m (35 ft) thick RCC bridging mat, which will replace unsatisfactory weathered limestone between an El. of 3.35 and -7.3 m (11 and -24 ft). The RCC bridging mat will be supported by the underlying Avon Park limestone beginning at an El. of -7.3 m (-24 ft). The bearing capacity of the Avon Park limestone depends on the rock mass strength parameters, which are a function of the geologic strength index (GSI), material constant (m_i), E_{rm}, and elastic modulus reduction factor. The staff focused its review on the derivation of these material parameters to verify that the strength parameters used in the applicant's engineering analyses were conservative.

2.5.4.4.2.1 Geological Strength Index (GSI)

The staff reviewed the derivation of the GSI, an indicator of the rock mass strength and structural integrity. In RAI 2.5.4-7a, the staff asked the applicant to describe how it determined the GSI. The staff also asked the applicant to discuss how it factored joint sets, bedding planes, and low or no recovery zones into the GSI determination.

In its April 2, 2009, response, the applicant stated that for every core run, it obtained the rock mass rating (RMR) using the systems proposed by Bieniawski (1989) and Robertson (1988). To estimate the GSI, the applicant used the correlation between RMR and GSI developed by Hoek and Brown (1997) which explicitly considers joint sets and bedding planes in its determination of GSI. Specifically, the discontinuity spacing, discontinuity conditions, and orientation of the discontinuities are integral to the calculation of GSI. To account for the presence of weaker materials not recovered, the applicant applied reductions in the measured strength to those rock cores that exhibited low recovery rates. The applicant concluded that because it obtained GSI values from an extensive dataset consisting of every core run at the LNP site, and conservatively considered the no recovery zones, its determination resulted in lower-bound GSI values. The applicant subsequently used these lower-bound GSI values to determine conservative rock mass strength properties for the bearing capacity sensitivity analyses discussed in this SER Section 2.5.4.4.10. The applicant concluded that the input parameters are conservative.

In its response to RAI 2.5.4-7a, the applicant also stated its intent to gather additional data in order to evaluate the properties of materials, which were not recovered during core drilling. In a January 19, 2010, supplemental response to RAI 2.5.4-7a, the applicant stated that, based on the results of the offset boring program discussed in detail in Section 2.5.4.4.4.3, the rock mass property analysis, including the determination of GSI, is conservative.

The staff reviewed the applicant’s response to RAI 2.5.4-7, the RMR systems presented in the USACE Engineering Manual 1110-1-2908, and the GSI rating criteria presented in the Hoek-Brown method as described in Marinos and Hoek (2000). Based on the Hoek-Brown state-of-the-art method, the staff concludes that the estimated GSI is reliable because it considers joint sets and bedding planes, the condition of the discontinuities and the orientation of the discontinuities. In considering zones where core drilling did not recover rock cores, the applicant reduced the strength of the intact cores to account for the missing information in its determination of the GSI. Because the applicant later determined through the offset drilling program that the “no recovery” zones were weathered-in-place Avon Park limestone, and not voids or soil infill, the staff concludes that the applicant conservatively accounted for the presence of weaker materials. This conclusion is supported by the range in V_S measurements made in the no recovery zones, which are the same as zones where core was recovered. Accordingly, the staff concludes that the Hoek–Brown method of determining the GSI used by the applicant as described by Marinos and Hoek (2000) is acceptable.

Based on its review of the results of the offset boring program, the staff also noted that the presence of weathered limestone in the offset borings yields three very important conclusions:

(1) the no recovery zones indicated in the borings are not karst features; (2) the elastic modulus as derived from the V_s measurements is reliable; and (3) the GSI is conservatively determined. To demonstrate that the GSI is conservative, the staff consulted SER Figure 2.5.4-8, which presents a typical GSI range for limestone from Marinos and Hoek (2000).

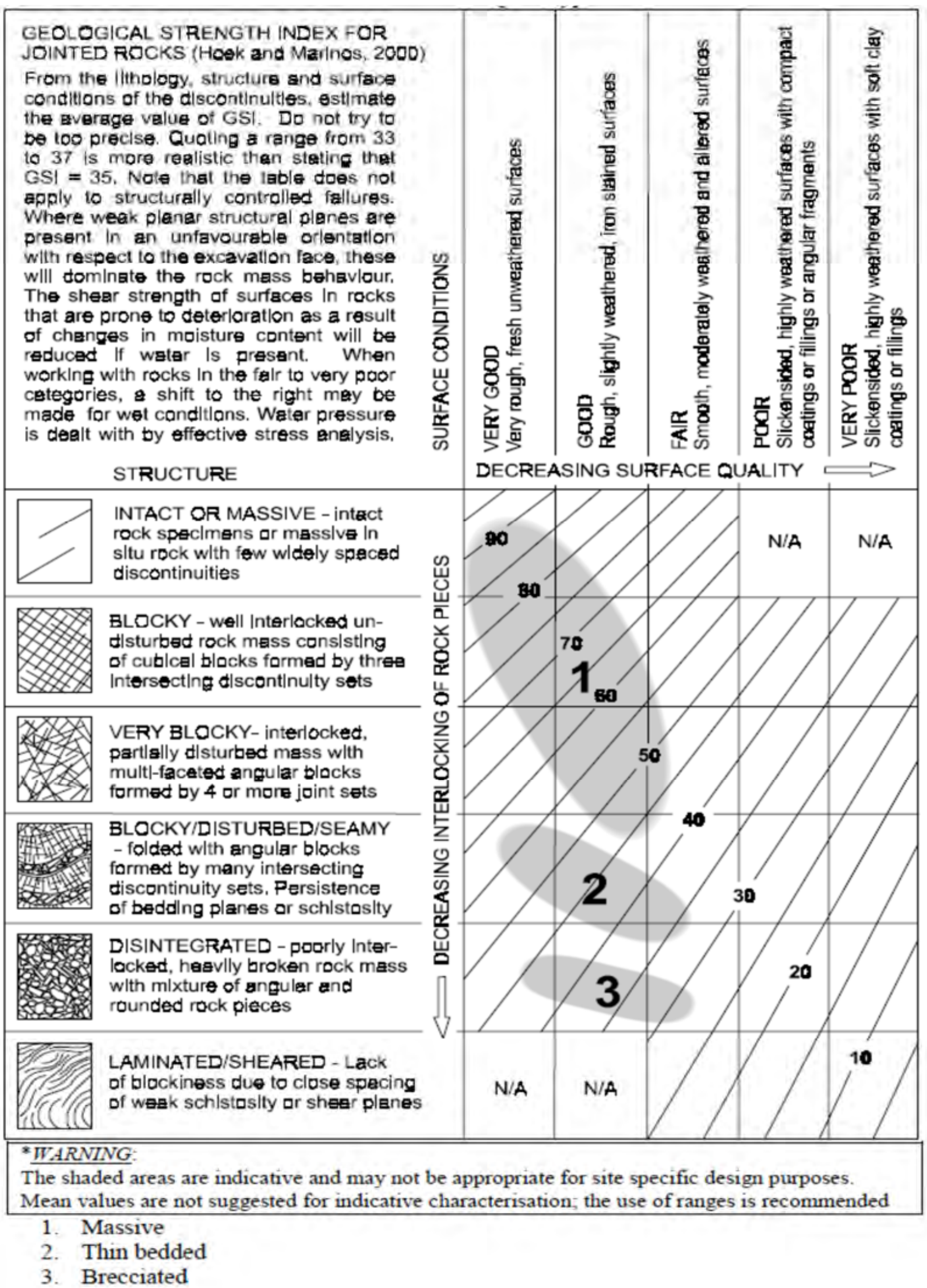


Figure 2.5.4-8. Typical GSI Factors for Limestone
(Modified from Marinos and Hoek, 2000)

In SER Figure 2.5.4-8, Marinos and Hoek (2000) show the typical limestone GSI values in the shaded zones labeled 1, 2 and 3, which range from 28 to 75 for disintegrated to blocky limestone, with fair to good discontinuity surface quality. The staff noted that most of the Avon Park Formation at the LNP site would fall in this range, with the exception of the severely weathered Avon Park limestone recovered at bedding planes and eroded vertical joints. The staff then overlaid the applicant's estimated GSI range on this figure, shown as labeled, and observed that the applicant's estimated GSI range of 21 to 38 corresponds to a disintegrated to blocky limestone with discontinuity surface quality that would be described as good to very poor. Based on this information as well as its review of the borings and other field data, the staff concludes that this is a conservative representation of the Avon Park limestone. The staff also compared these values with typical limestone GSI values, shaded areas 1, 2 and 3 on SER Figure 2.5.4-8, and concludes that the applicant's estimations of GSI values are conservative. Thus, the staff concludes that the applicant's estimated GSI values adequately represent the observed structure of the Avon Park limestone. Accordingly, RAI 2.5.4-7a is resolved.

2.5.4.4.2.2 Material Constant (m_i) Value

Because the m_i value is a material constant also used as input to the Hoek-Brown failure criteria to determine the shear strength of the rock mass, in RAI 2.5.4-14, the staff asked the applicant to justify its selection of a m_i value of 8.

In its June 8, 2009, response to RAI 2.5.4-14, the applicant stated that the recommended values of m_i for micritic limestone evolved from 8 (Hoek and Brown, 1997) to 9±2 (Marinos and Hoek, 2000) to 8±3 (RocLab 1.031, 2007). The applicant also stated that Marinos and Hoek (2000) include m_i values of 9±3 for dolomite. The applicant concluded that because micritic limestone represents the lower bound carbonate limestone m_i value the selected value of 8 is conservative.

In order to confirm the applicant's m_i estimate, the staff reviewed Marinos and Hoek (2000) and considered the published m_i values of 9±2 for micritic limestone and 9±3 for dolomite. Because much of the Avon Park limestone has been dolomitized, the staff notes that the selection of 8 represents the lower bound as shown in SER Table 2.5.4-4. Because the m_i value is a measure of the frictional properties of intact rock, the staff also considered the relationship between GSI, friction angle and m_i shown in SER Figure 2.5.4-9 for additional evidence that this m_i value is conservative. SER Figure 2.5.4-9 shows that for the range of GSI of 20 to 40 determined for the LNP site, and a conservative assumption of friction angle equal to 30 degrees, the estimated m_i would be in the range of 11 or greater. Therefore the staff concludes that the m_i value that the applicant selected is in the lower bound of the frictional strength of the Avon Park limestone. Because this value is based on the most recently published m_i estimate for micrite (RocLab 1.031, 2007), the staff concludes that the m_i value of 8 is both reasonable and conservative for the LNP site. Accordingly, RAI 2.5.4-14 is resolved.

Table 2.5.4-4. (from Marinos and Hoek, 2000)

Rock type	Class	Group	Texture			
			Coarse	Medium	Fine	Very fine
SEDIMENTARY	Clastic		Conglomerates * Breccias *	Sandstones 17 ± 4	Siltstones 7 ± 2 Greywackes (18 ± 3)	Claystones 4 ± 2 Shales (6 ± 2) Marls (7 ± 2)
	Non-Clastic	Carbonates	Crystalline Limestone (12 ± 3)	Sparitic Limestones (10 ± 2)	Micritic Limestones (9 ± 2)	Dolomites (9 ± 3)
		Evaporites		Gypsum 8 ± 2	Anhydrite 12 ± 2	
		Organic				Chalk 7 ± 2

* indeterminate range of values

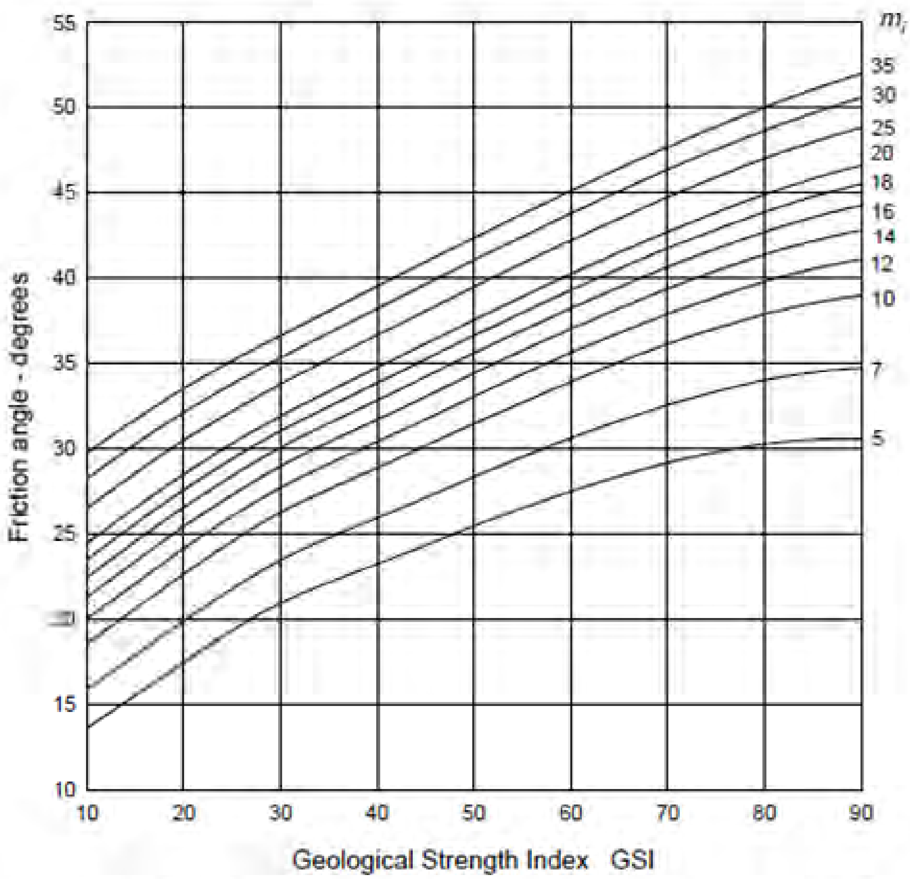


Figure 2. Friction angle ϕ for different GSI and m_i values, for depths more than 30m.

Figure 2.5.4-9. Friction Angle for Different GSI and m_i Values (from Marinos and Hoek, 2000)

2.5.4.4.2.3 Elastic Modulus Reduction Factor

The elastic modulus reduction factor is applied to the rock mass elastic modulus determined from small-strain seismic V_s measurements. The application of the reduction factor is used to estimate the elastic modulus operating at larger strains caused by static loading. The staff reviewed Deere et al. (1967) and noted that it recommended a reduction factor of 0.5 for a rock mass RQD of approximately 70 percent. The staff reviewed the RQD values for the A-series borings at the LNP Unit 2 site and questioned the justification of using a reduction factor of 0.5. In RAI 2.5.4-15, the staff asked the applicant to justify the use of a reduction factor of 0.5 in light of the Deere et al. (1967) relationship. In the same request, the staff also asked the applicant to discuss the elastic modulus values obtained from rock UCS and PMTs, since they are significantly lower than the values obtained from the V_s. The staff asked the applicant if the UCS and PMT results influenced the selection of rock mass elastic modulus values used in the design.

In its June 23, 2009, response to RAI 2.5.4-15, the applicant stated that the modulus reduction factor of Deere et al. (1967) is not applicable to the LNP site because it is an estimate based on data from high strength granite gneiss of excellent rock mass quality located within 5.4 m (18 ft) from the surface.

The applicant also explained that the depth of the PMTs was limited by the instability of open holes and was performed in only one borehole per unit. Accordingly, the applicant excluded the PMT results from the development of the elastic modulus values. The applicant also judged the Hoek-Brown factors recommended to reduce the elastic modulus based on UCS tests results to be overly conservative. The applicant also noted that since the V_s values take the site variability into account more so than the other methods, the elastic modulus values derived from the seismic measurements are the most complete account of the site variability. Therefore, the applicant concluded that the elastic modulus values derived from V_s measurements are the most representative because these values measured the highest achievable stiffness for the rock mass, including discontinuities, and a reduction factor of 50 percent accounted for the degradation of the elastic modulus due to range of deformation likely to occur at the site.

The staff reviewed the UCS test and PMT results and compared those values with the V_s-derived elastic moduli. The staff noted that there are insufficient PMT results to enable the applicant to assign material stiffness to the layers of the Avon Park Formation due to problems with keeping the borehole open during testing. Thus, the staff concludes that the PMT results could not be used for analysis purposes. The applicant noted, and the staff concurs, that similar problems limit the usefulness of the UCS test results. The staff also notes that the elastic moduli computed from the available UCS test results are typically 10 to 40 percent of the stiffness determined from the V_s results, indicating that the sampling process had a deleterious effect on the testable samples and testing unconfined samples is not representative of the in-situ stress regime.

The staff, therefore, concludes that the UCS-derived elastic modulus values were affected by sampling disturbance and unconfined testing of the samples, and concurs with the applicant that the results are not representative of the in-situ stiffness of the Avon Park limestone. The staff also concludes that the elastic moduli from the suspension P-S velocity logging surveys are the best available data to use in the engineering analyses because these data provide the most complete description of the variability at the site, represent the highest achievable stiffness measured at very small strains, and include the natural discontinuities at the in-situ effective stresses. Because the V_s were obtained in-situ at intervals of 0.5 m (1.6 ft) for the full depth of the boring, the staff notes that it provides a nearly continuous record of the stiffness of the rock mass. Furthermore, because a different rock type was used to develop the relationship proposed by Deere et al. (1967), the staff concurs that the relationship proposed by Deere et al. (1967) is not applicable to the LNP site. The staff also independently reviewed the recommendations of Mayne et al. (2002), and concludes that a reduction factor of 50 percent is adequate since it is based on a FS of 3 and is within strain levels appropriate for deformation analyses. Accordingly, the staff considers RAI 2.5.4-15 resolved.

2.5.4.4.2.4 Conclusion for Properties of Subsurface Materials

The staff reviewed the subsurface material properties, the methods used to determine those properties, and the input parameters used to estimate rock mass shear strength parameters and stiffness properties that were used as inputs in the engineering analyses. The staff observed that the applicant was conservative in its selection of the GSI, m_i, and elastic modulus reduction factor in the determination of the rock mass strength parameters. The staff therefore concludes that the use of these results in the Hoek-Brown criteria resulted in conservative rock mass strength parameters.

Based on the near continuous measurements of V_s, the staff concludes that the V_s results are the most complete picture of the in-situ conditions. Since the applicant measured the V_s profiles using the suspension P-S velocity logging methods and downhole seismic methods at LNP Units 1 and 2, and the results were consistent, the staff concludes that this proves the reliability of the V_s data. The staff concludes that the V_s data accurately characterizes the velocity profile at the LNP site, which in turn confirms the static and dynamic stiffness of the foundation materials, since those properties are derived from the V_s measurements. The use of the measured V_s and V_p to produce the maximum shear modulus and maximum elastic modulus required the applicant to apply a factor of 0.5 to reduce the elastic modulus to a value consistent with the strain level that will exist under the applied loading.

Based on its review of Mayne et al. (2002), the staff confirms that this reduction factor was supportable. Accordingly, the staff concludes that the applicant applied adequate conservatism in its selection of the material properties based on ample borings, proper sample preparation, adequate numbers of tests, redundant testing, and conservative assessments of geologic conditions at LNP Units 1 and 2. The staff further concludes that the applicant adequately addressed COL Information Item 2.5-6 and that the field and laboratory data are sufficient to determine the subsurface properties and foundation conditions in accordance with RG 1.132, Revision 2; and RG 1.138, Revision 2, and meet the criteria of 10 CFR Part 50, Appendix A, GDC-2, and Appendix S; and 10 CFR 100.23.

2.5.4.4.3 Foundation Interfaces

The staff focused its review of LNP COL FSAR Section 2.5.4.3 on the applicant's description of the topographic layout, diaphragm wall, removal and replacement of the subsurface materials down to an El. of -7.3 m (- 24 ft), RCC bridging mat, remedial grouting, and structure locations with respect to the foundation materials supporting the LNP Units 1 and 2 safety- and nonsafety-related structures.

The staff noted that many of the core runs failed to fully recover the rock core, and poor rock core recovery was a persistent occurrence across the LNP site. Due to insufficient recovery of samples of the foundation layers, the staff questioned the nature and lateral extent of the materials in the no recovery zones. Although the applicant relied on the V_s results from the suspension P-S velocity logging surveys to characterize these materials, the staff needed more information to determine if the V_s measured in the no recovery zones were representative of those materials. Therefore, the staff asked a series of questions to obtain more information about the nature of the materials that were not recovered.

2.5.4.4.3.1 Offset Boring Program

To address the staff's concerns, the applicant completed an offset boring program consisting of six newly drilled boreholes in close proximity to existing borings to better characterize the zones where material was not recovered.

The staff performed a thorough review of the offset boring program report. The borings were drilled in close proximity to A-Series borings that recorded the worst recovery, drilling four borings at LNP Unit 1 and two borings at LNP Unit 2. The offset borings, drilled to depths of 62.4 to 73.1 m (205 to 240 ft) relative to the existing surface, were offset 1.5 m (5 ft) from A-Series borings. The applicant used precision drilling tools and techniques in an effort to increase the recovery and measure the strength of the materials in the former no recovery zones. The applicant also noted the drilling time, drill bit revolution speed and drill bit thrust while coring to provide additional data that could be used to characterize the materials. The applicant also employed soil sampling and testing equipment in an effort to determine the strength of the softer materials, but this effort was largely unsuccessful as it became obvious that the softer materials were not soils and therefore not subject to soil testing techniques. The applicant replaced the double tube core barrel with a triple tube core barrel to improve recovery of the badly fractured Avon Park limestone and reduced the fluid circulation pressures from up to 3,447 kPa (500 psi) in the A-series borings to 1,034 to 2,068 kPa (150 to 300 psi) in the offset borings. The applicant also reduced the core run from 1.5 m (5 ft) down to 0.76 m (2.5 ft) to reduce the likelihood that the bottom portion of the core run was being pulverized by the upper portion of the core lodged in the core barrel. The staff observed recorded rod drops in the offset borings, which indicated the potential for voids or possibly soft materials not capable of supporting the weight of the drilling tools, but these were typically in the range of 0.06 to 0.30 m (0.2 to 1.0 ft), consistent with previous data collected and presented.

The recovery rates improved from 65 to 85 percent in the offset borings at LNP Unit 1, but because the RQD values remained essentially the same, the staff concludes that rock soundness was not the cause of the greater recovery. The O-series borings demonstrates that the low recovery rates were more closely related to drilling technique in soft rock than actual voids, and that the no recovery zones recorded in previous series borings typically resulted from weathered limestone fragments being ground up and washed away by the production drilling methods employed in the pre-offset program borings.

The results of the offset boring program also confirmed that the assumption of 13 soft zones was conservative, and that the material, which was previously postulated as soft soil infill was actually variably weathered Avon Park limestone. The staff therefore concludes that the modeling of the bedding planes with an elastic modulus of 113 MPa (16.5 ksi) in the sensitivity studies is conservative.

Finally, based on the results of the offset boring program, the staff concludes that extensive soil-filled karst features do not exist at the site, and the V_s measurements are representative of the in-place materials and can be relied upon to perform the engineering analyses. The RAIs issued to address the staff's concerns and considerations leading up to the offset boring program are detailed in the following paragraphs.

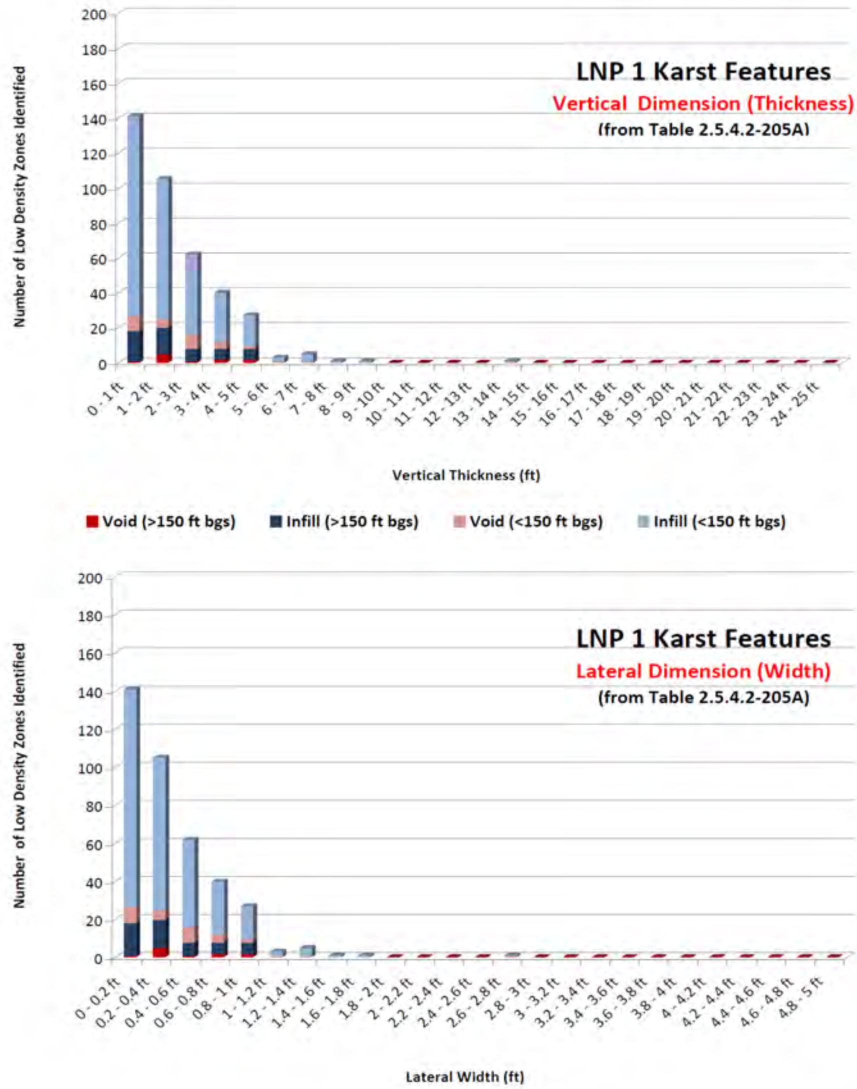
2.5.4.4.3.2 Karst Features and Voids

The staff reviewed the borings completed at the LNP site and noted that the borings revealed karst features. Accordingly, in RAI 2.5.4-1, the staff asked the applicant to justify that the boring spacing was adequate to characterize the karst features at depth, and support the conclusion of no connectivity of voids between boreholes.

In its November 20, 2008, response to RAI 2.5.4-1, the applicant stated that the potential for karst features at depth is reduced due to the nature of the karst features and the resistance of the Avon Park Formation to undergo further dissolution. The applicant characterized the karst as erosional features having a "plus-sign" morphology created by dissolution of the limestone along near-vertical fractures and at the junctures with horizontal bedding planes as the fractures dissolved. The applicant stated that the potential for ground water to dissolve limestone decreases with depth due to the reduction in the acidity of the ground water as it seeps to greater depths. Also, the applicant noted that the Avon Park Formation is highly dolomitized making it more resistant to dissolution because the dolomitic crystalline makeup inhibits the rate of karst formation.

The staff reviewed the individual borings, the geologic descriptions and driller's notes provided on the boring logs, the seismic and non-seismic geologic data, and the LNP COL FSAR tables that list the incidences of voids and soft zones encountered at the LNP Units 1 and 2 sites, respectively. The staff also reviewed the procedures the applicant used to determine the vertical and lateral dimensions of the karst features listed in the aforementioned tables and the histograms of void and soil-filled void occurrence presented in SER Figures 2.5.4-10 and 2.5.4-11 for LNP Units 1 and 2, respectively.

LNP COL 2.5-1, LNP COL 2.5-5



Progress Energy Florida
Levy Nuclear Plant
Units 1 and 2
Part 2, Final Safety Analysis Report

LNP 1 Karst Features
Vertical / Lateral Dimension
FIGURE 2.5.4.1-201A

Rev 0

Figure 2.5.4-10. Distribution of Vertical and Lateral Dimension of Voids at LNP Unit 1 Below Ground Surface (bgs) (FSAR Figure 2.5.4.1-201A)

The staff finds the method the applicant used to estimate void size acceptable. The applicant calculated the theoretical volume of the borehole and subtracted that volume from the total grout take consumed in backfilling the borehole. Depending on whether the applicant determined the void was vertically or horizontally oriented, the applicant increased the excess volume by either 50 percent for the vertically oriented voids or 100 percent for the laterally oriented voids. The total volume of excess grout volume was then applied to a specific void located in the borehole to calculate the void dimensions. The staff concludes that this approach resulted in conservative estimates of the void dimensions, because the applicant increased the volume of grout take by 50 and 100 percent as explained above.

SER Figures 2.5.4-10 and 2.5.4-11 show the distribution of voids and soil-filled voids observed in the borings above and below 45 m (150 ft) below the ground surface. These figures illustrate that the majority of the karst features are infilled. The applicant determined from the offset boring programs that what was initially postulated as infilled voids is now recognized as being severely weather Avon Park limestone, hence the frequency of postulated voids would be dramatically reduced in this figure. Some actual voids noted by rod drops were observed in the exploration borings preceding the drilling of the offset borings and are accounted for in SER Figures 2.5.4-10 and 2.5.4-11. The staff also observed that the frequency of occurrence of voids is greatest at LNP Unit 1 and typically occurs above a depth of 45 m (150 ft). The offset boring program, which was drilled with greater precision, also had some rod drops. The staff noted that these rod drops could either represent actual voids or very soft soils, but whatever the case, the vertical drops were small, typically less than 0.3 m (1 ft) in height. Based on SER Figure 2.5.4-11, the staff further observed that the largest postulated soil filled void has a vertical dimension of 6 m (19.5 ft). This karst feature was encountered at LNP Unit 2 in boring A-11, which is within the footprint of the nuclear island between the depths of 70.4 and 76.3 m (231 and 250.5 ft). The A-11 boring log does not indicate rod drops, and notes that the drilling time throughout this interval was 2 to 3 minutes. Since there was no recovery, the applicant included it as a postulated soil-filled void, but with the better understanding obtained from the offset boring program, the applicant stated the more likely explanation is that this zone is weathered, soft Avon Park limestone.

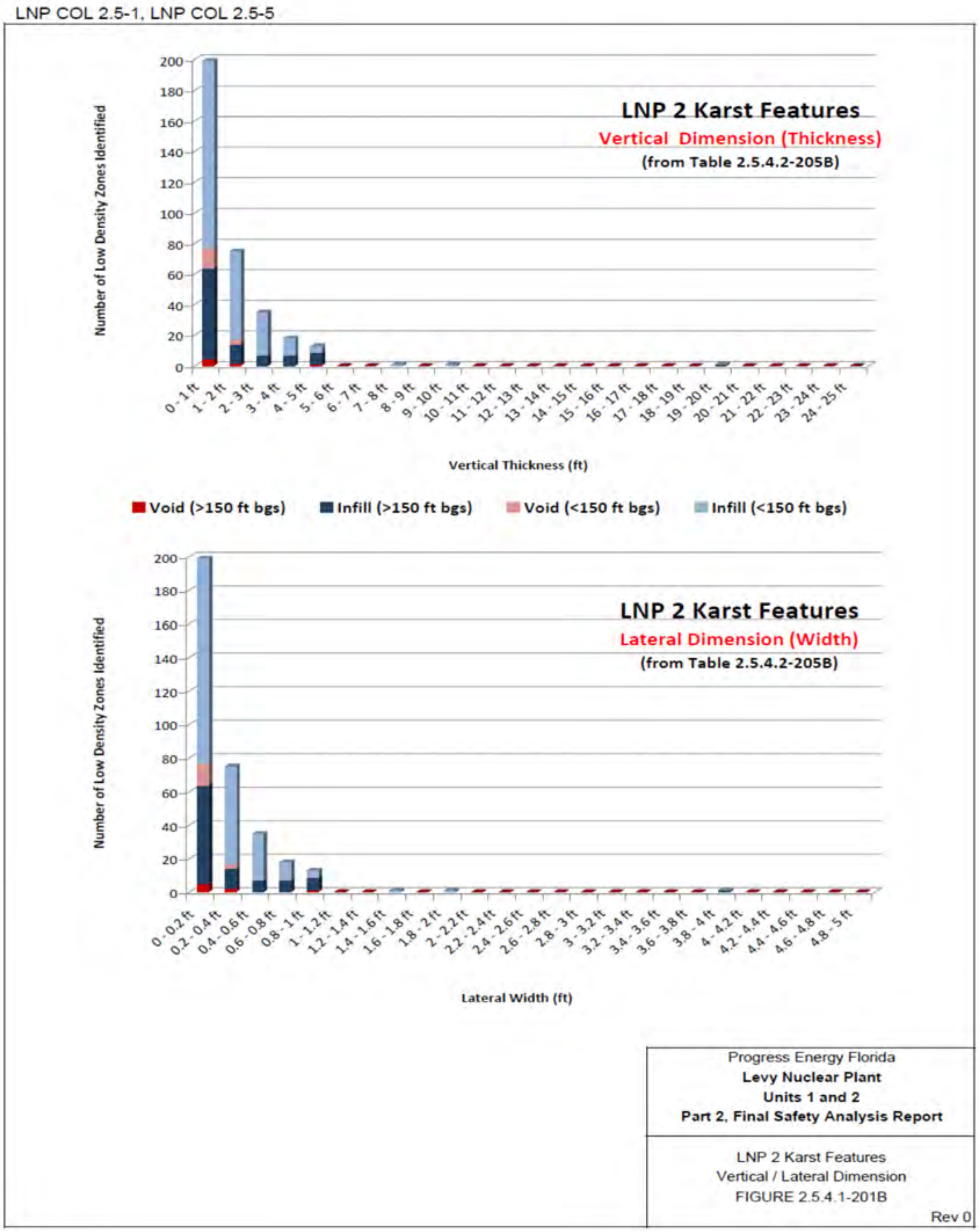


Figure 2.5.4-11. Distribution of Vertical and Lateral Dimension of Voids at LNP Unit 2 Below Ground Surface (bgs) (FSAR Figure 2.5.4.1-201B)

Given that karst in the region commonly developed in association with the "plus-sign" morphology, in which most dissolution occurs along vertical joints and at intersections of the joints with bedding planes, the staff concludes that much of the void development is limited to vertical joints and junctures between the vertical joints and bedding planes. Many borings deeper than 61 m (200 ft) intersected horizontal bedding planes without yielding evidence of extensive voids, leading the staff to conclude that maximum lateral void dimensions were conservatively estimated. Also, because data shown in SER Figures 2.5.4-10 and 2.5.4-11 illustrate that karst features are predominately located within a depth of 45 m (150 ft) below the existing ground surface, the staff concludes that certain parts of the applicant's remedial ground improvement plan could potentially help to minimize concerns about extensive voids in materials underlying safety-related structures. For example, the depth range that includes most of the karst features will be grouted in the interval from 22.8 to 45 m (75 to 150 ft) and the grouted zone excavated from the ground surface down to a depth of 22.8 m (75 ft), effectively minimizing the risk of collapse due to the presence of karst features beneath the nuclear island. However, the staff recognizes that no part of the secondary, primary, or tertiary grouting programs is intended or required by the applicant to perform a safety function.

Due to the small dimensions of actual voids, the staff concludes that borehole spacing is sufficient and further assessment of the connectivity of dissolution features between boreholes is not necessary. Furthermore, given the applicant's remedial ground improvement plan combined with the reduced ability for further dissolution due to dolomitization, as well as the lack of impact of voids at depth on safety-related structures, the staff concludes that the characterization of karst features is adequate. Based on the details of the drilling program in response to RAI 2.5.4-5 and the staff's conclusion that the material in the no-recovery zones is weathered-in-place Avon Park limestone, RAI 2.5.4-1 is resolved.

2.5.4.4.3.3 Uniformity Criteria Adherence

The staff reviewed the uniformity criteria outlined in the AP1000 DCD described below and speculated that, due to the presence of karst features and highly variable RQDs, the LNP site may be non-uniform. In RAI 2.5.4-2d, the staff asked the applicant to provide a detailed explanation of how the limestone supporting the RCC bridging mat meets the uniformity requirements for subgrade reaction described in the AP1000 DCD.

The applicant stated that, consistent with the AP1000 DCD, the Avon Park limestone meets the uniformity requirements for thickness, dip and variation in V_s down to the depth of interest at 36.5 m (120 ft) below grade. The applicant noted that "beneath the RCC bridging mat, one geologic unit is uniformly present to depths beyond [47.5 m] 150 feet below grade, consistently across all boreholes within the nuclear island footprint, meeting the thickness requirement of a uniform site." The applicant also noted that the dip angle is approximately 2 degrees for both LNP Units 1 and 2, which is within the 20 degree requirement for a uniform site given in the AP1000 DCD. Finally, the applicant noted that smooth variations in the average V_s exist between borings within the Avon Park limestone layers, but the averages between borings are within the 20 percent variation allowed by the AP1000 DCD. Based on the uniformity criteria of the AP1000 DCD, the applicant concluded that the LNP site was uniform.

The staff reviewed the boring logs presented in LNP COL FSAR Appendix BB, the results of downhole and suspension P-S velocity logging surveys, and the dip of the limestone layers beneath LNP Units 1 and 2. The staff confirmed that the thicknesses of the individual layers were uniform, and that the maximum dip of any layer was on the order of 2 degrees. The staff also noted that the average V_s in any boring was within 20 percent of the average of all the borings within a given layer and this uniformity exists to at least 36.5 m (120 ft) below grade. The staff compared these results to the AP1000 DCD and concludes that the site meets the uniformity criteria set forth in the DCD. Accordingly, RAI 2.5.4-2d is resolved.

2.5.4.4.3.4 Drilling Methods

The staff reviewed the subsurface exploration plan, including the applicant's extraction of 1.5 m (5 ft) long rock cores at various depths in the subsurface in order to obtain the RQD and recovery data. The staff noted that although the drilling time was recorded, the logs did not record the thrust or rotational speed of the drill bit, which would assist in characterizing the materials not recovered. In RAI 2.5.4-5, the staff asked the applicant to provide the drilling pressures that coincide with the time of core drilling, to aid the staff in its effort to determine if the no-recovery zones were voids, soil-filled karst features, or unrecoverable weathered limestone.

In its April 2, 2009, response to RAI 2.5.4-5, the applicant stated that, because it is not normal engineering practice, it did not record the drilling pressures or the drill bit revolutions per minute. The applicant also stated that suspension P-S velocity logging in the I-series boreholes was fair to poor or undecipherable because of the sonic drilling technique and poor coupling of the casing with the borehole sidewall, but noted that the results of other geophysical surveys yielded useful data. The applicant provided the additional caliper, acoustic televiewer, and downhole geophysical data used in the engineering analysis to define karst features in the subsurface. For the Avon Park Formation, the applicant used the mass properties in the engineering analyses, and assumed all karst features were voids. This removed the need to define the engineering properties of in-fill materials. Finally, the applicant described plans to obtain the strength and consolidation properties of in-filled and/or weathered-in-place materials as part of the offset boring program.

During the drilling of the offset borings, the applicant recorded drill pressures, rotational drill speed, time of drilling, as well as other data, and attempted to obtain samples for laboratory testing. The applicant compared the results of the offset boring program to those used in the geotechnical analyses performed at the LNP site and concluded that the engineering properties were conservative.

On January 19, 2010, the applicant supplemented its initial response to RAI 2.5.4-5 to include a description of and the results obtained from the offset boring program. The staff's review of the offset boring program is discussed above. Because the applicant recorded the time of drilling, drill bit rotational speed and drill pressures, the staff confirmed that the no recovery zones were not voids, nor contained soft infilled soils, but were characterized as variably weathered Avon Park limestone. Thus, RAI 2.5.4-5 is resolved.

2.5.4.4.3.5 Karst Feature and Void Dimensions Based on Grout Takes

The applicant had estimated the size of actual voids from grout takes measured while backfilling selected core borings made during its exploratory program. The staff reviewed the methodology the applicant employed in determining void size, which consisted of comparing the total grout take to the theoretical volume needed to backfill the boring. The applicant conservatively increased the excess grout volume by 50 or 100 percent depending on the orientation of the void under consideration, and used that volume to estimate the void dimensions. The staff determined that additional information was needed to ensure that this methodology was conservative as the staff postulated that some void volumes could be underestimated if voids contained soil infill, which would effectively reduce the amount of grout take. RAI 2.5.4-6 asks the applicant to confirm that void volumes measured by grout takes were representative of the dimensions of karst features.

In its April 2, 2009, response to RAI 2.5.4-6, the applicant referred to the LNP COL application supplemental information dated September 12, 2008, and the responses to RAIs 2.5.1-5 through 2.5.1-7, 2.5.4-1 and 2.5.4-3a, which describe how the grout take was used to estimate the lateral extent of karst features. The applicant also referred to the response to RAI 2.5.4-8 for the results of additional analyses that modeled “bedding planes” of infilled or weathered-in-place materials instead of voids.

On January 19, 2010, the applicant supplemented its response to RAI 2.5.4-6 to include the results of the offset boring program. The applicant identified the low recovery zones as severely weathered or degraded dolomite that was weathered-in-place and not infilled material as previously identified. Based on the results obtained, the applicant concluded that the grout data analyses to determine the extent of the possible karst features were adequately conservative. The results of the offset boring program indicate that what was once considered soil-filled karst features are actually weathered limestone zones; therefore, the applicant concluded that the size of postulated voids of 3 m (10 ft) in diameter is conservative. Finally, the applicant concluded that the use of soil properties for the material assumed to exist continuously along bedding planes is conservative. The applicant has revised the FSAR to incorporate this additional information.

The staff reviewed the LNP COL FSAR, the related supplemental materials, and the responses to the cited RAIs and concludes that the materials left undocumented in the “no recovery” zones in borings performed during previous explorations were not soil-filled voids as was initially postulated. The staff also reviewed the offset boring program and confirmed that the “no recovery” zones typically contained highly fractured, severely weathered in place materials from the Avon Park limestone parent rock. The staff also concludes that the applicant’s estimate of the size of the voids to be no larger than 3 m (10 ft) in diameter, as determined by grout takes, was sufficiently conservative and supported by the results of the offset boring program.

Finally, the staff reviewed the V_s results and concludes that the V_s measured in the weathered zones (no recovery zones) are similar to other zones of the Avon Park limestone where recovery was made. The staff concludes that had large voids been present in the rock profile, it would have been reflected by the V_s due to the 0.5 meter sampling interval. SER

Figure 2.5.4-12 shows the V_s measured in borings along the N-S profile at LNP Unit 1. The blue dots represent individual V_s measurements and the blue line is the average of those measurements. Adjacent to the V_s profiles are plotted the sample recovery percentages and the RQDs determined during drilling that correspond to the V_s profiles. The lighter red line in this figure represents the sample recovery and the solid green line the RQD. The staff observed no consistent correlation between low recovery and V_s. From this figure, it is apparent to the staff that even at very low recovery rates, the V_s is typically greater than 457 m/s (1,500 fps). From the V_s data the staff concludes that the low recovery zones do not represent voids or soil-filled voids, which was later confirmed by the offset boring program. The staff, therefore, concludes that the maximum size of a void of 3 m (10 ft) diameter is conservative. Accordingly, RAI 2.5.4-6 is resolved.

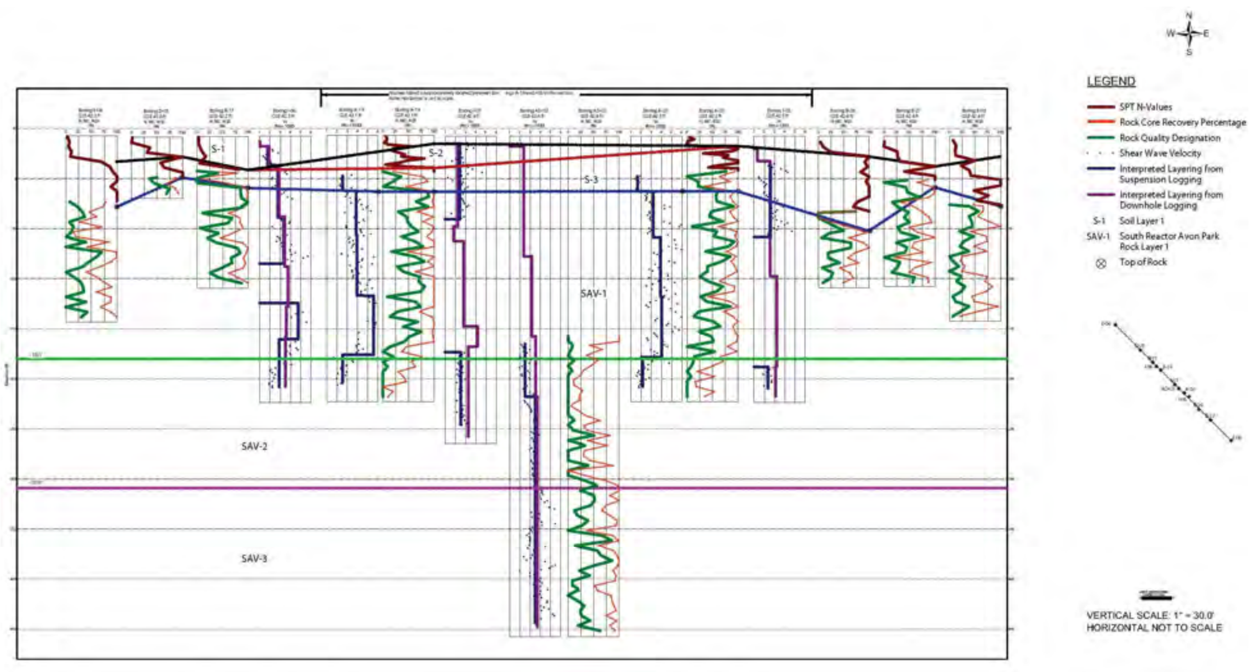


Figure 2.5.4-12. V_s Profiles in Boring AD-03 and Boring AD-20

- (Excerpted from FSAR Figure 2.5.4.2-204B)

2.5.4.4.3.6 Low Recovery of Core Samples

To the staff, it appeared that the recovery rates of rock core samples varied across the site and with depth. Considering that the low recovery rates could be indicative of softer rock, the staff asked the applicant in RAI 2.5.4-9 to describe what considerations it gave to the spatial variation of the low recovery rates.

In its April 2, 2009, response to RAI 2.5.4-9, the applicant stated that the higher recurrence of lower recovery in the center borings results from a vertical variation and noted that adjacent boreholes show similar lower recoveries at the same elevations. The applicant provided a figure illustrating the vertical distribution of the recovery in the boreholes beneath the nuclear island and stated it accounted for this vertical variability in the sensitivity analyses. The staff further noted that the sensitivity analyses conservatively consider the presence of continuous infilled and/or weathered-in-place material along the bedding planes, which accounts for much of the low or no recovery zone material.

The staff reviewed the applicant's response and the boring data. Based on the results of the borings, especially the additional O-series borings drilled in close proximity to the A-series borings, it was apparent to the staff that the high rate of no recovery or poor recovery was a product of the drilling equipment and practices the applicant used during the initial site exploration program. Based on the offset boring results, the staff concludes the recovery rates from the initial boring programs are not useful for identifying soft zones or variations in rock stiffness. The staff therefore relied on the results of the V_s profiles to gauge uniformity of mass rock stiffness, which proved to have uniform average velocity profiles, as observed in SER Figure 2.5.4-12. The staff further concludes that the sensitivity analyses the applicant performed adequately consider any spatial variation. Those analyses are considered in Section 2.5.4.4.10. Accordingly, RAI 2.5.4-9 is resolved.

2.5.4.4.3.7 Grouting of Karst Features

In LNP COL FSAR Section 2.5.4.7, the applicant stated that the purpose of the grouting program was to create a semi-impermeable barrier to reduce ground water inflow into the excavation thereby reducing dewatering requirements. In RAI 2.5.4-17, the staff asked the applicant to clarify this statement that all karst features will be eliminated by the grouting program, discuss any plans for additional exploration that will be implemented to identify karst features to target during the grouting phase, and describe how it will assess whether all the karst features have been eliminated.

In its June 9, 2009, response to RAI 2.5.4-17, the applicant stated that it did not plan any additional site explorations to identify karst features. The applicant also clarified that the statement in question refers to the elimination of known karst features, revised the FSAR to remove the statement in question and referred to FSAR Section 2.5.4.12 for additional details of the subsurface improvements at the site.

The staff reviewed the RAI response, including the FSAR revisions, the completed grout test program, the proposed grouting plan and the referenced FSAR Section 2.5.4.12, and concludes that the applicant proposed satisfactory engineering solutions to grout the eroded vertical joint sets and bedding planes. The staff also concludes that the proposed use of grout holes, including inclined grout holes if deemed necessary, spaced on 4.8 m (16 ft) centers as primary grout points, followed by split-spaced grout holes on 2.4 m (8 ft) centers to an El. of -30.1 m (-99 ft), is an acceptable approach to cutoff seepage. The staff notes that the combination of inclined and split spaced grout holes has a large probability of filling the stipulated vertical joint sets and bedding planes in the Avon Park Formation. The staff also finds that the applicant's commitment to perform a tertiary stage of grouting on 1.2 m (4 ft) centers during excavation activities if the first and second stage grouting does not achieve the desired seepage cutoff is acceptable. The staff also notes that the foundation system is designed to accommodate isolated voids up to 3 m (10 ft) in size, which is at least double the conservatively estimated lateral dimension of any actual void intercepted. Finally, the staff acknowledges that the grout program is not intended to strengthen the foundation, but only reduce inflow into the excavation. Filling of all the voids is therefore not required for stability.

The staff concludes that the proposed grouting plans will minimize seepage into the excavation, reduce pumping requirements, and stabilize the excavation bottom against uplift. The staff further concludes that the combination of the diaphragm wall, grouting program and RCC bridging mat will improve the foundation conditions without the need to fill every joint or open bedding plane. Thus, RAI 2.5.4-17 is resolved.

2.5.4.4.3.8 Conclusion for Foundation Interfaces

Based on the information and findings provided in LNP COL FSAR Section 2.5.4.3, as well as the results of the offset boring program, the staff concludes that the applicant implemented significant and adequate subsurface investigations in relation to the AP1000 safety-related structures at the LNP site to resolve COL Information Item 2.5-5 and COL Information Item 2.5-6 related to foundation interfaces. The staff further concludes that the applicant adequately investigated the subsurface materials beneath the nuclear island construction zone for LNP Units 1 and 2 and beneath the surrounding and adjacent structures. The staff based its conclusions on: (1) its review of plot plans showing the locations of all site explorations, such as borings, seismic and non-seismic geophysical explorations, piezometers, geologic profiles, and the locations of the safety-related facilities; (2) its review of the profiles the applicant presented, illustrating the detailed relationship of the foundations of all seismic Category I and other safety-related facilities to the subsurface materials; and (3) its review of core borings, SPT borings, V_s profiles and non-seismic geophysical logging results. Accordingly, the staff concludes that the foundation interfaces as described in FSAR Section 2.5.4.3 form an adequate basis for the characterization of the foundation interfaces at the LNP site and meets the requirements of 10 CFR Part 50, Appendix A, GDC 2 and Appendix S; and 10 CFR 100.23.

2.5.4.4.4 Geophysical Surveys

The staff focused its review of FSAR Section 2.5.4.4 on the adequacy of the applicant's geophysical investigations to determine soil and rock dynamic properties. The applicant performed both seismic and non-seismic geophysical surveys to characterize the subsurface geology beneath the LNP site. The applicant relied primarily on the suspension P-S velocity logging method to determine the site stratigraphy and provide the engineering properties of subsurface materials, particularly from V_s and V_p profiles. As a secondary method, the applicant performed downhole V_s surveys to confirm the results obtained from the suspension P-S velocity logging. In addition, the staff considered the acoustic televiewer surveys for information regarding verticality of the borehole including graphic images to examine joints and fractures and calculate dip and orientation of planar fractures. Non-seismic surveys of the boreholes included natural gamma, gamma-gamma (density), neutron-neutron (porosity) and induction (conductivity) surveys. The staff also referred to the results of the non-seismic tests for information on the lithology and stratigraphy, location of low density zones, presence of clay, and variations in moisture content.

Based on the results of the suspension P-S velocity logging surveys, the applicant developed the engineering properties for the various layers of the Avon Park limestone. The staff considered the possibility that the suspension P-S velocity logging surveys averaged the velocities of softer zones or voids with denser zones that might occur over the measurement interval. In RAI 2.5.4-4, the staff asked the applicant to discuss the possibility that near-horizontally oriented lenses of soft material were missed or averaged with the high velocities of the adjacent rock. The staff also asked the applicant to describe how it accounted for the variability of the suspension P-S velocity logger results.

In its April 2, 2009, response to RAI 2.5.4-4, the applicant stated that although there was a 1-m (3.2-ft) separation between the receivers used in the suspension logging probe, it measured at 0.5 m (1.6 ft) increments to ensure that the receivers would not completely miss any near-horizontally oriented lenses of soft material. The applicant noted that this interval would also reduce the effect of averaging that is apparent in larger increments. The applicant stated that although the analysis ignored the structural capability of infilled or weathered-in-place materials, these materials were considered in the development of the mass strength and stiffness properties. The applicant referred to the sensitivity analyses provided in response to RAIs 2.5.1-7 and 2.5.4-2, which show that these features are acceptable as voids without any structural capacity. Finally, the applicant performed a sensitivity analysis to address the potential variability of the subsurface materials, and stated that the properties assigned to the postulated continuous bedding features were shown to be less than the properties of the materials revealed by the offset boring program investigation described in SER Section 2.5.4.4.3.

Based on the response to RAI 2.5.4-4, the staff concludes that the smaller measurement interval of 0.5 m (1.6 ft) reduces the possibility that the near-horizontally oriented layers of highly weathered Avon Park limestone or soil in-filled zones would be completely missed and minimizes the effect of averaging softer layers with harder layers. The staff also observed that the variability in the measured velocities throughout the depth of the rock profile is a good indication that the suspension P-S velocity logger detected layers of softer materials interbedded with harder limestone. The staff further notes that because the offset boring program found that the interbedded materials are typically severely degraded weathered-in-place Avon Park limestone, as opposed to soil in-fill, the suspension P-S velocity logging results are representative of the V_p and V_s of individual layers within the Avon Park limestone. Accordingly, RAI 2.5.4-4 is resolved.

The staff also considered the results of non-seismic natural gamma, gamma-gamma, neutron-neutron and induction surveys and concluded that the results suggested continuous low density zones of large areal extent do not exist below the founding level of the RCC bridging mat. The staff further notes that in comparing low density zones to available V_S profiles at similar elevations, the V_s profiles do not fall below 457 m/s (1,500 fps), which is above the 305 m/s (1,000 fps) required by the AP1000 DCD. The staff also observed that the localized low density zones typically fall above the base of the RCC bridging mat, or within the zone to be grouted, and therefore will either be removed and replaced or improved where they do occur below the base of the RCC bridging mat.

The NRC staff reviewed the results of the geophysical surveys, specifically the profiles of V_s and V_p, RQD, percent recoveries, and SPT N-values presented on the geophysical cross-sections in LNP COL FSAR Section 2.5.4.4, the results of non-seismic geophysical surveys presented in response to RAI 2.5.4-5, the applicant's response to RAI 2.5.4-4, and boring logs presented in Appendix BB of the LNP COL FSAR to ensure that the applicant obtained sufficient data to ascertain the soundness and integrity of the rock mass and derived the static and dynamic engineering properties for use in engineering analyses. Based on the applicant's site investigation program and results, the staff concludes that the applicant performed a complete and thorough geophysical survey of the LNP site using a variety of geophysical testing methods. Accordingly, the staff concludes that the applicant adequately addressed COL Information Item 2.5-6. The staff also concludes that the V_s described in FSAR Section 2.5.4.4 addresses Interface Item 2.12. The staff further concludes that the geophysical tests and methods described in FSAR Section 2.5.4.4 form an adequate basis for the geophysical surveys of the LNP site and meets the requirements of 10 CFR 100.23.

2.5.4.4.5 Excavation and Backfill

The NRC staff focused its review of FSAR Section 2.5.4.5 on the horizontal and vertical extent of all seismic Category I excavations, fills, and slopes, ground water conditions and geologic features, the backfill sources, types and quantities of backfill, static and dynamic engineering properties of backfill, compaction specifications, and soil retention system. The staff also considered the applicant's description of the sequence of excavation and backfill plans, particularly the placement of grout between an El. of -7.3 and -30.1 m (-24 and -99 ft), the installation of the diaphragm walls, and the method of excavation and subgrade preparation. The staff noted the applicant's intent to remove and replace the subsurface materials down to an El. of -7.3 m (-24 ft) from which it will construct the 10.7 m (35 ft) thick RCC bridging mat. The applicant stated that backfill between the diaphragm wall and the nuclear island will consist of a low strength concrete-type backfill placed up to the top of the diaphragm wall at approximately an El. of 12.8 m (42 ft). Backfill added above the existing site topography to final site grade at an El. of 15.5 m (51 ft) will be an engineered backfill. The RCC bridging mat is a structural element and is reviewed and discussed in SER Sections 3.7 and 3.8.

2.5.4.4.5.1 Backfill Adjacent to the Nuclear Island

The staff reviewed the use of low strength concrete-type backfill, specifically the CLSMs as backfill material adjacent to the sidewalls of the nuclear island. The staff is familiar with the use of the CLSM for backfilling utility trenches. The use of CLSM has advantages over soil backfill. For example, it typically has strength greater than 3,450 kPa (500 psi) and is easier to place in confined spaces than conventional soil backfill. However, the staff needed additional clarification regarding the potential for long-term strength loss in CLSM due to the leaching out of cementatious bonding materials.

In RAI 2.5.4-22, the staff asked the applicant to justify use of CLSM and address the issue of long-term stability, and provide the design standards, as well as to discuss the construction quality control plans to ensure uniform placement of the CLSM.

In its June 23, 2009, response to RAI 2.5.4-22, the applicant referred to the response to RAI 2.5.4-19 for a discussion of the sliding stability of the nuclear islands under seismic loading conditions. The results of these analyses indicate that the low-strength concrete-type backfill requires no shear capacity and is not subject to long-term stability concerns. The staff reviewed the advantages of using the CLSM the applicant outlined, including ACI 229R-99, "Controlled Low Strength Materials," from which the applicant cited the typical engineering properties and quality control program. The staff notes that the applicant applied the same design standards used for volumetric backfill to the CLSM but this was not reflected in the FSAR. Accordingly, the applicant updated the FSAR to refer to ACI 229R-99.

The applicant revised its response to RAI 2.5.4-22, in a letter dated September 3, 2009, to refer to the revised response to RAI 2.5.4-19, which states that there is no requirement for passive resistance provided by the backfill material adjacent to the nuclear island to remain stable against sliding or overturning. The applicant concluded that there are no concerns about long-term stability of the CLSM because there is no shear capacity requirement for the CLSM.

The staff noted that there is no requirement for passive resistance to achieve sliding stability. This issue was resolved as part of AP1000 RAI TR85-SEB1-10R4 addressing sliding stability. NUREG-1793 and its supplements document the NRC staff's review of the sliding stability analyses performed by Westinghouse for a variety of soil and rock conditions. In NUREG-1793, the staff noted that no backfill passive soil resistance was considered in the analyses and that the AP1000 DCD applicant modeled a lower frictional resistance of 0.55 consistent with the waterproof barrier. The AP1000 DCD applicant performed the analyses using the SSE free-field peak ground acceleration of 0.30g with modified RG 1.60 response spectra, and determined that the displacements were negligible. As documented in Section 3.8.5 of NUREG-1793 and its supplements, the staff accepted the Westinghouse analyses and concluded that passive resistance is not required for sliding stability.

The staff finds that the seismic demand at LNP is significantly less than that used in the analysis for the AP1000 DCD indicating that the dynamic response would be proportionately smaller than that determined in the generic AP1000 design. Since there is no passive resistance requirement at the higher ground motion, the staff concludes CLSM does not have a strength requirement. The staff, therefore, concludes that the CLSM as backfill along the sidewalls of the nuclear island is acceptable. This resolves RAI 2.5.4-22.

2.5.4.4.5.2 Engineered Backfill

The staff noted that the LNP COL FSAR provides limited information regarding the engineered backfill to bring the site to plant grade at an El. of 15.5 m (51 ft). In RAI 2.5.4-26, which was issued in response to RAI 2.5.4-24, the staff asked the applicant for details regarding the source, quantity, compaction specifications and soil properties of the engineered backfill. The applicant was also asked to justify the assumed V_s of 304.8 m/s (1,000 fps) for the backfill used to determine the peak ground acceleration,

The applicant responded to RAI 2.5.4-26 specifying the properties of the engineered fill being placed to bring the site to plant grade. The applicant stated that it did not formally establish the source of the backfill. The applicant places the total volume of engineered fill in the range of 764 to 1,529 cubic meters (1,000 to 2,000 cubic yards) placed within the limits of the diaphragm wall. The applicant stated that the backfill will be a sand fill with variable amounts of silt and clay classified by the Unified Soil Classification System as well-graded sand (SW), silty sand (SM) or clayey sand (SC), compacted to 95 percent of the relative compaction in accordance with ASTM D-1557 (2009) at plus or minus 2 percent of the optimum moisture content. The applicant assumes that the wet unit weight will be on the order of 1,762 kilograms per cubic meter (kg/m^3) (110 pcf), with a V_s in the range of 152 to 305 m/s (500 to 1,000 fps).

In determining what value to use for the V_S, the applicant performed a dynamic sensitivity analysis, which varied the V_s of the engineered fill by values of 152, 259 and 305 m/s (500, 850 and 1,000 fps). The results of this analysis are provided in SER Figure 2.5.4-13, which compares the computed effective cyclic shear stresses between an El. of 10.9 and -41.1 m (36 and -135 ft) for variable V_s.

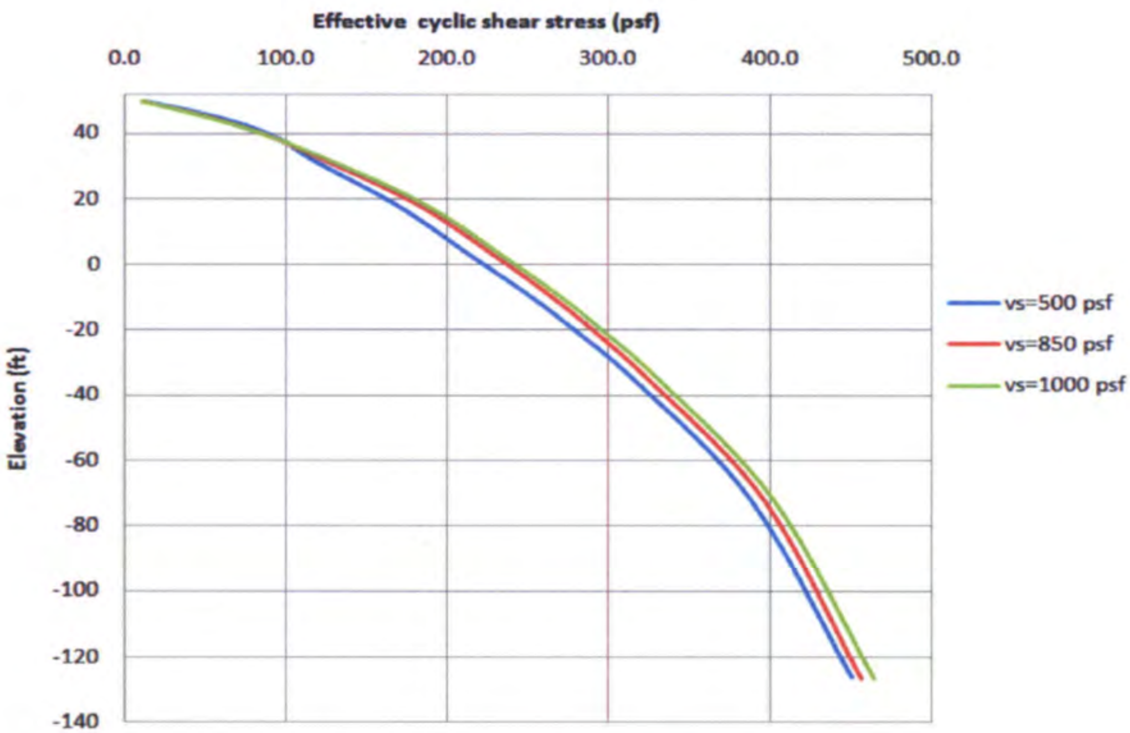


FIGURE RAI 02.05.04-26-2
LNP1 EFFECTIVE CYCLIC SHEAR STRESSES VS ELEVATION

Figure 2.5.4-13. Effective Cyclic Shear Stress as a Function Of Variable V_s Assumptions in the Engineered Fill (RAI Figure 2.5.4-26-2)

SER Figure 2.5.4-13 shows that the shear stresses generated in the underlying materials are only affected minimally by varying the V_s in the engineered fill, and that an assumption of 305 m/s (1,000 fps) results in the most conservative response, i.e., the highest generated shear stresses. The applicant concluded that this was conservative and selected 305 m/s (1,000 fps) to use in its liquefaction reanalysis.

The staff reviewed the response to RAI 2.5.4-26 and concludes that the applicant followed the guidance of RG 1.206 in providing the attributes of the engineered fill proposed for backfilling and bringing the site to final grade. The staff also concludes that the applicant adequately addressed the staff's concern regarding the use of V_s of 305 m/s (1,000 fps) for the engineered fill. The assumed values of 152, 259 and 305 m/s (500, 850 and 1,000 fps), span the range of V_s that could be expected placing a granular fill with variable fines to 95 percent relative compaction. The fact that the assumed V_s of 305 m/s (1,000 fps) results in the highest effective cyclic shear stresses addresses the staff concern that the assumption of 305 m/s (1,000 fps) was conservative. Finally, the staff concludes that the applicant satisfactorily addressed the staff's request for additional information regarding the engineered backfill; therefore, RAI 2.5.4-26 is resolved.

2.5.4.4.5.3 Conclusion for Excavation and Backfill

Based upon its review of LNP COL FSAR Section 2.5.4.5, the staff concludes that the applicant developed and described a complete excavation plan for the LNP site, including the extent of the excavations and the sequence of construction. The staff notes that the depth of the excavation extends to an El. of -7.3 m (-24 ft) with backfill to an El. of 3.3 m (11 ft) made by the placement of a 10.7 m (35 ft) thick RCC bridging mat. The staff concludes that the removal of the existing weathered Avon Park limestone and replacement with a uniform RCC is a significant improvement in the foundation conditions. Regarding the use of CLSM as backfill between the nuclear island sidewalls and the diaphragm wall, the staff concludes that this material will provide uniform backfill and fewer difficulties during placement than with attempting to place engineered fill in the space between the diaphragm wall and the nuclear island. Likewise, the applicant has not yet identified the source of the engineered fill proposed to bring the site to final grade, but the assumed properties of the engineered fill are conservative, and its potential for liquefaction is negligible. Since the engineered fill is not required for overturning or sliding stability, the staff concludes that the information provided in response to RAI 2.5.4-26 regarding the material properties of the engineered fill are sufficient to address COL Information Item 2.5-7. Accordingly, the staff concludes that the applicant adequately addressed COL Information Item 2.5-7. The staff further concludes that the excavation and backfill plans described in FSAR Section 2.5.4.5 form an adequate basis for the excavation for the nuclear islands, and the backfilling operations to bring the LNP site to grade, and meets the requirements of 10 CFR Part 50, Appendix A, GDC 2, and Appendix S.

2.5.4.4.6 Ground Water Conditions

The staff reviewed FSAR Section 2.5.4.6 where the applicant presented the ground water table conditions and construction dewatering plan. The staff reviewed the assumptions the applicant made in the design of the dewatering system and the uplift calculations. The applicant assumed a ground water table elevation of El. of 13.1 m (43 ft), which is coincident with the existing ground surface. The applicant plans to use the diaphragm wall and grouted zone between an El. of -7.3 to -10.6 m (-24 to -99 ft) to form a relatively impermeable barrier to lateral and upward seepage into the excavation. The staff noted that with these barriers in place, the applicant conservatively calculated an inflow rate of approximately 1,892 liters per minute (lpm) 500 gallons per minute (gpm). Considering this inflow rate, the applicant planned to dewater the excavation with six shallow wells using submersible sump pumps placed inside of the diaphragm wall, each with a capacity of 378 lpm (100 gpm). The applicant also planned to place sump and sump pumps at low points in the excavation to handle surface runoff. The applicant also planned for additional grouting to reduce the inflow rate if it should exceed the dewatering system capacity.

The staff noted that the applicant conducted an uplift analysis to ensure the safety of the bottom of the foundation considering the proposed dewatering scheme. In RAI 2.5.4-20, the staff asked the applicant to provide a sample calculation of the uplift analysis including figures showing the assumptions made.

In its June 9, 2009, response to RAI 2.5.4-20, the applicant presented the analyses for local piping conditions and general failure caused by uplift at the base of excavation. Piping in this context is the concentrated flow of water into the excavation caused by excess head. Regarding uplift, if the buoyant forces on the bottom of the excavation exceed the resistance offered by the weight and strength of the foundation, the foundation may heave. The applicant provided the uplift analysis for LNP Unit 2 because it is the more critical case due to the lower shear strength of the foundation limestone. In this case, the applicant assumed uplift on a block having a width equal to half of the diaphragm wall penetration depth. The applicant calculated a FS against uplift of 4.3.

The staff reviewed the applicant’s calculations and concludes that the calculated FS of 4.3 was satisfactory for the temporary condition. The staff notes that the applicant’s assumptions of unit weight and shear strength values used in the analysis were conservative, and that the calculated FS against uplift is sufficiently large to preclude a blowout of the foundation bottom. The staff also concludes that the cementatious nature of the limestone would prevent piping. Finally, the staff concludes that safety of the temporary excavation is further enhanced by the applicant’s plans for additional grouting or additional dewatering wells, if required, to control groundwater inflow and ensure a safe excavation bottom. Based on the computed FS, the conservatism in the assumptions, and the temporary nature of the excavation, the staff concludes that the foundation excavation is safe against heave and/or piping. Accordingly, RAI 2.5.4-20 is resolved.

Based upon its review of FSAR Section 2.5.4.6, the staff concludes that the applicant conservatively assumed the ground water table at the existing ground surface in the design of its dewatering system. The staff concludes that the dewatering plan is adequate to ensure the safety of the excavation. The staff further concludes that the description of the relationship between ground water, excavation, backfill, and the foundations of structures as described in FSAR Section 2.5.4.6 for the LNP site addresses COL Information Item 2.5-8, COL Information

Item 2.5-6 related to ground water conditions, and meets the requirements of 10 CFR Part 50, Appendix A, GDC2, and Appendix S; and 10 CFR 100.23.

2.5.4.4.7 Response of Soil and Rock to Dynamic Loading

In addition to the information addressing the response of soil and rock to dynamic loading presented in FSAR Section 2.5.4.7, the applicant also referred to FSAR Sections 2.5.3 and 2.5.2.5 for discussions of the capable tectonic fault sources and site response analyses and the development of the GRMS, respectively. FSAR Section 2.5.4.4 presents the velocity profiles used in the dynamic site response analysis. Since it was not possible to obtain undisturbed samples from soil layers S-2 and S-3, the applicant assumed dynamic soil properties from the literature cited by the applicant. The staff reviewed the applicant's assumed shear modulus and damping ratio relationships used to perform the site response analysis to obtain the GMRS. The staff reviewed the two sets of EPRI curves, Peninsula Range (PR) and Soft Rock (SR) that the applicant used to represent the range of soft rock behavior in the cemented soil layers S-2 and S-3, and also in the low velocity zone encountered in the Avon Park limestone between an El. of -48 to -67 m (-160 and -220 ft). Soil layer S-1 was not considered as it is either partially or completely removed in the vicinity of the nuclear island. The applicant found that using these two different relationships made little difference in the dynamic site response.

The staff reviewed FSAR Section 2.5.2 where the dynamic relationships for the PR and SR dynamic properties were presented. They are reproduced as shown in SER Figure 2.5.4-14 for convenience. In this figure, it is observed that the two rock types cover a wide range strain related behavior. From this and the wide margin between the site-specific GMRS and the AP1000 DCD CSDRS shown in SER Figure 2.5.4-15, the staff concludes that the choice of dynamic properties for soil layers S-2 and S-3 and the low velocity zone encountered in the Avon Park limestone are relatively unimportant to the determination of the GMRS.

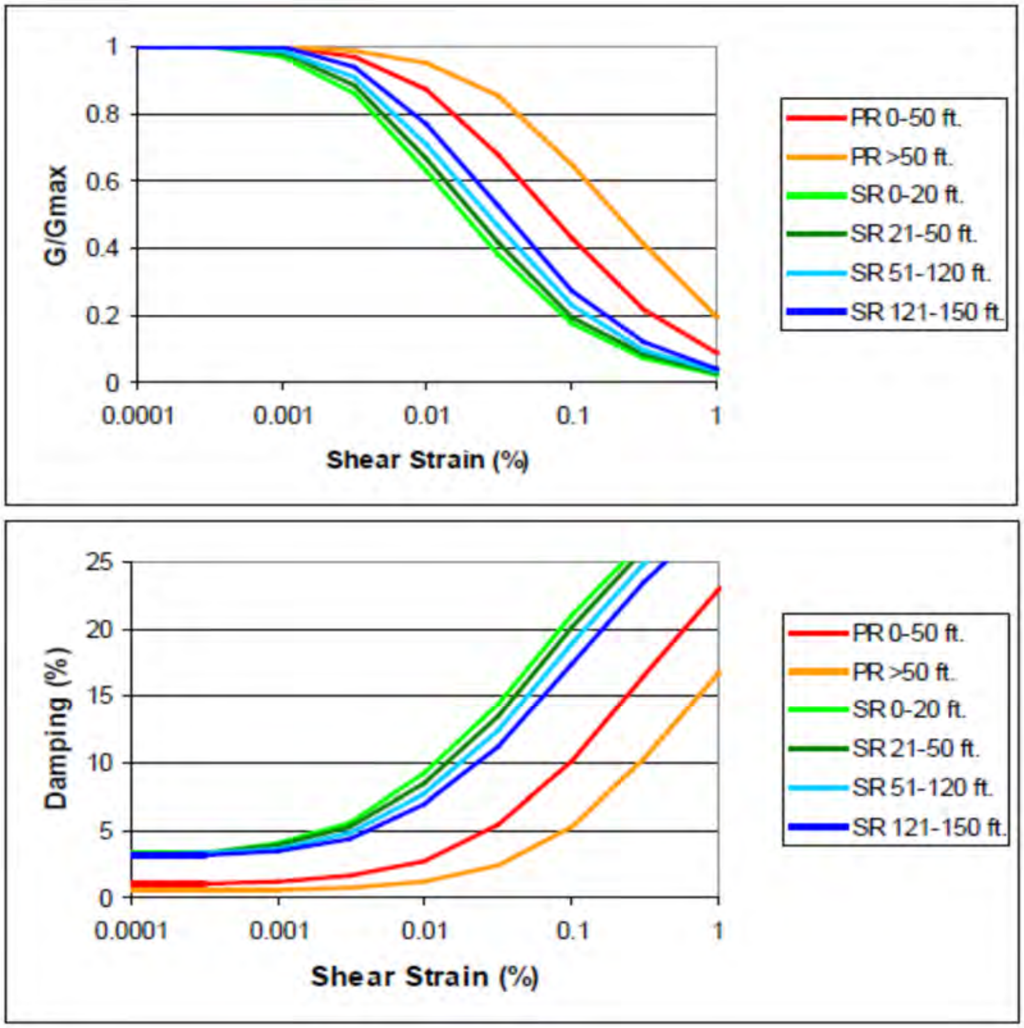


Figure 2.5.4-14. Strain Dependent Shear Modulus and Damping Relationship for Peninsula Rock and Soft Rock (after FSAR Figure 2.5.2-251)

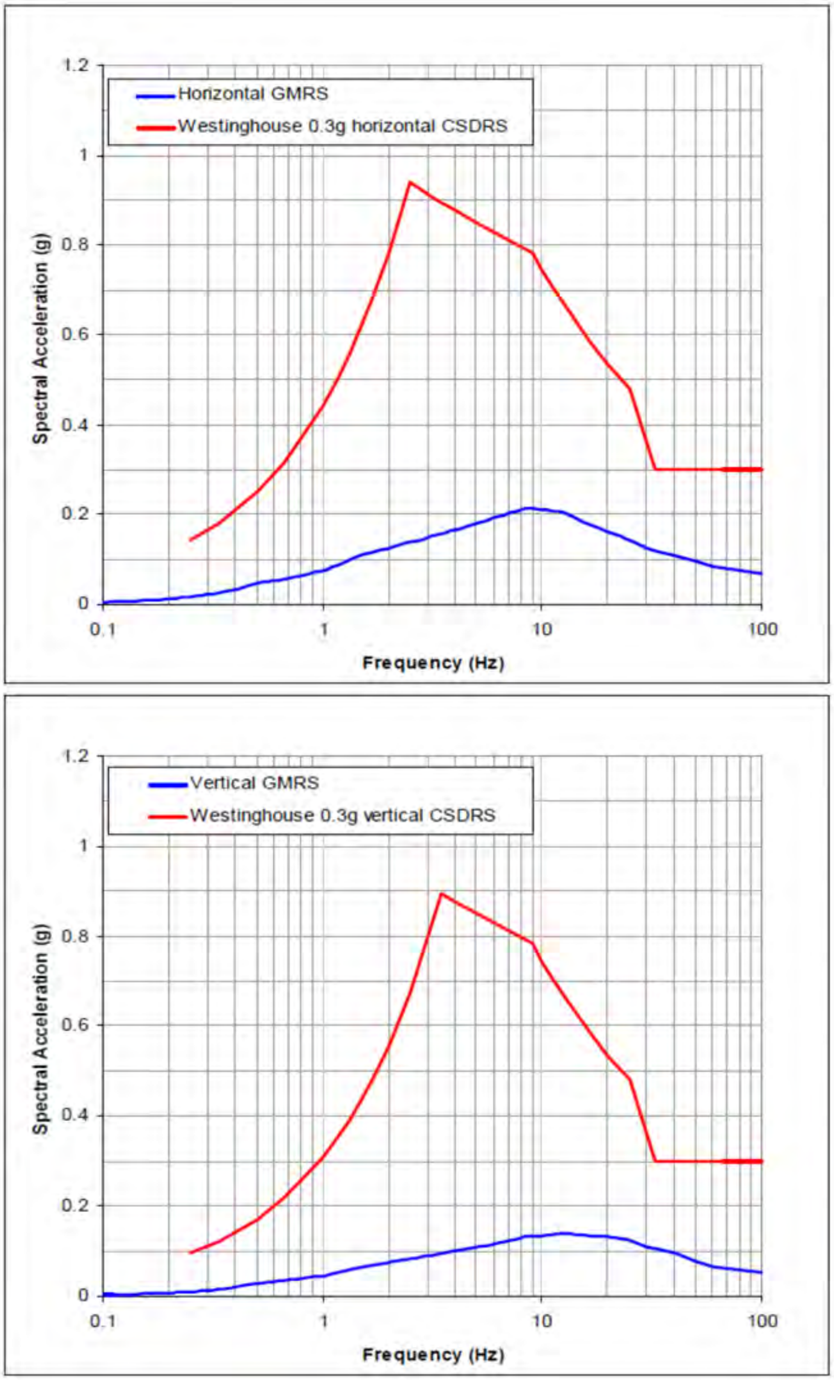


Figure 2.5.4-15. Comparison Between AP1000 Generic Design and Site-specific Response (FSAR Figure 2.5.2-296)

The staff reviewed the requirements for the characterization of the dynamic properties of the soil and rock provided in the AP1000 DCD and concludes that the applicant provided sufficient details to address the requirements of the DCD and satisfy COL Information Item 2.5-6 related to rock dynamic properties. The staff also concludes that the characterization of the dynamic properties of the subsurface materials as described in FSAR Section 2.5.4.7 and related FSAR Sections 2.5.2.5 and 2.5.4.4 forms an adequate basis for the assessment of the response of soil and rock to dynamic loading at the LNP site and meets the requirements of 10 CFR Part 50, Appendix A, GDC2, and Appendix S; and 10 CFR 100.23.

2.5.4.4.8 Liquefaction Potential

In FSAR Section 2.5.4.8, the applicant presented the results of its liquefaction analysis. Because both the Avon Park limestone and the RCC bridging mat are not prone to liquefaction, the applicant stated that liquefaction cannot occur below the nuclear island. However, the applicant found that liquefaction can occur in random zones within the overburden soils, primarily layer S-1, outside the limits of the diaphragm wall. The applicant stated that the random zones of soil with a low factor of safety against liquefaction do not adversely impact nuclear island sliding stability as those zones are isolated and negligible and are generally outside the wedge of soil that resists sliding. More importantly, the applicant concluded analyses by Westinghouse demonstrate that the passive resistance of the backfill is not required for sliding stability. Sliding stability of the nuclear island is evaluated in SER Section 3.8.5.

2.5.4.4.8.1 Liquefaction of Soils Beyond the Diaphragm Wall

The staff reviewed the results of the liquefaction analysis and noted that the applicant identified some foundation materials within the overburden outside the diaphragm wall that are considered to be liquefiable (SF≤1.1) during a SSE event. The staff also noted that these materials appear in isolated areas, some areas designated for removal and replacement with non-liquefiable engineered fill. The staff further notes that these soils are outside the limits of the reinforced concrete diaphragm wall, and would therefore not completely relieve at-rest pressures acting against the nuclear island. Perhaps more importantly, and as discussed earlier with respect to the CLSM, site-specific ground motions are inadequate to cause displacements that would require development of passive pressures, thereby reducing the need for a stable backfill. In order to provide NRC staff reviewing Sections 3.7 and 3.8 with the extent of the liquefiable zones, the staff requested additional information regarding potentially liquefiable soils in RAI 2.5.4-19.

In its June 23, 2009, response to RAI 2.5.4-19, the applicant referred to FSAR Section 2.5.4.8.5, which states that the random zones of soils with a low FS against liquefaction are irrelevant to the sliding stability issue because the zones are isolated and negligible, not required to provide passive resistance that prevents sliding of the nuclear island during the SSE, and/or replaced with non-liquefiable material. The applicant stated that it evaluated the sliding stability of the nuclear islands in a linear static analysis and calculated a FS against sliding of 1.7 irrespective of the passive resistance of the backfill surrounding the nuclear island.

In the September 3, 2009, supplemental response to RAI 2.5.4-19, the applicant addressed overturning as well as sliding and stated that there is no passive pressure required to maintain stability against overturning. The applicant also proposed changes to update FSAR Sections 2.5.4.5.4 and 2.5.4.8.5, LNP COL FSAR Table 2.0-201, and Part 10 of the COL application, Appendix B, Table 3.8-2. In an additional supplement to the response to RAI 2.5.4-19, dated November 5, 2009, the applicant described additional changes made to update FSAR Section 14.3.3.2 that change the minimum coefficient of friction to resist sliding from 0.7 to 0.55.

The staff reviewed the responses to RAI 2.5.4-19 and determined that a review of the linear analysis the applicant performed was needed before the staff could conclude that liquefaction of the backfill was irrelevant to the sliding stability of the nuclear island. Accordingly, in RAI 2.5.4-25 the applicant was requested to provide the linear analysis for the staff's review of LNP COL FSAR Section 3.8.5.

Prior to receiving the response to RAI 2.5.4-25, the applicant responded in a September 3, 2009, supplemental response to RAI 2.5.4-19, that the Westinghouse non-linear sliding analysis discussed in the Westinghouse response to AP1000 RAI TR85-SEB1-10R2 was the sole basis to conclude that isolated pockets of liquefiable zones will not affect the sliding stability of LNP Units 1 and 2. In addition, the applicant voided the site-specific calculation for sliding stability referenced in its original June 23, 2009, response to RAI 2.5.4-19 as it was no longer necessary to support the conclusions.

Subsequent revisions to AP1000 RAI-TR85-SEB1-10R2 resulted in acceptance of AP1000 RAI-TR85-SEB1-10R4, where staff concluded that sliding stability and overturning stability was not dependent on passive resistance of the soil backfill. The staff's evaluation is in NUREG-1793 and its supplements.

In its revised June 8, 2010, response to RAI 3.8.5-3, the applicant prepared plan and profile drawings showing the locations of the liquefied zones to answer questions related to LNP COL FSAR Section 3.8.5 stability concerns. This response was directed to the question of the impact on lateral stability of liquefiable soils surrounding the drilled piers. Evaluation of both the sliding stability and lateral stability of drilled piers are reviewed in SER Sections 3.7 and 3.8.5. The applicant's responses to RAIs 3.8.5-3 and 3.8.5-7 included proposed revisions to LNP COL FSAR Section 2.5.4.5 and Section 2.5.4.8.5 to add information about the liquefied zones. The NRC finds these changes acceptable. Because the applicant has provided the details requested in response to RAI 3.8.5-3, the staff considers RAIs 2.5.4-19 and 2.5.4-25 resolved. The incorporation of changes in a future revision to the LNP COL FSAR is being tracked as **Confirmatory Item 2.5.4-1**.

Resolution of Confirmatory Item 2.5.4-1

Confirmatory Item 2.5.4-1 is an applicant commitment to update section 2.5.4 of its FSAR. The staff verified that LNP COL FSAR Section 2.5.4 was appropriately updated. As a result, Confirmatory Item 2.5.4-1 is now closed.

2.5.4.4.8.2 Revised Liquefaction Analysis for Proposed Backfill

The staff observed that the liquefaction analysis did not include the engineered backfill to be placed between the existing site grade at an El. of 13.1 m (43 ft) to the final plant grade at an El. of 15.5 m (51 ft). In RAI 2.5.4-24, the staff asked the applicant to update the liquefaction evaluations to include the planned backfill.

In its January 19, 2010, response to RAI 2.5.4-24, the applicant presented revised liquefaction evaluations for the modified soil profile, to include the added overburden, and re-calculated the FS. The applicant again followed the guidance of RG 1.198 with respect to calculation of the liquefaction potential and identified the zones for which it calculated the low or intermediate FSs. The calculations included data from the boreholes completed as part of the offset boring program discussed in SER Section 2.5.4.4.3. The applicant's response included replacement tables for FSAR Tables 2.5.4.8-202A and 2.5.4.8-202B that include borehole data from the offset boring program (Tables RAI 2.5.4-24-1 and RAI 2.5.4-24-2, respectively). The applicant concluded that the results of the liquefaction analysis are consistent with the earlier conclusions that liquefaction is confined to isolated pockets.

The staff reviewed the liquefaction analysis results and concludes that in addition to the previously identified zones of liquefaction some additional zones will liquefy that were not identified in the applicant's initial analysis. However, as noted earlier, liquefaction will not impact the foundation of the nuclear island as it is founded on a 10.7 m (35 ft) thick RCC bridging mat resting on the Avon Park limestone, neither of which is liquefiable. Additionally, the staff notes that neither the CLSM backfill immediately surrounding the nuclear island, nor the densely compacted engineered fill that brings the site to plant grade, have the potential for liquefaction. The staff considers RAI 2.5.4-24 resolved. The incorporation of Tables RAI 2.5.4-24-1 and RAI 2.5.4-24-2 in a future FSAR revision is being tracked as **Confirmatory Item 2.5.4-2**.

Resolution of Confirmatory Item 2.5.4-2

Confirmatory Item 2.5.4-2 is an applicant commitment to update Section 2.5.4 of its FSAR. The staff verified that LNP COL FSAR Section 2.5.4 was appropriately updated. As a result, Confirmatory Item 2.5.4-2 is now closed.

The overburden soil layer S-1 that is subject to liquefaction is outside the limits of the 1.06 m (3.5 ft) thick reinforced concrete diaphragm wall, and is partially removed and replaced during construction of the non-safety related structures. The staff notes that limited zones of liquefaction of natural soils occurs in isolated areas surrounding the nuclear island and surrounding some of the drilled pier locations that support the Turbine, Annex and Radwaste

Buildings. To address the liquefaction concerns, the applicant has designed a drainage system consisting of 6 inch diameter vertical drains capped by a 2 ft thick horizontal drainage blanket. The purpose of the drainage system is to relieve the buildup of pore water pressure in the potentially liquefiable zones during earthquake shaking. The pore water pressure relief prevents liquefaction from occurring.

The staff reviewed the design of the drainage system and concludes that the addition of the drainage system, fully penetrating 6 in diameter relief wells, discharging into a 2 ft thick horizontal drainage blanket, has effectively eliminated the liquefaction concerns. The staff further concludes that eliminating the potential for liquefaction preserves the lateral support at the below ground nuclear island walls and at the drilled pier locations.

2.5.4.4.8.3 Liquefaction Potential of CEUS SSC and Seismic Margins Analysis

To evaluate the seismic hazard at LNP site against the new hazard calculation requested by NRC RAI Letter 108, the applicant provided a liquefaction potential assessment using the CEUS SSC model (NUREG-2115) in its FSAR Section 2.5.4.8.7. The staff's detailed review of the applicant's CEUS SSC liquefaction potential evaluation is documented in Subsection 20.1.4.5 of this SER. Based on its review, the staff concludes that the liquefaction evaluations based on the updated EPRI-SOG (design basis) ground motions bound those from the CEUS SSC ground motions.

For the purpose of seismic margins analysis, the applicant also assessed liquefaction potential for ground motions in excess of the site responses corresponding to the GMRS and PBSRS in its FSAR Section 2.5.4.8, and performed sensitivity analysis of the median centered liquefaction potential for 10^{-5} UHRS in its FSAR Section 2.5.4.8.6. The staff's detailed review of the applicant's site-specific seismic margins analysis for liquefaction potential is documented in Subsection 20.1.7.5 of this SER. Based on its review, the staff concludes that the applicant's assumed ground motion based on EPRI-SOG 10^{-5} UHRS for seismic margin considerations is conservative, and concludes that the locations and elevations of hypothesized liquefaction based on 10^{-5} UHRS are almost identical with that based on the design basis.

2.5.4.4.8.4 Conclusion for Liquefaction Potential

Based upon its review of LNP COL FSAR Section 2.5.4.8, the staff concludes that no liquefaction can occur below the nuclear island as the RCC bridging mat and Avon Park formation are both non-liquefiable. The staff further concludes that the site-specific analysis provides an adequate basis to resolve COL Information Item 2.5-9. The staff notes that with the addition of the drainage system in the nonsafety related structure areas, where liquefaction was predicted to occur in the unconsolidated sand layers, liquefaction will be effectively eliminated. The staff therefore concludes that because there is no requirement for passive resistance of the backfill, and because liquefaction is eliminated by the presence of the drainage system, widespread liquefaction of the natural soils surrounding the diaphragm wall and drilled piers will not occur, and potential adverse impacts to the stability of the nonsafety-related structures is effectively controlled. NUREG-1793 and its supplements provide the NRC staff's evaluation of the sliding stability of the Westinghouse AP1000 design indicating no passive resistance requirement for backfill. The review and evaluation of the stability of the drilled piers supporting the seismic Category II and nonsafety-related structures are presented in Sections 3.7.2 and 3.8.5 of this SER.

The staff concludes that the liquefaction analysis described in LNP COL FSAR Section 2.5.4.8 forms an adequate basis for the assessment of the potential for liquefaction at the LNP site and meets the requirements of 10 CFR Part 50, Appendix A, GDC 2, and Appendix S; and 10 CFR 100.23.

2.5.4.4.9 Earthquake Site Characteristics

LNP COL FSAR Section 2.5.4.9, "Earthquake Site Characteristics" refers to FSAR Section 2.5.2 for a detailed discussion of the GMRS. A detailed evaluation of FSAR Section 2.5.4.9 is presented in SER Section 2.5.2.4.

2.5.4.4.10 Static Stability

As part of its review of FSAR Section 2.5.4.10, the staff considered the determination of the bearing capacity, settlement and earth pressures at LNP Units 1 and 2. The following sections discuss these determinations of static stability in greater detail.

2.5.4.4.10.1 Bearing Capacity

The staff reviewed the determination of the bearing capacity at the LNP Units 1 and 2 site, including the information provided to resolve COL Information Item 2.5-10 verifying that the Avon Park limestone is capable of supporting the maximum bearing reaction determined from the analyses described in DCD Appendix 3G of 426 kPa (8,900 psf) static loading and described in LNP COL FSAR Section 3.7.2.4.1.6 of 1,149 kPa (24,000 psf) on soft rock under all combined loads, including the site-specific SSE.

The applicant performed the bearing capacity analyses using both FEM analysis methods and closed form solutions based on plasticity theory to determine the bearing capacity of the Avon Park limestone.

2.5.4.4.10.2 FEM and Closed Form Solutions for Bearing Capacity

Due to the complexity of the rock profile at the LNP site, including possible karst features, the applicant used FEM analyses to confirm bearing capacity results obtained using bearing capacity equations based on plasticity theory. The staff asked the applicant in RAI 2.5.4-2a to provide a detailed explanation of how variability within the supporting rock profile was modeled in the FEM analysis. RAI 2.5.4-2b asked the applicant to describe the FEM results, and RAI 2.5.4-2c asked the applicant to describe how it determined the rock mass properties for use in the USACE bearing capacity equations. (USACE EM 1110-1-1905, 1992)

In its November 20, 2008, response to these three parts of RAI 2.5.4-2, the applicant stated that the layered rock modeled for the FEM analysis consisted of three layers at LNP Unit 1 and four layers at LNP Unit 2 based on the geophysical test results, field data gathered during rock coring, and results of laboratory strength tests. The applicant utilized the SAP2000 software for the FEM analysis to generate a model of the foundation.

The applicant determined the rock mass strength parameters, cohesion and friction angle, from the Hoek-Brown criteria and used these parameters as input in the FEM and bearing capacity equations. The applicant modeled the potential voids in the Avon Park limestone by assuming a range of cavity sizes and assigning zero stiffness to the voids. Void sizes ranged from 3 m (10 ft) wide slots across the entire footprint to 3 and 6 m (10 and 20 ft) cubes located at various critical elevations and positions beneath the base of the RCC and below the bottom of the grouted zone.

The staff reviewed the results of the FEM approach and noted its primary advantage is that voids could be included in the model. The staff noted that the multiple FEM analyses the applicant performed for various cases, with design voids located at various positions below the RCC bridging mat, resulted in calculated FS of at least 3.0.

The applicant also provided additional information on the bearing capacity determinations using the USACE equation. The applicant calculated the bearing capacity for the local and general shear failure cases for LNP Units 1 and 2. For the static analysis, the applicant compared the ultimate bearing capacity to the average bearing pressure to calculate the FSs of 7.6 and 5.7 for the general and local shear failure cases at LNP Unit 2, and 7.2 and 5.3 for the general and local shear failure cases at LNP Unit 1. In the dynamic analysis, the applicant compared the ultimate bearing capacity at the bottom of the RCC bridging mat to the dynamic bearing demand. The applicant determined that the FSs against failure during the SSE were greater than 2.5 for the general shear failure condition.

The staff concludes that the two approaches, FEM and bearing capacity equations, yield factors of safety that are in general agreement with one another and are greater than or equal to factor of safety criteria for nuclear power plants, FS of 3 for the static case, and 2 for the dynamic case. In the finite element analysis, which allowed for the inclusion of postulated voids below the nuclear island, the applicant assumed conservatively sized potential void sizes greater than actual voids sizes based on the field data resulting in conservative assumptions used in the engineering analyses. Accordingly, RAIs 2.5.4-2a through 2.5.4-2c are resolved.

2.5.4.4.10.2.1 Bearing Capacity Sensitivity Analysis Using Closed Form Solutions

The staff reviewed the results of the bearing capacity analysis performed using the bearing capacity equations. In RAI 2.5.4-7b, the staff asked the applicant to describe any sensitivity analyses, which considered variations in the rock mass parameters determined from a statistical analysis of the UCS.

The applicant presented results where it calculated the FS against bearing capacity failure within the Avon Park Formation using three methods: the USACE (1992) method, Hoek, E., et al. (2002) method, and Serrano-Otalla (1994) method. Each of these methods considered three sets of strength parameters based on the mean, median and 84th percentile UCSs of the Avon Park limestone. Based on the results shown in SER Table 2.5.4-5, the staff concluded that the FSs against bearing capacity were adequate. SER Table 2.5.4-5 shows the FS results of the sensitivity analyses were approximately 3.0 for the mean, median and lower bound strength parameters for the general bearing capacity case.

The staff reviewed the bearing capacity sensitivity analyses and performed its own confirmatory analyses. The confirmatory analyses included confirmation that the rock mass properties were representative of the in situ conditions. The staff used the RocLab 1.031 computer program to confirm the rock mass Mohr-Coulomb strength parameters, friction and cohesion, indicated in SER Table 2.5.4-5 for LNP Unit 2, Case 1, mean and lower bound UCS strength values. Using the USACE bearing capacity computer program CBEAR, based on EM 1110-1-1905, the staff determined the FS for LNP Unit 2, Case 1, lower bound UCS values, to be 2.9. This is in agreement with the applicant as shown in SER Table 2.5.4-5. Since the staff reproduced the applicant's results in its confirmatory analyses, the staff concluded that the results presented in SER Table 2.5.4-5 were reliable and the bearing capacity of the foundation rock at the LNP Units 1 and 2 was acceptable.

Table 2.5.4–5. Bearing Capacity Sensitivity Results (Table RAI 2.5.4.7-1)

		North (LNP 2)						South (LNP 1)					
		I*			II**			I			II		
		Mean UCS	Median UCS	Lower bound UCS	Mean UCS	Median UCS	Lower bound UCS	Mean UCS	Median UCS	Lower bound UCS	Mean UCS	Median UCS	Lower bound UCS
Rock Mass Properties	Unit Weight, kg/m^3 (pcf)	2,013 (125.7)			1,890 (118.0)			2,116 (132.1)			2,002 (125.0)		
	Cohesion, kPa (ksf)	201 (4.2)	158 (3.3)	90.9 (1.9)	143 (3.0)	114 (2.4)	71.8 (1.5)	167 (3.5)	143 (3.0)	86.1 (1.8)	153 (3.2)	138 (2.9)	86.1 (1.8)
	Friction Angle, degrees	20.0	18.3	14.8	16.3	14.8	11.6	20.3	19.2	15.8	15.5	14.8	11.9
USACE (1996) General Shear Failure	Ultimate Bearing capacity, kPa (ksf)	3,662 (76.5)	2,896 (60.5)	1,790 (37.4)	4,184 (87.4)	3,490 (72.9)	2,451 (51.2)	3,543 (74.0)	3,016 (63.0)	1,915 (40.0)	4,634 (96.8)	4,280 (89.4)	3,078 (64.3)
	FS	6.0	4.8	2.9	6.2	5.2	3.6	5.8	5.0	3.2	6.2	5.7	4.2
USACE (1996) Local Shear Failure	Ultimate Bearing Capacity, kPa (ksf)	2,743 (57.3)	2,078 (43.4)	1,158 (24.2)	-	-	-	2,599 (54.3)	2,145 (44.8)	1,235 (25.8)	-	-	-
	FS	4.5	3.4	1.9	-	-	-	4.3	3.5	2.0	-	-	-
Hoek et al. (2002)	Ultimate Bearing Capacity, kPa (ksf)	3,614 (75.5)	2,834 (59.2)	1,723 (36.0)	3,940 (82.3)	3,246 (67.8)	2,240 (46.8)	3,868 (80.8)	3,164 (66.1)	1,915 (40.0)	4,337 (90.6)	3,983 (83.2)	2,805 (58.6)
	FS	6.0	4.7	2.8	5.9	4.8	3.3	6.4	5.2	3.1	5.8	5.3	3.8
Serrano-Otalla (1994)	Ultimate Bearing Capacity, kPa (ksf)	5,362 (112.0)	4,036 (84.3)	2,259 (47.2)	4,893 (102.2)	3,974 (83.0)	2,523 (52.7)	5,798 (121. 1)	4,591 (95.9)	2,552 (53.3)	5,305 (110. 8)	4,802 (100.3)	3,184 (66.5)
	FS	8.8	6.6	3.7	7.3	5.9	3.8	9.5	7.6	4.2	7.1	6.4	4.3

*I refers to the bearing capacity at the top of the Avon Park Formation NAV-1 for (LNP2) and SAV-1 for (LNP1).
**II refers to the bearing capacity at the top of the lower strength zones NAV-3 (LNP2) and SAV-2 (LNP1).

Based on the FEM analyses and the USACE bearing capacity equation solution results, the staff concludes that the Avon Park limestone has an adequate margin of safety for the static and dynamic loads that will be imposed by the RCC bridging mat and nuclear island under both static and dynamic cases, and the bearing capacity meets or exceeds the bearing capacity criteria set forth in the AP1000 DCD. Accordingly, RAI 2.5.4-7b is resolved.

2.5.4.4.10.3 Settlement

The staff focused its review on the calculations of total and differential settlement for the nuclear island and the surrounding seismic Category II and nonsafety-related structures. The staff reviewed: (1) the effect of voids below the grouted zone on settlement; (2) the effects of continuous soft bedding layers on settlement; and (3) the effect of spatial variability in the limestone layer stiffness across the site on settlement. The staff issued the following RAIs prior to the completion of the offset boring program that the applicant performed to characterize the materials in the no recovery zones and that is discussed in detail in SER Section 2.5.4.4.3.

2.5.4.4.10.3.1 Effect of Voids at Depth on Settlement

During the review of the boring logs, the staff noted that very few borings went deeper than an El. of -45.7 m (-150 ft). In RAI 2.5.4-3, the staff asked the applicant to discuss the basis for the conclusion that larger voids do not exist below an El. of -45.7 m (-150 ft). The staff also requested that the applicant provide a sample settlement calculation.

In its November 20, 2008, response to RAI 2.5.4-3, the applicant characterized the karst features in the site vicinity as solution channels in the Avon Park limestone oriented along near-vertical fractures with cavities developing as the fracture walls dissolve. The applicant cited four borings that extended to an El. of -137 m (-450 ft), and an additional 28 borings that extended between an El. of -45.7 and 83.8 m (-150 and -275 ft), and concluded that these borings support the evaluation of karst features described in the FSAR. Finally, the applicant stated that engineering analyses incorporating a conservatively sized void of 6 by 6 m (20 by 20 ft) located below an El. of -45.7 m (-150 ft) demonstrated the safety of the foundation structure.

The applicant based its soil and rock profiles on the geotechnical site investigation data and provided the layered subsurface profiles used in the settlement analyses for LNP Units 1 and 2. The elastic properties of the mass rock were derived from small strain V_s measurements and reduced by 50 percent to account for larger strains. The applicant provided a sample settlement calculation, which concluded that total settlements were less than 0.50 cm (0.2 in).

The staff reviewed the applicant’s response to RAI 2.5.4-3, the borings in Appendix BB, the result of the offset boring program (O-series), and the referenced responses and supplements to other RAIs.

Based on the review, the staff concludes that the karst features appear to occur along vertical fractures and at junctures with horizontal bedding planes in the “plus sign” morphology the applicant described. Additionally, the staff concludes that voids having dimensions greater than the design void of 3 m (10 ft) are not anticipated below an El. of -45.7 m (-150 ft) based on the distribution of voids encountered during the exploration of the site, the increasingly dolomitized nature of the Avon Park limestone with depth, and the reduced ability of downward directed seepage to dissolve limestone as surface water percolates downward. The staff therefore concludes that it is reasonable to assume that larger voids do not exist below an El. of -45.7 m (-150 ft).

The staff also reviewed the FEM analysis results, which show that a 6 m (20 ft) cube void located below the grouted zone, and subjected to the nuclear island static loading, results in the same magnitude settlement, approximately 0.5 cm (0.2 in), as what occurs when no void is present. The staff also noted from the FEM analysis that at two times the static load, the deformation remains essentially linear and the settlement is only 1.3 cm (0.5 in). The staff concludes that this settlement will occur during construction given the stiffness of the Avon Park limestone. The staff finds this predicted settlement acceptable, because it is within the AP1000 DCD limits. Accordingly, RAI 2.5.4-3 is resolved.

2.5.4.4.10.3.2 Settlement Sensitivity to Variations in Elastic Modulus

The staff reviewed the settlement sensitivity to variations in assumed elastic modulus, postulated embedded soft layers, and zones of higher/lower RQDs.

2.5.4.4.10.3.2.1 *Variation in Assumed Elastic Modulus*

The staff reviewed the V_s profiles and observed that some variability from the mean exists, particularly in SAV-1 and NAV-1. In RAI 2.5.4-7c, the staff asked the applicant to describe any settlement sensitivity analyses performed that accounted for variations in the stiffness of the average properties assumed for the layered Avon Park limestone.

The applicant presented the results of a settlement sensitivity analysis varying the stiffness of the Avon Park formation by reducing the mean elastic modulus by one-third, one-half and one standard deviation. The applicant reported that settlements computed by the sensitivity analysis remained well within the range of allowable settlements. Later, in a supplemental response, the applicant compared the properties obtained from the offset boring program described in SER Section 2.5.4.4.3 with those assumed for the sensitivity analyses and confirmed the conservatism of the elastic moduli used in the sensitivity analyses.

The staff reviewed the results of the sensitivity analyses and performed confirmatory calculations of the settlements using elastic theory. Once the staff confirmed that the V_s results accurately represented the in-situ conditions, the staff performed settlement calculations at LNP Units 1 and 2 using profiles provided in response to RAI 2.5.4-3. The relationship for elastic deformation was based on the following equation (Bowles, 1988):

$$\Delta\delta = \sum_i \frac{H_i \Delta\sigma_i}{E_{mc}}$$

where:

$\Delta\delta$ is the total elastic settlement
H_i is the thickness of layer i
$\Delta\sigma_i$ is the incremental increase in vertical stress due to foundation loading at the i^{th} layer
E_{mc} is the average constrained elastic modulus derived for large strains from the small strain Vs profiles.

This equation is also used in the American Society of Civil Engineers, “Bearing Capacity of Soils,” which is referenced in NUREG-0800. The settlement analysis was performed to a total depth of 132 m (434 ft). The staff assumed very conservative lower bound elastic modulus values for each of the layers at LNP Units 1 and 2. The elastic modulus values were based on the minimum V_S recorded in the respective layers published in FSAR Table 2.5.4.2-214. As can be seen in this table, the minimum small strain V_S are typically one-half of the average V_S. The elastic modulus values computed from the minimum small strain V_S were then corrected for large strains using the correction factor of 0.5. The staff computed a maximum total settlement of 2.8 cm (1.1 in) for LNP Unit 1 and 2 cm (0.8 in) for LNP Unit 2. Though these calculations were based on conservative E_{mc}’s, the total settlement values are still bounded by the maximum total settlement allowed by the AP1000 DCD, 15.2 cm (6 in). Therefore, the staff concludes that the settlements at LNP Units 1 and 2 are well within the range of acceptable settlements required by the AP1000 DCD. Thus, RAI 2.5.4-7c is resolved.

2.5.4.4.10.3.3 Postulated Embedded Soft Layers

The staff had questions about how the applicant incorporated joints and soft bedding layers into the FEM analyses for settlement. In RAI 2.5.4-8, the staff requested that the applicant describe how it modeled joint patterns and soil filled bedding planes in the FEM analysis for the evaluation of settlement.

In its April 2, 2009, response to RAI 2.5.4-8, the applicant stated it implicitly and explicitly modeled the joints and bedding planes in the FEM analyses and noted that it considered highly conservative shapes, sizes, physical properties and locations of postulated voids. The applicant also stated that its review of the geophysical test results yielded a list of potential soft/infill locations, 13 of which were identified across two or more borings at the LNP site. The applicant modeled these as continuous features to evaluate the total and differential settlement associated with their presence at the LNP site. The applicant did not take any credit for the subsurface improvement that results from grouting between an El. of -7.3 and 30.1 m (-24 and -99 ft). Based on the results of the sensitivity analyses, the applicant concluded that the presence of soft bedding planes would be tolerated by the RCC bridging mat and the settlement would still be within the AP1000 DCD requirements. The applicant further concluded that, based on the highly conservative assumptions used in the sensitivity analysis, an adequate safety margin exists at the LNP site.

On January 19, 2010, the applicant supplemented the response to RAI 2.5.4-8 to include the results of the offset boring program. The applicant compared the conservative properties assumed for the elastic modulus, Poisson’s ratio and unit weight during previous sensitivity analyses with the properties estimated from the results of the offset boring program and concluded that the sensitivity studies were adequately conservative.

The staff reviewed the results of the settlement sensitivity analysis assuming the inclusion of 13 soft continuous 0.3 m (1 ft) thick layers underlying the nuclear island. The staff concludes that the inclusion of these 13 continuous layers within the Avon Park limestone is conservative from the standpoint that they are not present in all borings and are therefore discontinuous. Additionally, the staff concludes that the properties assigned to these layers are conservative based on the results of the boring offset program that demonstrated that the materials are not soil infill, but consist instead of variably weathered limestone. The applicant assigned the soft layers an elastic modulus equivalent to that of loose sand, or about 3 percent of the value assigned to rock layer NAV-1 at LNP Unit 2. The results of the analysis demonstrated that total and differential settlements would only nominally increase, total settlement being less than 1.3 cm (0.5 in). The staff compared the AP1000 DCD settlement criteria of 15.2 cm (6 in) total settlement and/or 1.3 cm (0.5 in) differential settlement in 15.2 m (50 ft) to the total settlement of 1.3 cm (0.5 in) calculated given the conservative assumptions of 13 soft layers, and concludes that the settlement criteria is met. The staff conducted a confirmatory settlement analysis using elastic theory and the applicant's material property assumptions and obtained similar results to those of the applicant. Accordingly, RAI 2.5.4-8 is resolved.

2.5.4.4.10.3.4 Sensitivity and Variability in the Avon Park Formation

For completeness, the staff asked the applicant to determine settlements for the condition where stiffness of the Avon Park limestone varies laterally. In RAI 2.5.4-11, the staff asked the applicant to discuss the settlement sensitivity due to the discontinuous soft bedding planes revealed in the borings.

In its June 23, 2009, response to RAI 2.5.4-11, the applicant used a 3D FEM to perform the sensitivity analysis, which evaluated the settlements considering the static loads and the weight of the RCC bridging mat.

To address the settlement sensitivity to lateral variation in layer stiffness as observed in zones of higher and lower RQDs, the applicant submitted the results of a sensitivity analysis in which it varied the elastic properties across the foundation footprint based on RQD values. The applicant devised two zones for the settlement sensitivity analysis, as shown in SER Figure 2.5.4-16. One zone consists of limestone exhibiting medium to high RQDs of greater than 50 percent, and a second zone consisting of medium to low RQDs of less than 50 percent. The applicant noted that the zoning based on these RQD values created localized zones of softer material surrounded by zones of stiffer material, consistent with the conclusion that soft bedding layers are limited in extent and do not extend across the entire footprint of the LNP site.

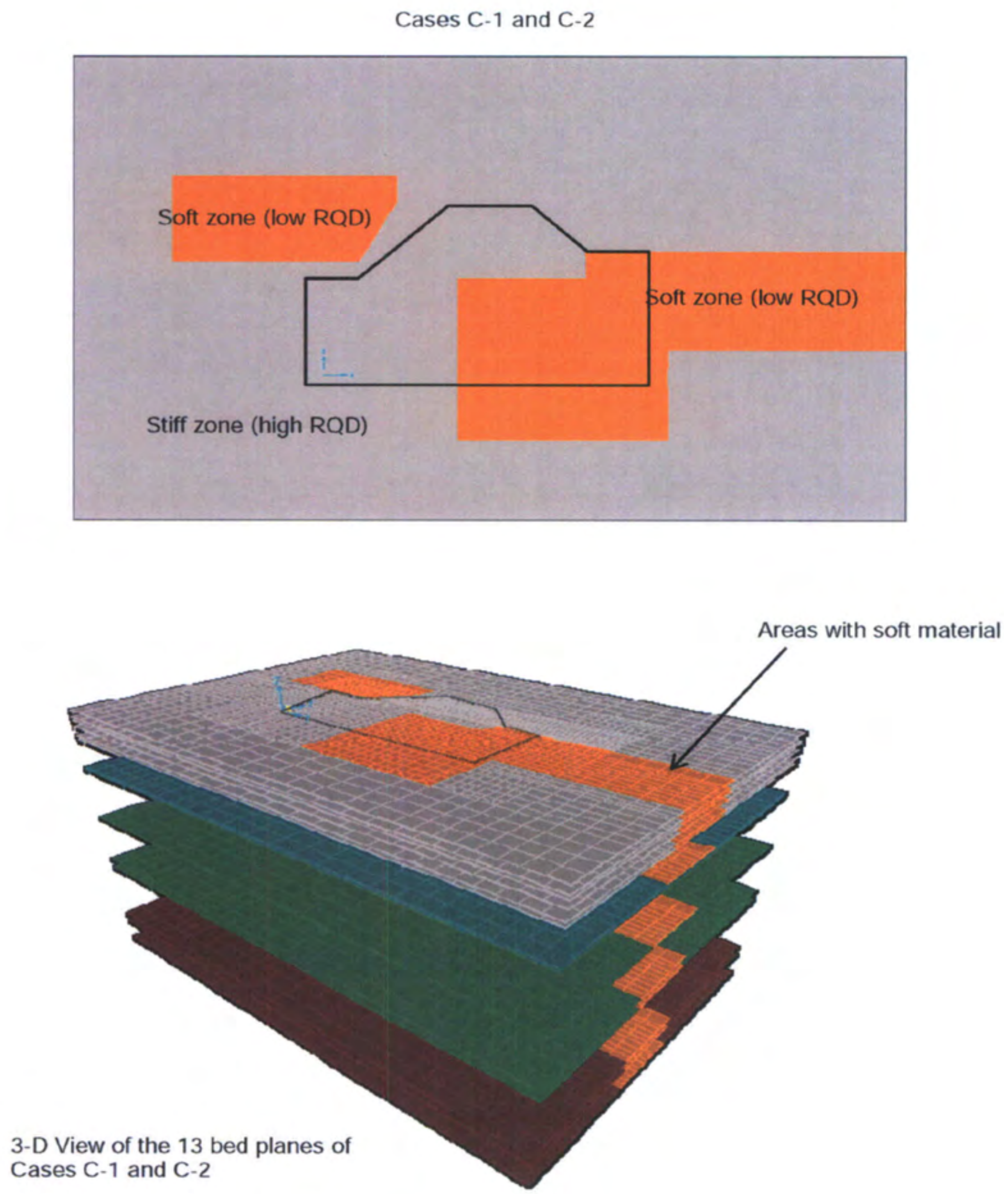


Figure RAI 02.05.04-11-02
Horizontal Variation of Soft Zones (Cases C-1 and C-2)

Figure 2.5.4-16. Distribution of Postulated Soft/Stiff Regions and Thirteen Soft 0.3-m (1-ft) Thick Bedding Layers (RAI 2.5.4-11 Response Figure 2.5.4-11-02)

The applicant considered three cases and compared the results to a base case of no soft zones. The applicant calculated the total and differential settlements for all cases to be less than approximately 1.3 cm (0.5 in). The applicant also noted that the largest total and differential settlements occur when the soft bedding planes were modeled as continuous soft layers.

The staff reviewed the results of the sensitivity analyses the applicant completed. Because the sensitivity analyses considered the lower bound values of elastic modulus for the layered Avon Park limestone profile, horizontal variations in elastic properties suggested by variations in RQD across the site, and postulated soft layers that may exist in the rock profile based on limited data in “no recovery” zones, the staff concludes that the sensitivity analyses are sufficient. Accordingly, the staff concludes that the total and differential settlements are acceptable because they are within the AP1000 DCD limits. Thus, RAI 2.5.4-11 is resolved.

2.5.4.4.10.3.5 Settlement Monitoring

The staff reviewed the applicant’s plans to monitor settlement at the LNP site. In RAI 2.5.4-10, the staff asked the applicant to estimate the settlement beneath the seismic Category II and nonsafety-related structures to observe the magnitude of differential settlement between structures, and describe the monitoring program proposed to ensure that the actual and differential settlements do not exceed the DCD settlement criteria.

In its June 8, 2009, response to RAI 2.5.4-10, the applicant provided a table of the estimated total settlements for the Turbine, Annex, Radwaste, and Diesel Generator Buildings for LNP Units 1 and 2 and noted that these total settlements result in differential settlements within acceptable limits. The applicant also revised the FSAR to describe the installation of settlement benchmarks at the nonsafety-related structures to measure the differential settlement during and after construction.

The staff notes that the AP1000 DCD limits the acceptable total settlement of structures to 15.2 cm (6.0 in), and differential settlement between structures to 7.6 cm (3.0 in). Likewise, differential settlement across the nuclear island foundation mat is limited to 1.3 cm (0.5 in) in 15.2 m (50 ft). The staff notes that the settlement estimates of the structures surrounding the nuclear island range from 0.3 to 0.5 cm (0.1 to 0.2 in). Because the average total settlement for the LNP Units 1 and 2 nuclear islands are 0.5 cm (0.2 in), the staff concludes that the total and differential settlement predictions are well within the allowable limits for total settlement, differential settlement between buildings, and tilt or distortional settlement within the nuclear island basemat. The staff also reviewed the changes to the FSAR, including the description of the installation of settlement benchmarks to measure the differential settlement and concludes that the method of measuring the differential settlement at the LNP site is adequate. Accordingly, RAI 2.5.4-10 is resolved.

2.5.4.4.10.3.6 Modeling Discontinuities in the FEM Analysis

The staff reviewed FSAR Section 2.5.4.1, which describes the fracture patterns at the site. In RAI 2.5.4-23, the staff asked the applicant to explain how it incorporated the information related to the observed local fracture patterns into the 3D FEM analysis. The staff also asked the applicant to: (1) clarify whether more closely-spaced fractures occur in the two outcrops discussed; (2) explain whether the fractures are characteristic of the fracture sets at the site location; and (3) explain how the design analyses account for settlement due to discontinuities.

In its June 23, 2009, response to RAI 2.5.4-23, the applicant used data from the Grout Test Program to confirm that the fracture orientation observed in the field is consistent with the regional orientation. The applicant stated that the fractures are typically less than 3 cm (0.1 ft) in width. The applicant stated that the four different cases modeled in the FEM sensitivity analysis combined the multiple 3 cm (0.1 ft) wide fractures into a 3 m (10 ft) wide fracture or orthogonal fracture set.

As modeled, the applicant stated that the 3.04 m (10 ft) fracture placed through the center of the nuclear island as shown in SER Figure 2.5.4-17 represents 100 fractures of 0.3 cm (0.1 ft) thickness and produces a maximum elastic settlement of 0.68 cm (0.27 in) and a differential settlement of 0.43 cm (0.17 in), which are less than the allowable settlement allowance of the AP1000 DCD.

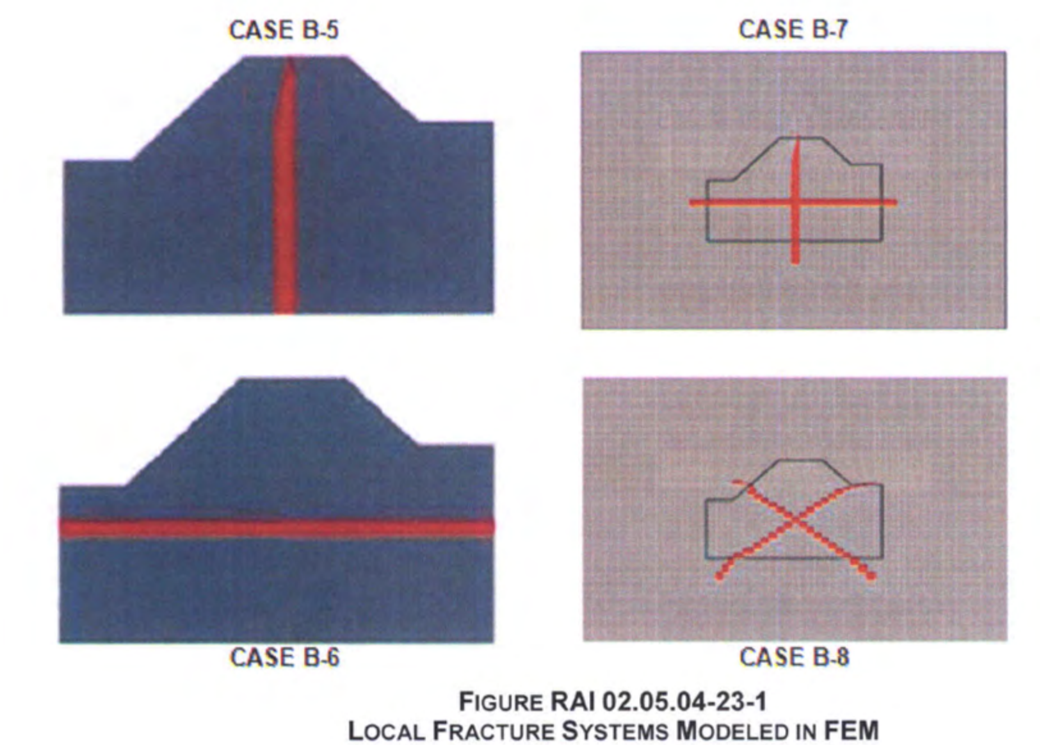


Figure 2.5.4-17. Representation of Local Fracture System in Finite Element Method Sensitivity Analyses (RAI Figure 2.5.4-23-1)

The staff reviewed the RAI response discussing the FEM results, and concludes that the aggregation thin 0.3 cm (0.1 ft) orthogonal fractures spaced a minimum of every 5.7 m (19 ft) into a large, orthogonal 3 m (10 ft) wide fracture placed through the center of the nuclear island represents the critical case and is conservative. The staff also notes that the subgrade surface preparation with dental concrete will eliminate all the vertical fractures to a minimum depth of 1.5 m (5 ft). The grouting program will also fill the larger joint openings and bedding plane voids down to El. -30.1 m (-99 ft), leaving little opportunity for large voids to exist within the grouted zone. Because voids greater than 0.9 m (3 ft) in lateral extent were not encountered, the staff concludes that the placement of continuous 3 m (10 ft) wide fracture in the patterns shown in SER Figure 2.5.4-16 are conservative because they are larger and more severe than any single discontinuous void. Accordingly, RAI 2.5.4-23 is resolved.

2.5.4.4.10.4 Lateral Earth Pressures

The staff reviewed FSAR Section 2.5.4.10.4 and FSAR Table 2.5.4.10-205, and determined that a sample calculation was needed in order to complete its review. In RAI 2.5.4-21, the staff asked the applicant to provide sample calculations for both the seismic at-rest and hydrodynamic pressures.

In its June 9, 2009, response to RAI 2.5.4-21, the applicant presented Wood's method (ASCE 4-98, 2000) to calculate the seismic at-rest pressure. The applicant stated that it used Wood's method because the walls are unyielding, which generates greater forces on the wall than those obtained by the Mononobe-Okabe method, which assumes the wall is free to move. A flexible-wall assumption underestimates the dynamic lateral forces generated on a rigid, unyielding wall. Using a Poisson's ratio of 0.3 for the undifferentiated sediments and a thrust factor of 0.98, the applicant concluded that the seismic induced lateral load is 470 kPa (9.83 ksf) and provided a figure showing the dynamic soil pressure resultant force as well as a seismic earth pressure diagram. For the hydrostatic water thrust, the applicant used Westergaard's equation (Westergaard, 1933) and provided a sample calculation and a figure illustrating the hydrostatic pressure.

The staff reviewed the sample calculations and figures and concludes the applicant used conservative material properties and conservative methods in the determination of the static and dynamic lateral earth pressures. Thus, RAI 2.5.4-21 is resolved.

2.5.4.4.10.4.1 Subsurface Instrumentation

In FSAR Section 2.5.4.10.3.5, the applicant addressed the construction and long-term instrumentation monitoring program. The staff considered the details of the construction monitoring plans, including the installation of piezometers to monitor drawdown of the water table and measure piezometric pressures on the bottom of the excavation during excavation and backfilling, heave points to measure heave of the foundation subgrade, and markers on the RCC bridging mat and nuclear island and surrounding structures to measure settlement during construction and until 90 percent of the expected settlement has occurred, or the rate of settlement stops. The staff concludes that long-term settlement will be negligible because of the strength of the foundation materials and the low levels of stress below the RCC bridging mat.

Nevertheless, post construction settlement will be monitored. The long-term monitoring program will be implemented after the construction monitoring program is completed and will monitor any long-term settlement occurring during the life of the structure. The applicant provided a conceptual plan in the FSAR and intended to finalize the instrumentation and monitoring plan during detailed design.

The staff reviewed the applicant's plans to monitor water levels during dewatering and excavation, bottom heave, and settlement of all the structures, and concludes that the applicant's conceptual plan adequately considered the construction features that require monitoring during construction.

2.5.4.4.10.4.2 Resolution of COL Information Items

The staff reviewed FSAR Section 2.5.4.10 and referenced the AP1000 DCD engineering criteria for settlement and bearing capacity. The staff also reviewed responses to related RAIs, and the references cited.

The staff concludes that the bearing capacity of the Avon Park limestone is sufficient to meet both the static and dynamic loading demands of the nuclear island and there is an adequate basis to resolve COL Information Item 2.5-10. The staff concludes that settlement, differential settlement of the nuclear island, and differential settlement between the nuclear island and surrounding structures due to either static or dynamic loading have been thoroughly examined and the estimated settlements are within the criteria set forth in the AP1000 DCD, and that there is an adequate basis to resolve COL Information Item 2.5-12 and COL Information Item 2.5-16. The staff further concludes that the use of Wood's Method to determine the lateral stresses was conservative and there is an adequate basis to resolve COL Information Item 2.5-11. Finally, the staff concludes that the instrumentation planned for monitoring during the construction phase and post-construction for the life of the plant is adequate and appropriate for the features being constructed and that there is an adequate basis to resolve COL Information Item 2.5-13.

2.5.4.4.10.4.3 Conclusion for Static Stability

In FSAR Section 2.5.4.10, the applicant considered the bearing capacity, settlement, lateral stresses, and performance monitoring at the LNP site. Based on the extensive analytical results, the staff concludes that the bearing capacity of the Avon Park limestone is sufficient to meet both the static and dynamic loading demands of the nuclear island. The response to the maximum static loads imposed by the nuclear island and overlying RCC bridging mat on the Avon Park limestone was satisfactory, limiting settlements to approximately 0.51 cm (0.2 in). It was also determined that a FS of 3 exists against bearing capacity failure, with or without a large (6 m (20 ft) cube-shaped) void, located below the grouted zone under the reactor building. Sensitivity analyses using closed form bearing capacity equations indicated acceptable FSs for lower bound material strength assumptions. Elastic settlement sensitivity analyses determined that under the most conservative of assumption of thirteen 0.3-m (1-ft) continuous soft layers located under the footprint of the nuclear island, settlement will be less than 1.3 cm (0.5 in).

NRC staff reviewed FSAR Section 2.5.4.10 and concludes that the applicant developed an accurate assessment of the static stability at the LNP site that addresses COL Information Items 2.5-10 through 2.5-13 and 2.5-16, including the minimum static bearing capacity; earth pressures; static stability of facilities; and subsurface instrumentation. The staff 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 FSAR Section 2.5.4.10 forms an adequate basis for the static stability at the site and meets the requirements of 10 CFR Part 50, Appendix A, GDC 2, and Appendix S; and 10 CFR 100.23.

2.5.4.4.11 Design Criteria

Based upon its review of LNP COL FSAR Section 2.5.4.11, including the AP1000 DCD design criteria, methods of analysis the applicant used, and the FS criteria, the staff concludes that the applicant applied good engineering judgment, state-of-the art analytical methods, appropriate design criteria and provided an adequate FS to ensure the safety of SSCs at the LNP site area. The staff concludes that the design values as described in LNP COL FSAR Section 2.5.4.11 form an adequate basis for the design criteria and meet the design values of the AP1000 DCD and the requirements of 10 CFR Part 50, Appendix A, GDC 2, and Appendix S.

2.5.4.4.12 Techniques to Improve Subsurface Conditions

In FSAR Section 2.5.4.12, the applicant summarized the measures that it will implement to improve the subsurface conditions. The applicant planned to grout the Avon Park limestone using grout holes, including inclined grout holes if deemed necessary, in multiple stages and install a reinforced concrete diaphragm wall surrounding the nuclear island to form an impermeable "bathtub" to minimize seepage into the excavation. The applicant will excavate in approximate 3 m (10 ft) depth increments to an El. of -7.3 m (-24 ft), at which point the subgrade will be cleaned, voids backfilled with dental concrete, and surface leveled prior to the construction of the 10.7 m (35 ft) thick RCC bridging mat.

The NRC staff reviewed FSAR Section 2.5.4.12. The applicant plans to remove weak, compressible, severely weathered Avon Park limestone and construct an RCC bridging mat. The staff concludes that the remedial measures the applicant proposed will improve the foundation conditions and provide a uniformly strong base of rock upon which the RCC bridging mat is founded to support the nuclear island. Though meant only to reduce seepage into the excavation during construction, the presence of the diaphragm wall and grouted limestone between an El. of -7.3 and -30.1 m (-24 and -99 ft) will also add to the future stability of the site by reducing the opportunity for future karst development.

Based upon its review of LNP COL FSAR Section 2.5.4.12, the staff concludes that the applicant adequately described its plans for improving and monitoring the subsurface conditions at the LNP site. The staff concludes that the methods of improvement and monitoring plans as described in FSAR Section 2.5.4.12 form an adequate basis for the improvement of subsurface conditions at the site and meet the requirements of 10 CFR Part 50, Appendix A, GDC 2, and Appendix S.

2.5.4.5 *Post Combined License Activities*

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

2.5.4.6 *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 stability of subsurface materials and foundations, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

Based on its review of LNP COL FSAR Section 2.5.4 and the applicant's responses to the RAIs, the staff concludes that the applicant adequately determined the engineering properties of the soil and rock underlying the LNP COL site through its field and laboratory investigations. The staff concludes that the applicant used the latest field and laboratory methods, in accordance with RG 1.132, Revision 2; RG 1.138, Revision 2; and RG 1.198, to determine the required site-specific engineering properties for the LNP site and to ensure that these properties met the design criteria outlined in the AP1000 DCD.

Based on the information in the FSAR, the staff concludes that the subsurface profile underlying the COL site has been properly characterized, that state-of-the-art analytical methods were used with conservative input values to determine factors of safety, and that the applicant considered all aspects of the foundation design that could impact the SSCs. Specifically, the staff concludes that the applicant adequately determined: (1) the soil and rock 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 soils; and (4) the static stability, including the bearing capacity, settlement, and lateral earth pressures.

The staff concludes that the applicant provided sufficient information in LNP COL 2.5-5 through LNP COL 2.5-13, and LNP COL 2.5-16 to adequately address the COL information items pertaining to FSAR Section 2.5.4.

The staff concludes that FSAR Section 2.5.4 is acceptable and meets the requirements of 10 CFR Part 50, Appendix A (GDC 2) and Appendix S; and 10 CFR 100.23.

2.5.5 Stability of Slopes

2.5.5.1 *Introduction*

LNP COL FSAR Section 2.5.5 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 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 requirements prescribed within the "Contents of Application" sections of the applicable subparts to 10 CFR Part 52.

2.5.5.2 *Summary of Application*

Section 2.5 of the LNP COL FSAR, Revision 9, incorporates by reference Sections 2.5.5 and 2.5.6 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Sections 2.5.5, the applicant provided the following:

AP1000 COL Information Items

- LNP COL 2.5-14

The applicant provided additional information in LNP COL 2.5-14 to address COL Information Item 2.5-14 (COL Action Item 2.5.5-1), which addresses the static and dynamic stability of site-specific soil and rock slopes with regard to how their failure could adversely affect the nuclear island.

- LNP COL 2.5-15

The applicant provided additional information in LNP COL 2.5-15 to address COL Information Item 2.5-15 (COL Action Item 2.5.6-1), which addresses the static and dynamic stability of site-specific embankments and dams with regard to how their failure could adversely affect the nuclear island.

The applicant developed FSAR Section 2.5.5 for evaluation of slope stability at the LNP site based on information derived from site investigations, geotechnical characterization studies, and excavation and backfill profiles presented in FSAR Sections 2.5.4.1 thorough 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, in situ testing, laboratory testing, and geophysical surveys.

2.5.5.2.1 Slope Characteristics

FSAR Section 2.5.5 describes the lack of existing permanent slopes, or dams, both natural and man-made, at the LNP site. The applicant stated that the only sloping ground at the LNP site consists of minor elevation changes to accomplish positive drainage away from the nuclear islands. The applicant also stated that the AP1000 does not utilize safety-related dams and that no dams exist that could affect the nuclear islands. The applicant concluded that no permanent slopes or dams exist for which failure would adversely affect the safety-related structures of LNP Units 1 and 2.

2.5.5.3 *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 stability of slopes are given in Section 2.5.5 of NUREG-0800.

The applicable regulatory requirements for reviewing the applicant's discussion of stability of slopes are:

- 10 CFR Part 50, Appendix A, GDC 2, 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, as it applies to the design of nuclear power plant SSCs important to safety to withstand the effects of earthquakes.

- 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 acceptance criteria from Section 2.5.5 of NUREG-0800 are as follows:

- Slope Characteristics: In meeting the requirements of 10 CFR Parts 50 and 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 Category I facilities; and (3) a summary and description of groundwater, seepage, and high and low groundwater conditions.

- Design Criteria and Analyses: In meeting the requirements of 10 CFR Parts 50 and 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: In meeting the requirements of 10 CFR Parts 50 and 100, the applicant should describe the borings and soil testing carried out for slope stability studies and dam and dike analyses.

- Compacted Fill: In meeting 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, Revision 4; RG 1.132, Revision 2; RG 1.138, Revision 2; RG 1.198; and RG 1.206.

2.5.5.4 *Technical Evaluation*

The NRC staff reviewed Section 2.5.4 of the LNP 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 LNP COL application.

The staff reviewed the information in the LNP COL FSAR:

AP1000 COL Information Items

- LNP COL 2.5-14

The NRC staff reviewed LNP COL 2.5-14 in Section 2.5.5 of the LNP COL FSAR, related to the stability of all earth and rock slopes both natural and manmade (cuts, fill, embankments, dams, etc.) whose failure, under any of the conditions to which it could be exposed during the life of the plant, could adversely affect the safety of the plant. The COL information item in AP1000 DCD Section 2.5.5 states:

> Combined License applicants referencing the AP1000 design will address site-specific information about the static and dynamic stability of soil and rock slopes, the failure of which could adversely affect the nuclear island.

With respect to COL Information Item 2.5-14, the applicant stated that there are no soil or rock slopes the failure of which could adversely affect the safety-related structures at the LNP site. The applicant stated that the only slopes consist of minor grading for drainage away from the nuclear islands at LNP Units 1 and 2. The staff reviewed the site plans and concludes that the applicant has appropriately characterized the site conditions. The only sloping boundaries are related to drainage around the nuclear islands and these slopes do not constitute a slope stability concern. The staff concludes that there are no slopes or dams at the site that could adversely affect LNP Units 1 and 2. The staff concludes that the applicant met the criteria of COL Information Item 2.5-14.

- LNP COL 2.5-15

The NRC staff reviewed LNP COL 2.5-15 in Section 2.5.5 of the LNP COL FSAR, related to the stability of embankments and dams, the failure of which could adversely affect the plant. The COL information item in AP1000 DCD Section 2.5.6 states:

> Combined License applicants referencing the AP1000 design will address site-specific information about the static and dynamic stability of embankments and dams, the failure of which could adversely affect the nuclear island.

Regarding COL Information Item 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 LNP site. The staff considered the results of site investigations, as well as the applicant's assertion that there are no man-made earthen or rock dams present at the site. The staff concludes that there are no dams or embankments, which might adversely affect Units 1 and 2, and therefore the applicant addressed the criteria of COL Information Item 2.5-15 for the LNP site.

2.5.5.5 *Post Combined License Activities*

There are no post-COL activities associated with this FSAR section.

2.5.5.6 *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 stability of slopes, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

As set forth above, the applicant presented and substantiated information to establish the stability of all earth and rock slopes, both natural and manmade at the plant site. The staff reviewed the site investigations performed for LNP Units 1 and 2, and the site plans to confirm that there were no slopes or dams that could adversely affect the safe operations of the LNP Units 1 and 2. The staff concludes that the applicant provided sufficient information to addresses COL Information Items 2.5-14 and 2.5-15. The staff concludes that the relevant information presented in LNP COL FSAR Section 2.5.5 is acceptable and meets the requirements of 10 CFR Part 50, Appendix A, GDC 2; 10 CFR Part 50, Appendix S; and 10 CFR 100.23.

3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.1 Conformance with NRC General Design Criteria

Section 3.1 of the Levy Nuclear Plant (LNP) combined license (COL) Final Safety Analysis Report (FSAR), Revision 9, 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 LNP COL FSAR, the applicant provided the following:

Departure

- LNP DEP 6.4-1

The applicant provided additional information about LNP 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 LNP 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 reviewed the application and checked the referenced DCD to ensure that 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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements. Section 21.2 of this report evaluates the departure from the DCD provided in LNP DEP 6.4-1.

3.2 Classification of Structures, Components, 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 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

[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).

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. The regulatory treatment of nonsafety systems (RTNSS) process is applied to define seismic requirements for SSCs that are nonsafety-related but perform risk-significant functions.

The methodology in the referenced AP1000 DCD classifies SSCs into three categories: seismic Category I, seismic Category II and nonseismic (NS). Those plant features that are designed to remain functional, if an SSE occurs, are designated seismic Category I. Seismic Category I applies to both functionality and integrity, and seismic Category II applies only to integrity. NS items located in the proximity of safety-related items, the failure of which during an SSE could result in the loss of function of safety-related items, 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*

Section 3.2 of the LNP COL FSAR, Revision 9, incorporates by reference Section 3.2 of the AP1000 DCD, Revision 19. Section 3.2 of the DCD includes Section 3.2.1.

In addition, in LNP COL FSAR Section 3.2, the applicant provided the following:

Departures

- LNP DEP 3.2-1

The applicant provided additional information about LNP 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 LNP DEP 3.2-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.1 of the SER.

Supplemental Information

- LNP Supplement (SUP) 3.2-1

The applicant provided supplemental information by adding text to the end of DCD Section 3.2.1, "Seismic Classification," stating that there are no safety-related SSCs at LNP Units 1 and 2 outside the scope of the DCD, except for roller compacted concrete (RCC), which is classified as a seismic Category I, safety-related structure. The applicant also states that the nonsafety-related SSCs outside the scope of the DCD are classified as NS.

- LNP SUP 3.2-2

The applicant provided supplemental information by adding text to the end of AP1000 DCD Section 3.2.1.3, "Classification of Building Structures," stating that the seismic classification of the makeup water pump house, Unit 1 freshwater raw water pump house, Unit 2 freshwater raw water pump house, Unit 1 potable water pump house, and Unit 2 potable water pump house are provided in LNP COL FSAR Table 3.2-201.

3.2.1.3 *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 seismic classification are given in Section 3.2.1 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 supplemental information of defining the scope of safety-related SSCs is established in General Design Criteria (GDC) 2, "Design Bases for Protection Against Natural Phenomena," which requires that all SSCs 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 NRC staff reviewed Section 3.2 of the LNP 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 seismic classification. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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 DCD 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 LNP COL FSAR:

Supplemental Information

- LNP SUP 3.2-1

The NRC staff reviewed LNP SUP 3.2-1, related to the seismic classification of safety-related SSCs included under Section 3.2.1 of the LNP COL FSAR, which states that there are no safety-related SSCs outside the scope of the DCD, except for roller compacted concrete (RCC), which is classified as a seismic Category I, safety-related structure at LNP Units 1 and 2. Therefore, the seismic classification is acceptable.

- LNP SUP 3.2-2

The NRC staff reviewed LNP SUP 3.2-2 included in LNP COL FSAR Section 3.2.1.3, related to the seismic classification of the makeup water pump house, Unit 1 freshwater raw water pump house, Unit 2 freshwater raw water pump house, Unit 1 potable water pump house, and Unit 2 potable water pump house, which are classified as NS and are provided in LNP COL FSAR Table 3.2-201. Therefore, the seismic classification is acceptable.

The following portion of this technical evaluation section is reproduced from Section 3.2.1.4 of the VEGP SER:

> *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.*

<u>Seismic Classification of Other Site-Specific SSCs</u>

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 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 seismic classification, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP 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:

- LNP 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.

- LNP SUP 3.2-1 is acceptable because the LNP COL FSAR states that there are no safety-related SSCs outside the scope of the AP1000 DCD, except for RCC, which is classified as a seismic Category I, safety-related structure. The LNP COL FSAR also states that the nonsafety-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.

- LNP SUP 3.2-2 is acceptable because the seismic classification of the makeup water pump house, Unit 1 freshwater raw water pump house, Unit 2 freshwater raw water pump house, Unit 1 potable water pump house, and Unit 2 potable water pump house, which are classified as NS are provided in LNP COL FSAR Table 3.2-201. 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 AP1000 Classification Systems (Related to RG 1.206, Section C.III.1, Chapter 3, C.I.3.2.2, "System Quality Group Classification")

3.2.2.1 *Introduction*

The system and component quality group classification addresses, in part, the general design criterion that nuclear power plant SSCs 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 nonsafety-related but perform risk significant function.

The system and component quality group classification in combination with the RTNSS process define appropriate classifications, codes and standards and special treatment 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 systems and the appropriate quality standards.

3.2.2.2 *Summary of Application*

Section 3.2 of the LNP COL FSAR, Revision 9, incorporates by reference Section 3.2 of the AP1000 DCD, Revision 19. Section 3.2 of the DCD includes Section 3.2.2.

In addition, in LNP COL FSAR Section 3.2, the applicant provided the following:

Supplemental Information

- LNP SUP 3.2-1

The applicant provided supplemental information by adding text stating that there are no safety-related SSCs at LNP Units 1 and 2 outside the scope of the DCD, except for roller compacted concrete (RCC), which is classified as a seismic Category I, safety-related structure.

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 the Commission regulations for the system quality group classification are given in Section 3.2.2 of NUREG-0800.

The basis for acceptance of the supplemental information that defines the scope of safety-related SSCs is established in RG 1.26 and applicable American Society of Mechanical Engineers (ASME) Codes and industry standards, which provide assurance 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 NRC staff reviewed Section 3.2 of the LNP 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 system quality group classification. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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 DCD 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 staff reviewed the information in the LNP COL FSAR:

Supplemental Information

- LNP SUP 3.2-1

The NRC staff reviewed LNP SUP 3.2-1 related to the seismic classification of safety-related SSCs included under Section 3.2.2 of the LNP COL FSAR, which states that there are no safety-related SSCs outside the scope of the DCD at LNP Units 1 and 2, except for RCC, which is classified as a seismic Category I, safety-related structure.

The NRC staff reviewed LNP SUP 3.2-1 related to quality group classification of systems included under Section 3.2.2 of the LNP COL FSAR. LNP SUP 3.2-1 identifies that there are no safety-related structures, systems, or components outside the scope of the DCD, except for RCC which is classified as a seismic Category I, safety-related structure. Quality Group addressed in RG 1.26 is limited to pressure-retaining systems and their supports and does not apply to structures. Structures are specifically excluded from the scope of the NUREG-0800 Section 3.2.2 review. As discussed below, there are no site-specific nonsafety-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 LNP COL FSAR Section 3.2.

The following portion of this technical evaluation section is reproduced from Section 3.2.2.4 of the VEGP SER:

> *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.*

<u>Codes and Standards</u>

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.

<u>Consistency with RG 1.26, Revision 4</u>

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 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 system quality group classification, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP 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:

- LNP SUP 3.2-1 is acceptable with regard to quality group classifications because no change was made to the quality group classifications in Section 3.2 and there are no site-specific nonsafety-related SSCs outside the scope of the AP1000 DCD. Quality Group does not apply to the site-specific RCC that is classified as a seismic Category I, safety-related structure. Therefore, 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, as required by GDC 2 in Appendix A to 10 CFR Part 50. This states that SSCs 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 SER, 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 SER.

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 gust factors from the standpoint of use in defining the input parameters for the appropriate structural design criteria for wind loading.

3.3.1.2 *Summary of Application*

Section 3.3 of the LNP COL FSAR, Revision 9, incorporates by reference Section 3.3 of the AP1000 DCD, Revision 19. Section 3.3 of the DCD includes Section 3.3.1.

In addition, in LNP COL FSAR Section 3.3.1, the applicant provided the following:

AP1000 COL Information Items

- LNP COL 3.3-1

The applicant provided additional information in LNP COL 3.3-1 to address COL Information Item 3.3-1 (COL Action Item 3.3.2.2-1) by stating that the wind velocity characteristics for the LNP site are given in LNP COL FSAR Section 2.3.1.2.2. The applicant states 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 states 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 LNP COL 3.3-1 relating to design tornado site characteristics and the effects of wind on the safety-related SSCs due to failures in an adjacent AP1000 plant is reviewed in SER Section 3.3.2.

- LNP COL 3.5-1

The portion of LNP COL 3.5-1 included in LNP COL FSAR Section 3.3.1 is identical to the information added by LNP COL 3.3-1, and is addressed by the staff in its evaluation of LNP COL 3.3-1 in this SER section. The additional information in LNP COL 3.5-1 included in LNP COL FSAR Section 3.5 is addressed in Section 3.5 of this SER.

3.3.1.3 *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 wind loadings are given in Section 3.3.1 of NUREG-0800.

The regulatory basis for LNP COL 3.3-1 is 10 CFR Part 50, Appendix A, GDC 2, and the regulatory guidance is in RG 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," Revision 1, which states that 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.

3.3.1.4 *Technical Evaluation*

The NRC staff reviewed Section 3.3 of the LNP 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 wind loadings. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the LNP COL FSAR:

AP1000 COL Information Item

- LNP COL 3.3-1

The NRC staff reviewed LNP COL 3.3-1 related to design wind loads applied on safety-related SSCs included under Section 3.3.1.1 of the LNP COL FSAR.

The commitment was also captured as COL Action Item 3.3.2.2-1 in NUREG-1793, Appendix F, “Combined License Action Items,” which states:

> COL applicants referencing the AP1000 certified design will address site interface criteria for wind and tornadoes.

The applicant states in LNP COL 3.3-1 that the wind velocity characteristics for LNP Units 1 and 2, are given in LNP COL FSAR Section 2.3.1.2.2. The applicant states that these values are bounded by the DCD design wind velocity values for the standard AP1000 plant.

For consistency, the NRC staff reviewed the resolution to the site specific information item LNP COL 3.3-1 on the site related parameters with those contained in COL FSAR Section 2.3.1.2.2. ASCE 7-05 was used by the staff to validate wind design information in relation to the LNP site. The applicant has presented consideration to ASCE 7-05 in Table 2.0.201, “Comparison of AP1000 DCD Site Parameters and LNP Site Characteristics.” In this Table, the applicant stated that the AP1000 Design Wind Speed of 145 mph envelopes the LNP Maximum 50-year and 100-year return 3-second gust wind speeds based on Table C6-7 of ASCE 7-05. However, historic records show at least two hurricanes that exceed the wind speed for the AP1000 Design. For this, the staff issued RAI 2.3.1-8.

In its April 1, 2009, response to RAI 2.3.1-8, the applicant stated that the 3-second gust wind speed will be increased from 139 mph to 185 mph. Exceeding the 145 mph design wind speed for the AP1000 Design will require site specific calculations to convert winds in pressure loads over safety related structures, following ASCE 7-05. Also, the site specific wind pressure load shall be considered for the structural analysis done in Section 3.8. This analysis of the site specific wind pressure load was not included with the applicant’s response to RAI 2.3.1-8. For this, staff issued RAI 3.3.1-1.

The applicant’s August 24, 2009, response to RAI 3.3.1-1 stated that the 50-year and 100-year wind speeds for the Levy Site were 120 mph and 128 mph, respectively. Also, the applicant indicated that the most severe historically reported wind speed for the site is 144 mph, which is enveloped by the AP1000 Design wind speed of 145 mph. Given that this historically recorded wind speed was just 1 mph less than the AP1000 design wind speed, the staff issued RAI 3.3.1-2 to understand how historic wind speed was considered, with sufficient margin, as required by 10 CFR 50 Appendix A, GDC 2. In its January 10, 2010, response to RAI 3.3.1-2, the applicant provided additional wind speed data which places the 144 mph recorded wind speed outside the Levy County site.

The exceeding wind speeds are part of the site-specific meteorological evaluation of FSAR Chapter 2.3. The staff verified the applicant’s site-specific meteorological information, provided in the revised RAI response for RAI 2.3.1-8, to confirm that the different wind speeds exceeding the AP1000 design wind were outside of the 100 nautical mile radius described in NUREG-0800 Section 2.3. Also, the staff concluded that the applicant's approach to determine the site specific wind speeds, and method of informing severe weather conditions around the vicinity of the site, was acceptable because it is consistent with guidance in Section 2.3.1 of NUREG-0800. RAIs 2.3.1-8, 3.3.1-1, and 3.3.1-2 are resolved. Additional information regarding the analysis and adequacy of site specific wind speeds is included in Section 2.3 of this SER.

Based on the above review, the staff finds that the information supplied to close Action Item 3.3-1 for site interface criteria for wind by the applicant is adequate in meeting the NRC regulatory requirements.

3.3.1.5 *Post Combined License Activities*

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

3.3.1.6 *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 wind loadings, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP COL FSAR is acceptable and meets the requirements of GDC 2. The staff based its conclusion on the following:

- LNP 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 SER, and is 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 Section 3.3.2, "Tornado Loadings," of the AP1000 DCD. Section 3.3.2 of the AP1000 DCD addresses tornado loadings for seismic Category I structures using applicable tornado design parameters to determine forces on structures as explained in Section 3.3.1.2 of the AP1000 DCD. Also in Section 3.3.2.1 of the AP1000 DCD, it is stated that the estimated probability of tornado wind speeds to be greater than the design basis tornado is between 10^{-6} and 10^{-7} per year for an AP1000 at a "worst location" anywhere within the contiguous United States.

The specific areas of review in accordance with Section 3.3.2 of NUREG-0800 include:

- the tornado wind translational and rotational speeds
- the tornado-generated atmospheric pressure change
- the spectrum of tornado-generated missiles

Similar considerations to hurricanes in coastal and tropical regions, per RG 1.221 "Design Basis Hurricanes and Hurricane Missiles for Nuclear Power Plants," include:

- the hurricane wind speeds
- the spectrum of hurricane-generated missiles

3.3.2.2 *Summary of Application*

Section 3.3 of the LNP COL FSAR, Revision 9, incorporates by reference Section 3.3 of the AP1000 DCD, Revision 19. Section 3.3 of the DCD includes Section 3.3.2.

In addition, in LNP COL FSAR Section 3.3.2, the applicant provided the following:

AP1000 COL Information Items

- LNP COL 3.3-1

The applicant provided additional information in LNP COL 3.3-1 to resolve COL Information Item 3.3-1 (COL Action Item 3.3.2.2-1). In LNP COL 3.3-1, the applicant states that tornado characteristics for LNP Units 1 and 2, given in Section 2.3.1.2.2 of the LNP COL FSAR 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 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 unit. The portion of LNP COL 3.3-1 relating to design wind velocity characteristics is reviewed in SER Section 3.3.1.

- LNP COL 3.5-1

The portion of LNP COL 3.5-1 included in LNP COL FSAR Section 3.3.2 is identical to the information added by LNP COL 3.3-1, and is addressed by the staff in its evaluation of LNP COL 3.3-1 in this SER section. The additional information in LNP COL 3.5-1 included in LNP COL FSAR Section 3.5 is addressed in Section 3.5 of this SER.

- STD COL 3.3-1

The information provided in LNP COL FSAR Section 3.3.2.3 to address Standard (STD) COL 3.3-1 is identical to the information provided in LNP COL FSAR Section 3.3.2.3 to address LNP COL 3.5-1. As noted above, the portion of LNP COL 3.5-1 included in LNP COL FSAR Section 3.3.2 is addressed by the staff in its evaluation of LNP COL 3.3-1 in this SER section. Therefore, STD COL 3.3-1 will not be addressed further in this SER.

3.3.2.3 *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 tornado loading are given in Section 3.3.2 of NUREG-0800.

Acceptance of the information addressing LNP 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 using acceptable procedures must meet the requirements of 10 CFR Part 50, Appendix A, GDC 2, which states that 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 NRC staff reviewed Section 3.3.2 of the LNP 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 tornado loading. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the LNP COL FSAR:

AP1000 COL Information Item

- LNP COL 3.3-1

The NRC staff reviewed LNP COL 3.3-1 included under Sections 3.3.2 and 3.5.1 of the LNP COL FSAR. Specific information provided by the applicant to address COL Action Item 3.3.2.2-1 includes development of site-specific parameters and verification of bounding conditions, site arrangement and building construction. This information is provided to satisfy the commitment documented in Appendix F of NUREG-1793, which states:

> COL applicants referencing the AP1000 certified design will address site interface criteria for winds and tornadoes.

In LNP COL 3.3-1, the applicant states that the tornado characteristics for LNP Units 1 and 2, given in Section 2.3.1.2.2 of the LNP COL FSAR, 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 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 unit.

In Section 2.3.1 of this SER, 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 LNP site are in compliance with GDC 2.

The scope of LNP COL 3.3-1 also includes the effects of wind and tornado on the safety-related SSCs due to failure of nonsafety-related buildings in an adjacent AP1000 plant and LNP Units 1 and 2. The applicant states that these effects are bounded by the evaluation of the buildings and structures in a single unit.

In order 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 were offered three options in Section 3.3.2.3 of the AP1000 DCD:

(1) Design the adjacent nonsafety-related structure to the design basis tornado loading.

(2) Analyze the effect of failure of adjacent nonsafety-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 LNP COL 3.3-1, the applicant used Option (2), indicating that the effects of wind and tornado on the safety-related SSCs due to failure of an adjacent nonsafety-related building are bounded by the evaluation of the structures in a single unit at LNP. The analysis of the impact of building collapse on the NI structures is in Section 3.7.2.8 of the AP1000 DCD. The staff's review of this analysis is provided in NUREG-1793 and its supplements.

RG 1.221 provides new guidance that the NRC staff considers acceptable for use in selecting the design-basis hurricane wind speed and hurricane-generated missiles that a new nuclear power plant should be designed to withstand to prevent undue risk to public health and safety. As described in Section 2.3 of this SER, the staff compared the information provided in FSAR 2.3 regarding hurricane winds against the information in RG 1.221. In response to RAI 2.03.01-20, with regards to hurricane wind and hurricane missile effects on safety-related structures, the applicant provided a comparison between DCD Tier 1 tornado generated missiles and those in RG 1.221. The evaluation of the three missiles (1-in diameter steel sphere, 6-in diameter pipe, and 4,000 lb automobile) compared velocities generated from both hurricane and tornado events. For the 1-in diameter steel sphere, both horizontal and vertical velocities were bounded by the tornado event. For the 6-in diameter pipe, the applicant performed a local analysis calculation to confirm that the wall thicknesses of the NI structures are sufficient to prevent penetration and scabbing generated by the pipe's impact. This calculation was verified by the staff to be acceptable, based on a staff confirmatory analysis. Impact energy was also evaluated by the applicant; however the staff was more concerned with the impact analysis for the automobile, due to its higher impact kinetic energy relative to the other analyzed missiles. The applicant mentioned that a confirmatory calculation were to be provided separately from the RAI response.

On February 9, 2012, the staff performed an audit of the applicant's calculation (# LNG-1000-S3R-001) related to the impact analysis of the automobile missile. Five different wall samples were used to estimate the shear stresses, impact energy, loads, and ductility. The applicant identified assumptions such as, not using Dynamic Impact Factor and corner impact to reduce shear perimeter, which the staff found to be conservative. Design codes and subject matter references including ACI-349 "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary," and a Bechtel Topical Report (R. B. Linderman, et al., "Design of Structures for Missile Impact," BC-TOP-9-A, Revision 2, 1974), were used to demonstrate that shear stresses and ductility are still within code acceptable limits. The staff determined that the applicant's approach for estimating structural global response is acceptable because the applicant's approach is consistent with standard engineering practices. Additionally, the staff performed a confirmatory calculation to verify the adequacy of the LNP NI structures against the hurricane generated automobile missile. Based on the staff's confirmatory analysis, the applicant's evaluation of the hurricane generated missile effects on the NI structures for the LNP site was found to be acceptable.

Based on the above discussion, the NRC staff considers LNP COL 3.3-1 to be 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 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 tornado and hurricane loading, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP COL FSAR Section 3.3.2 is acceptable and meets the requirements of 10 CFR Part 50, Appendix A, GDC 2. The staff based its conclusion on the following:

LNP COL 3.3-1, as it relates to design tornado loads, is acceptable based on the design tornado site characteristics, which are reviewed in Section 2.3 of this SER, match the AP1000 DCD design tornado site characteristics and therefore comply with GDC 2. LNP COL 3.3-1, as it relates to the effects of wind and tornado on the safety-related SSCs due to failure of nonsafety-related buildings in an adjacent AP1000 plant and LNP, is acceptable because the applicant incorporated by reference acceptable methodology from AP1000 DCD Section 3.7.2.8.

LNP COL 3.3-1, as it relates to design hurricane loads, is acceptable based on the design hurricane wind and hurricane missile site characteristics, reviewed in Section 2.3 of this SER, matching the RG 1.221 site characteristics and, therefore, complies with GDC 2.

LNP COL 3.5-1, as it relates to hurricane missiles that are more energetic than the tornado missiles in the AP1000 DCD, is acceptable based on the evaluation of hurricane missile effects on the LNP safety-related structures in response to RAI 2.03.01-20, and therefore, complies with GDC 2.

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*

Section 3.4 of the LNP COL FSAR, Revision 9, incorporates by reference Section 3.4 of the AP1000 DCD, Revision 19. Section 3.4 of the DCD includes Section 3.4.1.

In addition, in LNP COL FSAR Section 3.4, the applicant provided the following:

AP1000 COL Information Item

- LNP COL 3.4-1

The applicant provided additional information in LNP COL 3.4-1 to resolve COL Information Item 3.4-1 (COL Action Item 3.4.1.1-1), which addresses plant-specific information on site-specific flooding hazards protective measures. LNP COL 3.4-1, in LNP COL FSAR Section 3.4.1.3, “Permanent Dewatering System,” states that no permanent dewatering system is required because site groundwater levels are 2 feet (ft) or more below site grade level as described in LNP COL FSAR Section 2.4.12.5.

LNP COL 3.4-1, in LNP COL FSAR Section 3.4.3, “Combined License Information,” states that the site-specific water levels given in LNP COL FSAR Section 2.4 satisfy the interface requirements identified in AP1000 DCD Section 2.4.

3.4.1.3 *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 flood protection measures are given in Section 3.4.1 of NUREG-0800.

Further, the acceptance criteria associated with the relevant requirements of the Commission regulations for the identification of floods and flood design considerations are given in Section 3.4.1.II of NUREG-0800.

3.4.1.4 *Technical Evaluation*

The NRC staff reviewed Section 3.4 of the LNP 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 flood protection measures. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the LNP COL FSAR:

AP1000 COL Information Item

- LNP COL 3.4-1

The NRC staff reviewed LNP COL 3.4-1, which addresses the permanent dewatering system and site-specific water levels in Sections 3.4.1.3 and 3.4.3 of the LNP COL FSAR, respectively.

The applicant provided additional information in LNP COL 3.4-1 to address COL Information Item 3.4-1. COL Information Item 3.4-1 states:

> The Combined License [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 Combined License [COL] applicant may propose protective measures as discussed in Section 2.4.

The commitment was also captured as COL Action Item 3.4.1.1-1 in Appendix F of NUREG-1793, which states:

> The COL applicant will evaluate events leading to potential flooding and demonstrate that the design will fall within the values of these site parameters.

In LNP COL FSAR Section 3.4, the applicant provided the following plant-specific information to resolve COL Information Item 3.4-1 (COL Action Item 3.4.1.1-1) on site-specific flooding hazards protective measures:

- LNP COL 3.4-1, in LNP COL FSAR Section 3.4.1.3, "Permanent Dewatering System," states that no permanent dewatering system is required because site groundwater levels are 2 ft or more below site grade level as described in LNP COL FSAR Section 2.4.12.5.

- LNP COL 3.4-1, in LNP COL FSAR Section 3.4.3, "Combined License Information," states that the site-specific water levels given in LNP COL FSAR Section 2.4 satisfy the interface requirements identified in AP1000 DCD Section 2.4.

In Section 2.4.12 of this SER, the staff accepted the LNP applicant's position that no permanent dewatering system is required and that the site-specific groundwater characteristics for the LNP site fall within the Tier 1 and Tier 2 DCD parameter values. Therefore, the staff concludes that the site-specific information in LNP 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 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 appropriate consideration of flood causing mechanisms and flood protection measures as described in section 2.4.2 and 2.4.10 of the FSAR. The NRC staff has determined that there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP COL FSAR is acceptable and meets the regulatory guidance in Sections 2.4.12 and 3.4.1 of NUREG-0800. The staff based its conclusion on the following:

- LNP COL 3.4-1, is acceptable based on 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 SER regarding the adequacy of the site-specific groundwater levels.

3.4.2 Analytical and Test Procedures (Related to RG 1.206, Section C.III.1, Chapter 3, C.I.3.4.2, "Analysis Procedures")

Analysis methods and 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.

Section 3.4 of the LNP COL FSAR, Revision 9, incorporates by reference, with no departures or supplements, Section 3.4.2, "Analytical and Test Procedures," of Revision 19 of the AP1000 DCD. Section 3.4.2 of the AP1000 DCD states that the analytical approach for external and internal flooding events is described in DCD Section 3.4.1.2, "Evaluation of Flooding Events." 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 LNP COL application are documented in NUREG-1793 and its supplements.

3.5 <u>Missile Protection</u>

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). Once a potential missile is identified, its statistical significance is determined (a significant missile is one which 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*

SSCs important to safety are protected against internally generated missiles (outside containment), in accordance with Section 3.5.1.1 of NUREG-0800. The missiles generated outside containment by rotating or pressurized (high-energy fluid system) equipment are included.

The design credits only safety-related systems to establish and maintain safe shutdown conditions. The safety-related systems and components needed to bring the plant to safe shutdown, including the main control room and the recirculating service water system, are located inside the containment shield building and the auxiliary building. Both buildings are seismic Category I NI structures having thick structural concrete walls that provide internal and external missile protection. No nonsafety-related systems or components that require protection from missiles are housed in these buildings.

All SSCs that are necessary to perform safety functions are to be protected against damage from the following:

- Internally generated missiles (outside containment)
- Internally generated missiles (inside containment)
- Turbine missiles
- Missiles generated by tornadoes and extreme winds
- Site proximity missiles (except aircraft)
- Aircraft hazards

3.5.1.2 *Summary of Application*

Section 3.5 of the LNP COL FSAR, Revision 9, incorporates by reference Section 3.5 of the AP1000 DCD, Revision 19. Section 3.5 of the DCD includes Section 3.5.1.

In addition, in LNP COL FSAR Section 3.5, the applicant provided the following:

AP1000 COL Information Item

- LNP COL 3.3-1 and LNP COL 3.5-1

The applicant provided additional information in LNP COL 3.3-1 to resolve COL Information Item 3.3-1 (COL Action Item 3.3.2.2-1) and LNP COL 3.5-1 to resolve COL Information Item 3.5-1 (COL Action Item 3.5.1.5-1). LNP COL 3.3-1 and LNP COL 3.5-1, in LNP COL FSAR Section 3.5.1.5, "Missiles Generated by Events Near the Site," states that the buildings and structures at the LNP site are common structures that are located at a nuclear power plant. They are of similar design and construction to those that are typical at nuclear power plants.

Therefore, any missiles resulting from a tornado-initiated failure are not more energetic than tornado missiles postulated for design of the AP1000. Also, LNP COL 3.5-1 states that the missiles generated by events near the site are evaluated in accordance with FSAR Section 2.2.3, and concludes effects of these events on Units 1 and 2 safety-related components are insignificant.

The applicant provided additional information under LNP COL 3.5-1 related to hurricane missile parameters. This information was provided under Section 3.5.2 of the FSAR and is reviewed under Section 3.5.2 of this report.

Supplemental Information

- STD SUP 3.5-1

The applicant provided supplemental information by adding text to the end of AP1000 DCD Section 3.5.1.3. This supplemental information states that the potential for a turbine missile from another AP1000 plant in close proximity has been considered for LNP Units 1 and 2 in accordance with RG 1.115, "Protection Against Low-Trajectory Turbine Missiles," Revision 1.

- STD SUP 3.5-2

The applicant provided supplemental information by stating that the turbine system maintenance and inspection program is discussed in AP1000 DCD 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 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for missile selection and description are given in Sections 3.5.1.1 through 3.5.1.6 of NUREG-0800.

The regulatory basis for acceptance of LNP COL 3.5-1 is based on the development of site-specific parameters and verification of bounding conditions compared to the DCD interface criteria for missile generation, site arrangement, and building construction. The design of AP1000 safety-related structures for protection against missiles using acceptable procedures must meet the requirements of 10 CFR Part 50, Appendix A, GDC 4, "Environmental and Dynamic Effects Design Bases." 10 CFR 100.21(e), "Non-seismic site criteria," provides regulatory requirements for potential hazards associated with nearby transportation routes, industrial and military facilities.

Additional regulatory guidance related to the review of the issues in this SER section are given in RG 1.91, "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants," Revision 1; and RG 1.115, "Protection Against Low-Trajectory Turbine Missiles."

3.5.1.4 *Technical Evaluation*

The NRC staff reviewed Section 3.5 of the LNP 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 missile protection of safety-related SSCs. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 staff reviewed the information in the LNP COL FSAR:

AP1000 COL Information Items

- LNP COL 3.3-1 and LNP COL 3.5-1

The NRC staff reviewed the COL information item LNP COL 3.5-1 and LNP COL 3.3-1 related to missiles generated by events near the site included under Section 3.5.1.5 of the LNP COL FSAR. 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 NRC staff confirmed that no pressure criteria were exceeded, 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.

COL Information Item LNP COL 3.5-1 in LNP COL FSAR Section 3.5.1.6, "Aircraft Hazards," states that based on the discussion in LNP COL FSAR Section 2.2.2.7, the applicant concludes that the calculated total aircraft crash hazard probability is 7.011×10^{-6} per year, which results in a core damage frequency (CDF) of 4.10×10^{-11} per year. This is not considered a safety concern since the resultant CDF when combined with other risk factors is less than the acceptable CDF for the AP1000 of 1.0×10^{-8} per year. Therefore, the applicant concludes the aircraft hazards pose no undue risk to the health and safety of the public. The staff's review of CDF calculations is presented as part of Probabilistic Risk Assessment (PRA) in Section 19.2 of this SER.

The applicant evaluated potential aircraft hazards 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 whether aircraft accidents resulting in radiological consequences would exceed the 10 CFR Part 100 radiological dose requirements was determined by the applicant based on the following:

Revision 0 of LNP FSAR did not list any flight paths passing within 2 miles of the LNP site. The NRC staff noted two flight paths within 2 miles and issued RAI 2.2.1-2.2.2-2 requesting evaluation of the flight paths. In LNP's response to the RAI, LNP stated that: "The outer boundaries of five airways are routed within two miles of the LNP site: V7-521, VR 1006, J119, Q110-116-118 and Q112 (as shown on LNP FSAR Figure 2.2.1-204)," and revised FSAR Section 3.5.1.6 to include a risk analysis for aircraft crashes.

The applicant calculated the total probability of small aircraft crash into the plant to be on the order of 7.011×10^{-6} per year. This crash probability results in a core damage frequency of 4.10×10^{-11}per year which is less than current plant (AP1000) acceptance criteria of 1.0×10^{-8} per year. The probability of large aircraft crashing on a seismic category 1 structure is calculated as 3.093×10^{-8} per year. This meets the acceptance criteria of 1.0×10^{-7} per year in Subsection 19.58.2.3.1 of the DCD. The NRC staff reviewed and confirmed the acceptability of the applicant's methodology and conclusions using NUREG-0800, Section 3.5.1.6. Therefore, RAI 2.2.1-2.2.2-2 is closed.

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 LNP units within two miles of the noted flight paths 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). This conclusion is based on the staff's independent verification of the applicant's assessment of aircraft hazards at the site that resulted in a probability less than an 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 Section 3.5.1.4 of the VEGP SER:

<u>Supplemental Information</u>

- *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 $1x10^{-5}$ per year, and that the 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.

Therefore, the staff finds that the probability of generating a turbine missile meets the guidance in Section 3.5.1.3 of NUREG-0800 and the requirements of GDC 4, since the probability of a missile striking a safety-related component is acceptably low. As an additional conservative measure, the shield building and auxiliary building walls, roofs, and floors provide some inherent protection of the safety-related components, but are not credited in preventing turbine missile strikes of safety-related components. As a result, Open Item 1-1, as it relates to the probability of a missile striking a safety-related component, is closed for the VEGP application review.

3.5.1.5 *Post Combined License Activities*

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

3.5.1.6 *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 missile protection, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP COL FSAR is acceptable and meets the regulatory guidance in Sections 3.5.1.1 through 3.5.1.6 of NUREG-0800. The staff based its conclusion on the following:

- LNP COL 3.3-1 and LNP COL 3.5-1 are acceptable because they meet the acceptance criteria provided in Sections 3.5.1.5 and 3.5.1.6 of NUREG-0800.

- STD SUP 3.5-1 is acceptable because the turbine missile evaluation for collocated 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 the turbine maintenance and inspection program is included in Section 10.2 of this SER.

3.5.2 Protection from Externally Generated Missiles

3.5.2.1 *Introduction*

Systems required for safe shutdown are protected from the effects of missiles. Protection from external missiles, including those generated by natural phenomena, is provided by the external walls and roof of the seismic Category I nuclear island structures. The structural design requirements for the shield building and auxiliary building are outlined in AP1000 DCD Section 3.8.4. The external walls and roofs are reinforced concrete. Openings through these walls are evaluated on a case-by-case basis to provide confidence 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.

3.5.2.2 *Summary of Application*

Section 3.5.2 of the FSAR, Revision 9, incorporates by reference Section 3.5.2 "Protection from Externally Generated Missiles," of the AP1000 DCD, Revision 19.

In addition, in FSAR Section 3.5.2, the applicant provided the following:

AP1000 COL Information Item

- LNP COL 3.5-1

The applicant provided additional information in FSAR Section 3.5.1.4 and 3.5.2 to resolve LNP COL 3.5-1. FSAR Section 3.5.2 identifies the horizontal and vertical velocities of design-basis missiles generated by site-specific hurricane winds.

Supplemental Information

- LNP SUP 3.5-3

The applicant provided supplemental information by adding Table 3.5-202 to AP1000 DCD Section 3.5. This supplemental information provides a summary of the site-specific hurricane-generated missile parameters and compares them to AP1000 DCD Tier 1 Table 5.0-1 tornado-generated missile parameters.

3.5.2.3 *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 protection from externally generated missiles are given in Sections 3.5.1.4 and 3.5.2 of NUREG-0800.

The regulatory basis for acceptance of LNP COL 3.5-1 is based on the development of site-specific parameters compared to the DCD missile parameters. The design of AP1000 safety-related structures for protection against missiles using acceptable procedures must meet the requirements of 10 CFR Part 50, Appendix A, GDC 2, "Design Bases for Protection Against Natural Phenomena," and GDC 4, "Environmental and Dynamic Effects Design Bases."

Additional regulatory guidance related to the review of the issues in this SER section are given in RG 1.221, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants," Revision 1."

3.5.2.4 *Technical Evaluation*

The NRC staff reviewed Section 3.5 of the 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 missile protection of safety-related SSCs. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

COL Information Items and Supplements

- LNP COL 3.5-1 and LNP SUP 3.5-3

LNP COL 3.5-1 requests COL applicants to evaluate whether the site characteristics for wind and tornadoes satisfy the AP1000 site parameters for wind and tornado conditions. If there are exceedances they must be discussed and shown acceptable. In Section 3.5.2 of the FSAR, the applicant provided additional information to address this COL information item.

The staff reviewed the information contained in the FSAR Section 3.5.2. The review evaluated the applicant's assessment of possible hazards attributed to missiles generated by extreme winds (such as hurricanes and tornados) identified in FSAR section 3.5.

RG 1.221 provides new guidance that the NRC staff considers acceptable for use in selecting the design-basis hurricane wind speed and hurricane-generated missiles that a new nuclear power plant should be designed to withstand to prevent undue risk to public health and safety.

In response to RAI 2.03.01-20, the applicant addressed hurricane-generated missiles in LNP COL 3.5-1 and provided hurricane missile spectra and associated velocities based on RG 1.221, and a discussion on whether the individual missiles are bounded by the AP1000 DCD.

The applicant concludes in LNP COL FSAR Section 3.5.2 that the AP1000 DCD design-basis tornado missile horizontal and vertical velocities of the 8 inch (275 lbs) artillery shell and the 1 inch steel sphere bound similar missiles subject to the site-specific hurricane wind of 195 mph. However, the site-specific hurricane-generated automobile missile results in a horizontal velocity of 120 mph which exceeds the AP1000 DCD automobile tornado missile velocity of 105 mph. As a result, the applicant evaluated the impact of the site-specific hurricane-generated automobile missile on the exterior walls of the nuclear island and concluded the LNP nuclear island is adequately protected against the hurricane-generated automobile missile impact. The staff's evaluation of the wind and missile loading and structural engineering aspects of RAI 2.3.1-20 is in Section 3.3.2.4 of this SER.

In addition, the applicant provided LNP SUP 3.5-3, Table 3.5-202, which compares the site-specific hurricane generated missile spectra and associated velocities to AP1000 DCD Tier 1 Table 5.0-1 tornado-generated missile parameters.

The staff reviewed the additional and supplemental information provided by the applicant and verified that the methodologies used to calculate the site-specific hurricane missile spectra and associated velocities are consistent with Figure 2, Table 1, and Table 2 of RG 1.221. On the basis of its review, the staff concludes that the information in FSAR Section 3.5.2 associated with LNP COL 3.5-1 and LNP SUP 3.5-3 adequately addresses COL information item 3.5-1 and is acceptable because the site-specific hurricane missile parameters conform to the guidance of RG 1.221.

3.5.2.5 *Post Combined License Activities*

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

3.5.2.6 *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 missile protection, and there is no outstanding information expected to be addressed in the FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the FSAR is acceptable and meets the requirements of GDC 2 and GDC 4 with respect to missiles and environmental effects. The staff based its conclusion on the following:

- LNP COL 3.5-1 and STD SUP 3.5-3 are acceptable because they meet the acceptance criteria provided in Sections 3.5.1.4 and 3.5.2 of NUREG-0800, and conform to RG 1.221.

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.

Section 3.5 of the LNP COL FSAR, Revision 9, incorporates by reference Section 3.5.3, "Barrier Design Procedures," of the AP1000 DCD, Revision 19 without any departures or supplements. 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 LNP COL application are documented in NUREG-1793 and its supplements.

3.6 <u>Protection against Dynamic Effects Associated with the Postulated Rupture of Piping</u>

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

Section 3.6 of the LNP COL FSAR, Revision 9, incorporates by reference Section 3.6 of the AP1000 DCD, Revision 19. Section 3.6 of the DCD includes Section 3.6.4.

In addition, in LNP COL FSAR Section 3.6.4, the applicant provided the following:

AP1000 COL Information Items

- LNP COL 3.6-1

The applicant provided additional information in LNP COL 3.6-1 replacing 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. This evaluation will be based on a completed piping layout and will be completed to support the COL. The evaluations will be provided prior to fabrication and installation of the piping and connected parts. In a letter dated July 22, 2011, the applicant committed to remove this additional information because the standard content provided in Revision 2 of the FSAR provides all the necessary information for resolving STD COL 3.6-1.

- STD COL 3.6-1

The applicant provided additional information in STD COL 3.6-1 to address COL Information Item 3.6-1. Specifically, the applicant stated 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. The applicant further stated that the final design of these activities will be completed prior to fabrication and installation of the piping and connected components.

- STD COL 3.6-4

The applicant provided additional information in STD COL 3.6-4 to address COL Information Item 3.6-4, regarding LBB inspections.

License Condition

- Part 10, License Condition 2, Item 3.6-1

The applicant has proposed a license condition addressing the as-designed pipe rupture hazards analysis completion schedule.

Inspections, Tests, Analyses and Acceptance Criteria (ITAAC)

In its letter dated September 23, 2010, the applicant endorsed the letter dated April 23, 2010, from the VEGP applicant, that proposed ITAAC requiring the completion of an as-designed pipe rupture hazards analysis to demonstrate that SSCs required to be functional during and following a postulated pipe failure are protected against or qualified to withstand the dynamic and environmental effects resulting from postulated pipe failures.

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 the Commission regulations (GDC 4 of Appendix A to 10 CFR Part 50) for the piping design against pipe breaks, pipe break locations and characteristics in safety-related piping, and LBB evaluation procedures are given in Sections 3.6.1, 3.6.2, and 3.6.3 of NUREG-0800.

3.6.4 Technical Evaluation

The NRC staff reviewed Section 3.6 of the LNP 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 piping design against pipe break, pipe break locations and characteristics in safety-related piping, and LBB evaluation procedures. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 one confirmatory item in the standard content material retains the number assigned in the VEGP SER.

AP1000 COL Information Items

The following portion of this technical evaluation section is reproduced from Section 3.6.4 of the VEGP SER:

- *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.*

Resolution of Standard Content Open Item 3.6-1

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 14.3.3.# [where # is the next sequential number] related 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****.*

Resolution of Standard Content Confirmatory Item 3.6-1

Confirmatory Item 3.6-1 is an applicant commitment to revise its FSAR Section 3.6.4.1 and, Section 14.3.3.2, [Section 14.3.3.4 for LNP] *to verify the incorporation of the as-designed pipe rupture hazard analysis and add an ITAAC (Table 3.8-1)* [Table 3.8-5 for LNP] *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)* [Table 3.8-5 for LNP] *were appropriately updated. As a result, Confirmatory Item 3.6-1 is now closed.*

- LNP COL 3.6-1

The NRC staff reviewed LNP COL 3.6-1 included under Section 3.6.4.1 of the LNP COL FSAR. 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. This evaluation will be based on a completed piping layout and will be completed to support the COL. The evaluations will be provided prior to fabrication and installation of the piping and connected parts. In a letter dated July 22, 2011, the applicant committed to remove this additional information because the standard content provided in Revision 2 of the FSAR 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 LNP 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 condition, 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 ITAAC in SER Table 3.6-1.

- 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 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 pipe design against pipe break, pipe break locations and characteristics in safety-related piping, and LBB evaluation procedures and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the relevant information presented in the LNP COL FSAR is acceptable and meets the requirements of GDC 4 of Appendix A to 10 CFR Part 50. The staff based its conclusion on the following:

- STD COL 3.6-1 is acceptable because the applicant's proposed resolution to COL Information Item 3.6-1 in LNP 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 with these guidelines provides an acceptable basis for satisfying, in part, the requirements of GDC 4 of Appendix A to 10 CFR Part 50.

- STD COL 3.6-4 is acceptable because the applicant's proposed resolution to COL Information Item 3.6-4 in Section 3.6.4.4 of the LNP COL FSAR 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 with these guidelines provides an acceptable basis for satisfying, in part, the requirements of GDC 4 of Appendix A to 10 CFR Part 50.

3.7 Seismic Design

Seismic design of the AP1000 seismic Categories I and II structures, systems, equipment, and components are based on the safe shutdown earthquake (SSE). The operating basis earthquake (OBE) has been eliminated as a design requirement for the AP1000. 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. For the purposes of the shutdown criteria OBE for shutdown is considered to be one-third of the SSE.

Seismic Category I structures, system, and components (SSCs) are designed to withstand the effects of the SSE event and to maintain the specified design functions. Seismic Category II and non-seismic (NS) structures are designed or physically arranged (or both) so that the SSE could not cause unacceptable structural interaction with or failure of seismic Category I SSCs.

3.7.1 Seismic Design Parameters

3.7.1.1 *Introduction*

The input seismic design ground motion response spectra (GMRS) for the SSE in the free field at plant grade is addressed. The horizontal and vertical design GMRS for the AP1000 were developed based on the response spectra in Revision 1 of RG 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," with consideration of high-frequency amplification effects.

The bases for the seismic design of safety-related SSCs and equipment include the following:

- Design GMRS
- Design ground motion time histories
- Percentage of critical damping values
- Supporting media for seismic Category I structures
- COL action items

3.7.1.2 *Summary of Application*

Section 3.7 of the LNP COL FSAR, Revision 9, incorporates by reference Section 3.7, of the AP1000 DCD, Revision 19. Section 3.7 of the DCD includes Section 3.7.1.

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," and evaluate potential seismic hazards at the LNP site in light of these recommendations, the applicant performed sensitivity studies using the Central and Eastern United States Seismic Source Characterization model (NUREG-2115) to evaluate potential seismic hazards (e.g., changes to the GMRS) at the LNP site. The sensitivity studies indicated that the LNP site-specific soil-structure interaction analysis results remain conservatively bounded by the standard plant analysis results, and the staff conclusions regarding the adequacy of the site-specific soil-structure interaction analysis results remain unchanged. SER Section 20.1 presents the staff's evaluation of the sensitivity studies.

In addition, in LNP COL FSAR Section 3.7, the applicant provided the following:

<u>*Supplemental Information*</u>

- LNP SUP 3.7-3

The applicant provided supplemental information in LNP SUP 3.7-3 by adding Section 3.7.1.1.1 to the LNP COL FSAR, which addresses site-specific GMRS. In LNP SUP 3.7-3, the applicant states that the horizontal and vertical site-specific GMRS were developed as the Truncated Soil Column Surface Response (TSCSR) on the uppermost in-situ competent material at elevation 11 m (36 ft) and were compared to the AP1000 certified seismic design response spectra (CSDRS). Additionally, the applicant has developed performance based surface horizontal and vertical response spectra (PBSRS) at the design grade elevation of 15.5 m (51 ft). These PBSRS incorporate a scaling of the motion that is sufficient to achieve at least 0.1g horizontal peak ground acceleration at the foundation level of the NI. The PBSRS are also compared to the AP1000 CSDRS and shows that the CSDRS envelopes the scaled PBSRS.

In addition to the PBSRS, the applicant provided finished grade soil-structure interaction (SSI) analysis input surface spectra. These spectra were developed from the three soil columns (best estimate (BE), lower bound (LB), and the upper bound (UB) properties) using the soil column outcrop response spectra (SCOR) FIRS developed for elevation -7.3 m (-24 ft.), corresponding to the base elevation of planned excavation beneath the NI. Both horizontal and vertical SSI input response spectra were developed. The applicant states that the SSI input spectra from the UB, BE, and LB soil columns (Figures 3.7-202, 203, and 204) along with the corresponding acceleration time histories and corresponding UB, BE, and LB soil column profiles (Tables 2.5.2-228, 229, and 230) would be used for NI SSI analysis, if required. The envelope of the SSI input spectra from the UB, LB, and BE envelopes the PBSRS as required by DC/COL-ISG-017.

A comparison of the AP1000 CSDRS with the SSI input response spectra from the UB, BE, and LB soil columns for the horizontal ground motions for the North-South (H1) and the East-West (H2) directions is presented in Figures 3.7-202 and 203. The applicant states that, since the CSDRS envelops the SSI input response spectra from the three soil columns, site specific SSI analysis for horizontal ground motions is not required. The applicant states that for vertical ground motions in Figure 3.7-204, the CSDRS does not envelop the finished grade surface SSI input response spectra from the three soil columns in the high frequency range (greater than approximately 30 Hz). However, the applicant states that the CSDRS-based vertical in-structure response spectra envelopes the corresponding site-specific FIRS-based vertical in-structure response spectra.

The applicant also provided additional information in LNP SUP 3.7-3 by adding Section 3.7.1.1.2 to the LNP COL FSAR, which addresses foundation input response spectra (FIRS). The NI is supported on 10.7 meters (35 ft) of RCC over rock formations at the site as described in LNP COL FSAR Section 2.5.4.5. The seismic Category II Annex Building, Turbine Building, and Radwaste Building are supported on drilled shafts. The applicant compares foundation input response spectra (FIRS) for the NI at the base of the planned excavation beneath the NI and at the AP1000 foundation elevation in Figures 3.7-201 and 3.7-205, respectively. The applicant further states that the PBSRS are used to compute the maximum relative displacements of the drilled shaft foundations for the Annex Building, Turbine Building, and Radwaste Building with respect to the NI. These displacements are used to evaluate site-specific aspects of the seismic interaction of these buildings with the NI.

3.7.1.3 *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 (GDC 2 of Appendix A to 10 CFR Part 50; Appendix S, “Earthquake Engineering Criteria for Nuclear Power Plants” to 10 CFR Part 50; and 10 CFR 100.23, “Geologic and seismic siting criteria") for the seismic design parameters are given in Section 3.7.1 of NUREG-0800.

3.7.1.4 *Technical Evaluation*

The NRC staff reviewed Section 3.7 of the LNP 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 seismic design parameters. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the LNP COL FSAR:

Supplemental Information

- LNP SUP 3.7-3

LNP SUP 3.7-3 provides additional information on the design GMRS at LNP Units 1 and 2 to address COL Information Item 3.7-3. The NRC staff reviewed the resolution of the seismic input included under Section 3.7.1.1.1 of the LNP COL FSAR. The applicant compares the horizontal and vertical GMRS that were developed at the uppermost in-situ competent material at elevation 11 m (36 ft) to the CSDRS. Additionally, the applicant has developed performance based surface horizontal and vertical response spectra (PBSRS) at the design grade elevation 15.5 m (51 ft). These PBSRS incorporate a scaling of the motion that is sufficient to achieve at least 0.1 g horizontal peak ground acceleration at the foundation level of the NI. The PBSRS are also compared to the CSDRS and the applicant shows that the CSDRS envelops the scaled PBSRS.

In its review of the site-specific supplemental information in LNP COL FSAR Section 3.7.1.1.1 and Section 3.7.1.1.2, the staff indicated in RAI 03.07.01-1 that, since the FIRS and CSDRS are defined at different elevations, comparison of these spectra is inappropriate. The comparison that is needed is the AP1000 CSDRS to envelope the spectra generated at the ground surface for each of the three SSI profiles (BE, UB, and LB site profiles).

As part of its review, the NRC staff performed the following confirmatory analyses to evaluate the adequacy of the applicant's PBSRS and Soil Column Outcrop Response spectra (SCOR) FIRS.

NRC PBSRS Site Response Confirmatory Analysis

The applicant developed site amplification functions for the calculation of the PBSRS and associated FIRS following Subsection 5.2.1 of the Interim Staff Guidance DC/COL-ISG-017. The process used by the applicant to develop the PBSRS and FIRS similarly follows the process for the GMRS as described in Section 2.5.2 of this SER. The primary difference is that the PBSRS is developed for the plant finished grade and includes the effects of engineered fill, while the GMRS in Section 2.5.2 represents the ground motions at the overall LNP Units 1 and 2 sites and does not characterize building- or fill-specific ground motions.

To determine the adequacy of the applicant's PBSRS site response calculations at the elevation of 15.5 m (51 ft) NAVD88, the staff performed confirmatory site response analysis. As input, the staff used the static and dynamic soil properties provided in FSAR Table 2.5.2-222 for LNP Unit 1 and FSAR Table 2.5.2-223 for LNP Unit 2 sites. With hard rock located at the depth of 1,325 m (4,350 ft), the overlying static and dynamic soil property profiles consist of 29 layers that reach the elevation of +11 m (+36 ft) NAVD88 with an additional layer of 4.5 m (15 ft) of engineered fill. The average fill V_S is 259 m/s (850 ft/s) with a variation from 154 to 308 m/s (500 to 1000 ft/s). To model the nonlinear properties of the engineered fill, the staff used shear modulus reduction and damping relationships for the engineered fill material developed by Menq (2003).

The staff performed the site response calculations using the program STRATA (Kottke and Rathje, 2008). The staff calculated six site amplification functions, one for each of the three fill layer's V_S (154, 259, and 308 m/s (500, 850 and 1000 ft/s)) for both LNP1 and LNP2 sites. In each calculation of the site amplification function the staff used 60 randomized V_S profiles. Secondly, the staff used the following averaging scheme proposed by the applicant to calculate weighted average site amplification functions separately for the LNP Unit 1 and Unit 2 sites: 154 m/s (0.185), 259 m/s (0.63), and 308 m/s (0.185). Third, the staff took the maximum of the LNP1 and LNP2 site amplifications by enveloping the two weighted site amplification functions, and multiplied the uniform hazard response spectrum (UHRS) by the envelope function to calculate the PBSRS.

The staff's calculated PBSRS is enveloped by the applicant's. Therefore, the staff concludes that in the frequency range significant to a reactor's structures, systems, and components, there are no significant differences between the staff's and the applicant's calculated amplification functions and the PBSRS for the 10^{-4} and 10^{-5} hazard levels.

The two soil column outcropping response (SCOR) FIRS were developed by the applicant for the purpose of checking the requirement of the minimum level of ground motion specified in 10 CFR Part 50, Appendix S. The applicant developed the first SCOR FIRS for the elevation -7 m (-24 ft) NAVD88 since the site is to be excavated to this elevation. In addition, a second SCOR FIRS was developed at the reactor foundation elevation of +3.3 m (+11 ft) NAVD88 corresponding to the reactor placed on approximately 10.7 m (35 ft) of concrete backfill.

The staff performed confirmatory calculations for both FIRS using the same approach as described above for the PBSRS. For the V_S profiles for the FIRS at the elevation of -7 m (-24 ft), the staff used profiles shown in FSAR Tables 2.5.2-222 and 2.5.2-223. Modifications to these V_S profiles were made by removing the layers in the upper 15 m (50 ft). Figure 3.7.1-1 compares the staff's calculation of the FIRS with the CSDRS and FSAR (Fig.3.7-201).

The set of site amplification functions for the FIRS at the elevation of +3.3 m (+11 ft) was calculated using 10.7 m (35 ft) of concrete with V_S of 1066 m/s (3500 ft/s) (FSAR Table 2.5.4.5-201) on the top of profile used for the FIRS at the elevation of -7 m (-24 ft). Figure 3.7.1-1 compares the staff's calculation of FIRS with CSDRS and the applicant's from the LNP Unit 1 and 2 FSAR (Figure 3.7-205).

The staff's calculation of the FIRS at both elevations of -7 m (-24 ft) and +3.3 m (+11 ft) are enveloped by the applicant's FIRS and by the CSRDS. Therefore, the staff concludes that in the frequency range significant to a reactor's structures, systems, and components, there are no significant differences between the staff's and the applicant's calculated amplification functions and FIRS for the 10^{-4} and 10^{-5} hazard levels.

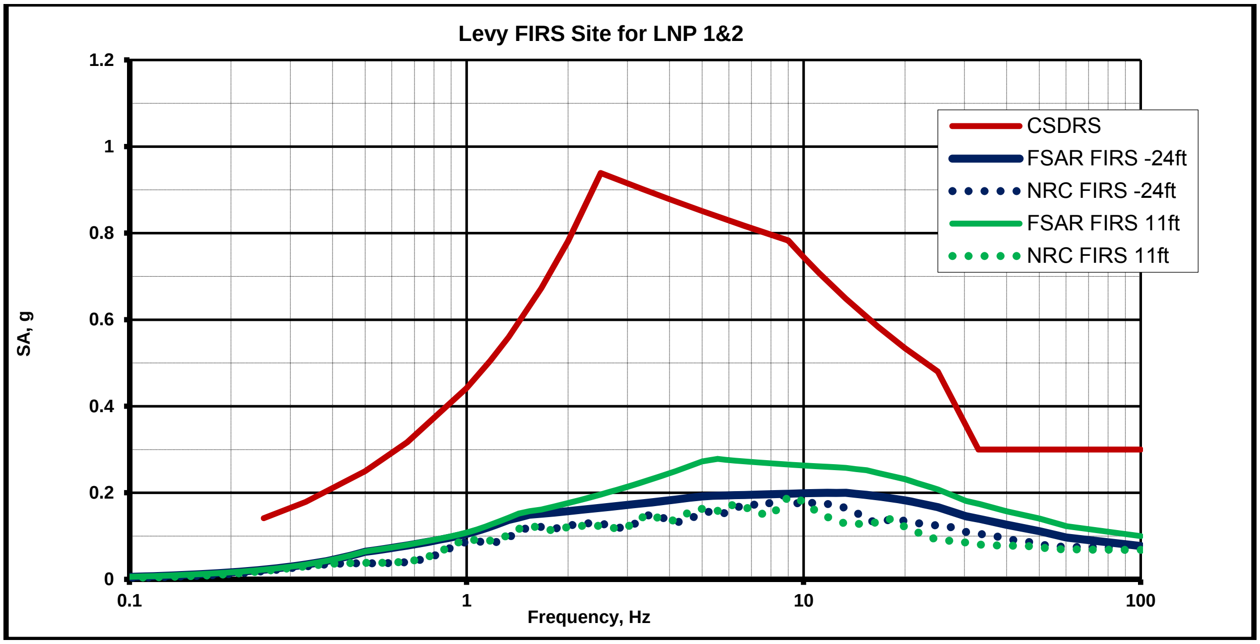


Figure 3.7.1–1. Comparison of the horizontal FIRS at the elevations of -7 m (-24 ft) and +3.3 m (+11 ft) calculated by the NRC staff, the horizontal FIRS at the same elevations calculated by the applicant, and the Westinghouse CSDRS.

In addition, the staff identified that discreet shear wave velocities as measured by the P-S logger (Figures 2.5.2-247 and -248) are quite variable, ranging by wide margins from the average value within the zone of influence of the foundation. The staff questioned whether the variability of shear wave velocities across the footprint of the NI was consistent with the assumptions in the AP1000 DCD SSI calculations and whether the impact of the variability on SSI response had been adequately addressed.

The staff also questioned whether the design of the NI structures, which are based on the intrinsic assumption that the properties of the soils to the side of the structure are the same as under the foundation, and whether the seismic gap between the NI and adjacent facilities is adequate to account for the relative displacements anticipated for the plant. In RAI 03.07.01-1, the staff requested that the applicant provide justification for the assumed uniformity of site soil layers structure for SSI response evaluations.

In RAI 03.07.01-1, the staff requested that the applicant justify the design of the drilled shafts supporting the structures adjacent to the NI to assure that the behavior of the adjacent structures under seismic loading had been adequately addressed and that the potential impact of loss of support of side soils on the drilled shaft design be included in the justification.

The staff generated RAI 03.07.01-2 based on the applicant's response to RAI 03.07.01-1, which requested the applicant to provide information regarding the extent of the planned excavation and the placement of engineered backfill as well an assessment of whether these changes are sufficiently extensive such that the surface ground motion would be modified. As part of this additional RAI, the staff requested the applicant to summarize the planned construction sequence of removal of near surface soils, placement of engineered fill, drilled shaft installation for adjacent structures, construction of the diaphragm wall, and excavation of soil material beneath the NI structures as it relates to potentially changing the ground motion as is inferred by incorporating the engineered fill in the SSI soil columns.

The applicant provided responses to the staff's information requests in their response to RAI 03.07.01-2, dated July 19, 2010. The applicant stated that the backfill to design grade was included in the free field response analysis as specified in DC/COL-ISG-017 and in the SSI analysis since the plant NI footprint (approximately 0.8 acres for each unit) is small compared to the approximately 347 acres where fill will be placed to raise the existing grade level. In addition the PBSRS is higher than the GMRS for the LNP site.

The applicant also provided a description of the basis for the extent of placement of controlled engineered fill. The applicant states that the backfill provides lateral support to drilled shafts that support the Turbine Building, Annex Building , and Radwaste Building. The applicant further stated that seismic II/I interaction evaluations show that for drilled shafts up to 6 ft. in diameter, the lateral stiffness of the drilled shafts is primarily dependent on the soil property of the top 4.9 m (16 ft) of soil. The ~9.1 m (~30 ft) lateral extent of the controlled engineered fill corresponds to the lateral extent of an assumed passive wedge extending through the engineered fill having a friction angle of 34 degrees as specified in Table 2.5.4.5-201.

The applicant developed an SSI model to calculate the ISRS for the Levy site-specific soil profile and foundation geometry. The SSI model incorporated the effects of the roller compacted concrete bridging mat beneath the NI and in-situ and disturbed soil properties. For the SSI analysis of the NI, the BE, LB, and UB soil profiles were considered and the potential degradation of soil due to foundation installation was considered by adding an additional Lower LB case (LLB). The applicant demonstrated that the ISRS considering the Levy site-specific soil profile, foundation geometry, and site specific ground motion is enveloped by the AP1000 DCD ISRS.

In RAI 03.08.05-3, the staff requested additional information related to the drilled shaft design. In a response dated June 8, 2010, the applicant provided a description of the drilled shaft conceptual design, key installation practices, and industry codes that will be specified for installation. The applicant stated that the lateral stiffness of the drilled shafts is primarily governed by soil properties in the top 10 ft for the drilled shaft up to 4 ft diameter and 16 ft for 6 ft diameter drilled shafts. A description of the civil construction sequence construction practices was also provided. Also, the applicant provided proposed ITAAC in FSAR Table 3.8-4 to ensure that the as built design of the drilled shafts provide adequate bearing capacity to safely sustain the vertical design load of the drilled shafts and required lateral stiffness of the drilled shaft to minimize seismic interactions with adjacent structures to NI. In a letter dated January

25, 2011, the applicant revised portions of its June 8, 2010 response letter to address the design of the drill shaft. The staff reviewed the applicant's revised response and found it to be acceptable. A full discussion of the contents of RAI 03.08.05-3 can be found in Section 3.8.5.4 of this SER.

The staff reviewed the applicant's responses to RAIs 03.07.01-1, 03.07.01-2, and 03.08.05-3, including the calculations and reports attached to the responses and concluded that the supplemental information provided is adequate to demonstrate that the Levy site specific demand is enveloped by the AP1000 CSDRS. The staff considers RAIs 03.07.01-1, 03.07.01-2, and 03.08.05-3 to be resolved pending the incorporation of changes in a future revision to the LNP COL FSAR. This is being tracked as **Confirmatory Item 3.7-1.**

Resolution of Confirmatory Item 3.7-1

Confirmatory Item 3.7-1 is an applicant commitment to revise the LNP COL in several locations as it pertains to RAIs 03.07.01-1, 03.07.01-2, and 03.08.05-3. The staff verified that the changes proposed in the responses dated July 19, 2010, June 8, 2010, and January 25, 2011, were incorporated into Revision 4 of the LNP COL FSAR. As a result, Confirmatory Item 3.7-1 is now closed.

3.7.1.5 *Post Combined License Activities*

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

3.7.1.6 *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 seismic design parameters, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP COL FSAR is acceptable, and meets the requirements of 10 CFR Part 50, Appendix A, Appendix S, and other staff guidance. The staff based its conclusion on the following:

> LNP SUP3.7-3 is acceptable because the applicant addressed the relevant information that meets the guidance in Section 3.7.1 of NUREG-0800. In conclusion, the applicant has provided sufficient information for satisfying 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*

Seismic analysis methods and acceptance criteria for all seismic Category I SSCs are described. The review includes a review of 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 SSI on the seismic responses of the NI structures are included in the review scope because the LNP site is considered a soil site (e.g., shear wave velocity is greater than 1000 ft/s at foundation level). The 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).

Specifically, the criteria and methods for the seismic analysis of safety-related SSCs and equipment include the following:

- Seismic analysis methods
- Natural frequencies and response loads
- Procedures used for analytical modeling
- SSI
- Development of FRS
- Three components of earthquake motion
- Combination of modal responses
- Interaction of NS Category II structures with seismic Category I SSCs
- Effects of parameter variations on FRS
- Use of constant vertical static factors
- Method used to account for torsional effects
- Methods for seismic analysis of dams
- Determination of seismic Category I structures overturning moments
- Analysis procedure for damping

3.7.2.2 *Summary of Application*

Section 3.7 of the LNP COL FSAR, Revision 9, incorporates by reference Section 3.7 of the AP1000 DCD, Revision 19. Section 3.7 of the DCD includes Section 3.7.2. In addition, in LNP COL FSAR Section 3.7.2, the applicant provided the following:

Supplemental Information

- LNP SUP 3.7-5

The applicant added supplemental information to the end of AP1000 DCD Sections 3.7.2.8.1, 3.7.2.8.2, and 3.7.2.8.3 regarding the Annex Building, the Radwaste Building, and the Turbine Building, respectively.

Departure

- LNP DEP 3.7-1

The applicant submitted a letter dated August 27, 2013, proposing a departure from the AP1000 certified design (DCD Revision 19) related to the maximum seismic loads for the drilled shaft foundations underlying the Annex Building and Turbine Building.

AP1000 COL Information Items

- LNP COL 3.7-1

The applicant provided additional information in LNP COL 3.7-1 regarding seismic analysis of dams near the site, to address COL Action Item 3.7.2.13-1 identified in NUREG-1793, Appendix F, and COL Information Item 3.7-1 discussed in Section 3.7.5.1 of the AP1000 DCD.

- STD COL 3.7-3

The applicant provided additional information in STD COL 3.7-3 to address COL Action Item 3.7.5-3 identified in NUREG-1793, Appendix F, and COL Information Item 3.7-3 discussed in Section 3.7.5.3 of the AP1000 DCD. Since the information added by STD COL 3.7-3 is the subject of a proposed license condition (Part 10, License Condition 2, Item 3.7-3, see below), this COL item will not be discussed further in this SER.

- STD COL 3.7-4

The applicant provided additional information in STD COL 3.7-4 to address COL Action Item 3.7.5-1 identified in NUREG-1793, Appendix F, and COL Information Item 3.7-4 discussed in Section 3.7.5.4 of the AP1000 DCD. Since the information added by STD COL 3.7-3 is the subject of a proposed license condition (Part 10, License Condition 2, Item 3.7-4, see below), this COL item will not be discussed further in this SER.

License Conditions

- Part 10, License Condition 2, Item 3.7-3

The applicant has proposed a license condition requiring a seismic interaction review for as-built information. This review is performed in parallel with the seismic margin evaluation and will follow the methodology in Section 3.7.5.3 of the AP1000 DCD. 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.

- 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.

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 the Commission regulations for the seismic system analysis are given in Section 3.7.2 of NUREG-0800.

3.7.2.4 *Technical Evaluation*

The NRC staff reviewed Section 3.7 of the LNP 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 seismic system analysis. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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 LNP Unit 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting.

The staff reviewed the information in the LNP COL FSAR and noted that the AP1000 DCD (Revision 19) Section 2.5.2.3 addresses site-specific seismic evaluation that should be performed by the Combined License applicant if the site-specific design response spectra exceed the CSDRS or if site soil conditions are outside the range evaluated for AP1000 design certification. According to the applicant's response to RAI Question 03.07.01-1, dated November 16, 2010, the site-specific surface design response spectra exceed the CSDRS in vertical motion at the LNP site. Although the applicant views that the CSDRS-based in-structure response spectra would envelop the corresponding site-specific FIRS-based in-structure response spectra, no quantitative evaluation was provided to justify the assumption. As for site soil conditions, no subsurface profile considered in the AP1000 DCD is similar to that of the LNP site which is characterized by stiff material (3500 psi minimum) immediately under the NI basemat with soft material to the sides (controlled low strength material). In addition, the design and analysis of AP1000 is based on subsurface conditions with uniform properties within horizontal layers, and the applicant response to RAI 03.07.01-1 does not fully justify this assumption of lateral uniformity of subsurface conditions.

In RAI 03.07.02-02, the staff requested that the applicant provide detailed site-specific seismic evaluation of the NI structures and the surrounding structures that may impact the safety function of NI structures. The staff requested that the evaluation fully incorporate the effects of soil-structure interaction and meet the Acceptance Criteria 4 of SRP Section 3.7.2.

In its supplement response to RAI 03.07.02-02, dated May 27, 2011, the applicant addressed the seismic evaluation of the NI and the surrounding structures by performing site-specific SSI analyses of the NI. The results of these analyses are also documented in applicant's response. The staff reviewed the LNP specific SSI analyses that utilize both three-dimensional (3D) and two-dimensional (2D) models and the SASSI Subtraction and Direct methods for computing in-structure floor response spectra. The 3D design basis model consists of a 5-Layer, 75-foot embedded Finite Element Model (FEM) developed for the BE soil case using the SASSI Direct method of analysis. This FEM was used to demonstrate that for the LNP site-specific SSI analysis, the use of the SASSI Subtraction Method results in computed responses that are comparable with the responses computed using the SASSI Direct Method.

The evaluation of the potential error that may be associated with the use of the Subtraction Method is made by comparing response spectra at six key NI locations (Figures A-1 through A-18 of the supplement response to RAI 03.07.02-02, dated May 27, 2011). The staff has reviewed the response spectra contained in these figures to assess the potential that the Subtraction Method used for the site-specific configuration predicts responses that are lower than those predicted by the Direct Method. Where the demand computed using the Subtraction Method is lower than that predicted by the Direct Method, the staff has assessed the significance of the non-conservatism as it relates to the comparison to the certified design. A summary of the review is as follows.

At most frequencies the Subtraction Method yields responses that are the same as the Direct Method. At some locations, over limited frequency ranges, the Subtraction Method results exceed the Direct Method results and are therefore conservative. The frequencies where the Subtraction Method results exceed the results for the Direct Method are at higher frequencies (greater than 20 Hz). For both methods in this higher frequency region, adequate design margin exists relative to the standard plant.

In a few locations, over a limited frequency range(less than 15 Hz), the site-specific response spectra computed using the Subtraction Method are lower than the spectra computed using the Direct Method. The frequency range at which this occurs is in the spectral frequency regions where the site-specific spectra are very much lower than the certified design spectra, and an adequate margin exists relative to the seismic demands to which the standard plant design is certified. In addition, these under-predictions generally fall in regions where broadening of the spectra developed using the Subtraction Method would minimize the potential under-prediction of in-structure response.

The staff also reviewed the results that were computed for the 5-Layer model that was refined after the March 2011 audit. The refinements, described in LNG-1000-S2R-804, Revision 4, Section 4.1, resulted in improving a number of the approximations in the analyses model, specifically increasing the number of frequencies of analyses. The effect of improving the model was to significantly reduce the computed peaks in the high frequency regions of the response spectra as seen by comparing the results shown in Figures 6.2-1 through 6.2-18 to those shown in Figures A-1 through A-18. Therefore, the staff concludes that the comparisons used to evaluate the effect of the Subtraction Method on the computed in-structure response described above and shown in Figures A-1 through A-18 under-state the available margin between the site-specific seismic demand and the seismic demand used in the certified design. The lower response computed using the refined model demonstrates that the actual demand will be lower than that used in the comparisons described in the previous paragraphs.

In a letter dated October 4, 2011 the applicant informed the NRC that an error in the calculations presented in the May 27, 2011 letter had been identified. The applicant corrected the error and resubmitted the results of the calculations to the staff. The staff found the corrected information to be acceptable.

An 8-Layer, 75-foot embedded 3D FEM was developed for sensitivity analysis of the LNP BE, UB, LB and LLB site soil cases utilizing the SASSI Subtraction Method. The results of analyses using the 8-layer model demonstrate that the BE case is the controlling case. Two 2D models, which use the Direct Method, were developed to address mesh size modeling, potential frequency filtering due to the model layering and to evaluate the lower boundary SASSI SITE profile depth. The results of the 2D SSI analyses determine the frequency-dependent ratio of Fine-to-Coarse response spectra (≥1.0), (i.e., Bump Factor), which was subsequently applied to the controlling 3D BE Design-Basis FRS for comparison to the AP1000 generic and HRHF FRS envelopes. The applicant demonstrated that the LNP FRS are enveloped by the AP1000 generic FRS at all of the six NI key nodes with sufficient margin.

In RAI 03.08.05-7, the staff requested additional information related to the drilled shafts for structures adjacent to the NI. In its response dated January 25, 2011, the applicant provided supplemental seismic analyses of the drilled shaft supported adjacent structures (Turbine, Annex, and Radwaste Buildings). The analyses results shows that the maximum relative displacement at foundation mat between NI and the adjacent structures for PBSRS was 1.96 cm (0.77 in) and for 10^{-5} UHRS 1.14 cm (0.45 in). These relative displacements are less than the 50 mm (2.0 in) gap per AP1000 DCD. A discussion of the full contents of RAI 03.08.05-7 can be found in Section 3.8.5.4 of this SER.

The staff concluded that the 5 Layer model analysis using the SASSI Direct Method is the design bases analysis based on the review described above and that the use of the Subtraction Method was adequate to determine the governing soil condition and the impact of the model refinement on the predicted responses by the Direct Method. The information provided by the applicant is sufficient to demonstrate that the ISRS are enveloped by the AP1000 ISRS and that the seismic gaps are adequate to prevent interaction between the NI and the adjacent structures.

Because the applicant has provided the details requested in response to RAI 03.07.02-2 the staff considers the RAI to be resolved, pending the incorporation of changes in a future revision to the LNP COL FSAR. This is being tracked as **Confirmatory Item 3.7-2.**

Resolution of Confirmatory Item 3.7-2

Confirmatory Item 3.7-2 was an applicant commitment to revise the LNP COL FSAR. The staff verified that the proposed changes to Chapters 2 and 3 of the LNP COL FSAR, and Revision 5 of the SSI Report, LNG-100-S2R-804, (ML113130557) were made as described in the applicant's letters dated January 25, May 27, and October 4, 2011. As a result Confirmatory Item 3.7-2 is now closed.

The staff reviewed the information in the LNP COL FSAR:

Supplemental Information

- LNP SUP 3.7-5

The NRC staff reviewed LNP SUP 3.7-5 related to adding information to the end of AP1000 DCD Sections 3.7.2.8.1, 3.7.2.8.2, and 3.7.2.8.3 regarding the Annex Building, Radwaste Building, and Turbine Building, respectively.

The staff noted that LNP FSAR Figure 2.5.4.5-201 B indicates that a cementitious fill will be placed adjacent to the NI structures and fills the region between the NI structures and the diaphragm wall. FSAR Section 3.7.2.8 indicates that structure-to-structure interaction will not occur since the gap between the NI and adjacent structures is larger than the expected movement based on the maximum displacement seen in the GMRS. Since the construction details provided in Figure 2.5.4.5- 201 B indicate that the adjacent buildings rest on the diaphragm wall it appeared that there was no gap between the diaphragm wall and NI, and thus the construction detail does not provide a gap as required by the AP1000 DCD. In RAI 03-07-02-1, the staff requested the applicant to clarify the detail to either demonstrate that the required seismic gap would be achieved or that the connectivity between the NI and the adjacent buildings had been properly considered.

Additionally, the staff noted that the GMRS is a ground motion that was developed based on a UHRS motion modified by a scale factor to account for the fragility inherent in the structural system. However, the level of relative displacement that is expected to occur at the ground surface is the displacement that is associated with the UHRS at the performance goal level without the scale factor included. The staff requested that the applicant provide the basis for the use of the GMRS associated displacement in lieu of that associated with the performance goal level UHRS.

The applicant responded to RAI 03.07.02-01 with letters dated July 23, 2010 and November 10, 2010. In a letter dated January 25, 2011, the applicant responded to RAI 03.08.05-7. In those three response letters, the applicant provided information related to seismic gap and the relative displacements between the NI and adjacent structures. This information included details about the drilled shaft to drilled shaft interaction effects, the soil column displacement, the maximum NI displacements at design grade elevation, the probable maximum relative displacements between the NI and the adjacent structures, and the median relative displacements between the NI and the adjacent structures. The applicant also provided figures showing the conceptual design for the interface between the NI and the drilled shaft foundation. The staff reviewed the responses provided by the applicant and concluded that the responses are sufficient to demonstrate that the seismic separation between buildings is adequate to prevent interaction with the seismic Category I NI structures as stated in this SER Section 3.7.2.4. The staff considers RAIs 03.07.02-1, and 03.08.05-7 to be resolved pending the incorporation of changes in a future revision to the LNP COL FSAR. This is being tracked as **Confirmatory Item 3.7-3.**

Resolution of Confirmatory Item 3.7-3

Confirmatory Item 3.7-3 is an applicant commitment to update the LNP COL FSAR in various sections of Chapter 2 and Chapter 3 as discussed in the responses cited above. The staff verified the proposed changes were made to the LNP COL FSAR. As a result, Confirmatory Item 3.7-3 is now closed.

Departure

- LNP DEP 3.7-1

On August 27, 2013, LNP submitted letter number NPD-NRC-2013-037 to address the drilled shaft foundation design criteria for the Annex and Turbine Buildings. The submittal included a departure from the AP1000 DCD Tier 2 information in Sections 3.7.2.8.1 and 3.7.2.8.3, LNP DEP 3.7-1, which addresses the use of site-specific seismic hazard for the lateral design of the drilled shafts supporting the seismic Category II portions of the Annex and Turbine Buildings. In the applicant's submittal, the applicant stated that the drilled shafts supporting the portions of the buildings adjacent to the NI do not conform to any of the six soil profiles described in Subsection 3.7.1.4 of the AP1000 DCD. The applicant further stated that in the conceptual design of the drilled shafts, the vertical seismic demands are consistent with the AP1000 CSDRS which exceed the site-specific vertical seismic demands at the LNP site. However, instead of the AP1000 CSDRS, the applicant used site-specific demands (e.g., PBSRS, RG 1.60 minimum FIRS, and scaled site-specific FIRS) to compute the maximum relative horizontal displacements of the Turbine, Annex, and Radwaste Buildings drilled shaft foundations with respect to the NI. The applicant concluded that the drilled shafts are designed for the AP1000 certified design vertical seismic loads and the site-specific horizontal seismic loads to ensure that the maximum relative displacement of the foundation of these buildings and the NI remains within the DCD limit.

The staff reviewed the applicant's departure, to use site-specific horizontal seismic response spectra for the design of the drilled shafts that support the seismic Category II portions of the Annex and Turbine Buildings. The staff's review focused on the impact of the departure as it relates to the potential seismic interaction between the NI and the adjacent structures. The staff's review found that the applicant used the site-specific horizontal seismic demands (e.g., PBSRS, RG 1.60 minimum FIRS, scaled site-specific FIRS) for the conceptual lateral design of the drilled shafts. The development and use of the site-specific horizontal demands as a representation for the seismic demands at the Levy site was reviewed and found acceptable by the staff in Sections 3.7.1 and 20.1.4.6.5 of this SER. Using the site-specific horizontal demands, the applicant computed a maximum relative displacement in Table 3.7-206 between the NI and the adjacent structures of 0.77 inches. This relative displacement is less than the minimum 2-inch gap at and below grade, and the 4-inch gap above grade as specified in the AP1000 DCD between the NI and the adjacent structures. The design method for the drilled shafts was reviewed and found acceptable in SER Section 3.8.5. Based on the adequacy of the site-specific seismic hazard development, the limited relative displacement as compared to the available gap between the NI and the adjacent structures under those seismic demands, and the adequate design method for the drilled shafts, the staff finds that there is reasonable assurance that the drilled shaft design under the horizontal site-specific seismic demands will be adequate to support the adjacent structures to the NI so as to preclude seismic interaction under the LNP site-specific seismic demands. Accordingly, proposed departure LNP DEP 3.7-1 is acceptable. The staff concludes that the relevant information presented by the applicant is acceptable and satisfies the guidance in Section 3.7.2 of NUREG-0800 and the requirements in 10 CFR Part 50, Appendix A, GDC 2; 10 CFR Part 50, Appendix S; and 10 CFR 100.23. The staff verified that the applicant has appropriately updated Sections 3.7.2.8.1 and 3.7.2.8.3 in Revision 7 of the LNP COL FSAR.

AP1000 COL Information Item

- LNP COL 3.7-1

The NRC staff reviewed the resolution to the COL information item related to the evaluation of existing and new dams included under Section 3.7.2.12 of the LNP COL FSAR. LNP 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 LNP COL FSAR Section 2.4.4 for the details of the evaluation. The applicant states that the LNP site is not subject to flooding from dam failures. The staff's review of LNP COL FSAR Section 2.4.4 is in Section 2.4.4 of this SER, which found the information included therein to be acceptable. Therefore, the NRC staff finds the information added to the LNP COL FSAR by LNP COL 3.7-1 to be acceptable.

The following portion of this technical evaluation section is reproduced from Section 3.7.2.4 of the VEGP SER:

License Conditions

- *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 performed by the licensee and 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 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 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 seismic system analysis, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP COL FSAR is acceptable, and meets the requirements of 10 CFR Part 50, Appendix A, Appendix S, and other staff guidance. The staff based its conclusion on the following:

- LNP SUP 3.7-5 is acceptable because the applicant addressed the relevant information that meets the guidance in Section 3.7.2 of NUREG-0800. In conclusion, the applicant has provided sufficient information for satisfying 10 CFR Part 50, Appendix A, GDC 2; 10 CFR Part 50, Appendix S; and 10 CFR 100.23.

- LNP DEP 3.7-1 is acceptable because the applicant addressed the relevant information that meets the guidance in Section 3.7.2 of NUREG-0800. In conclusion, the applicant has provided sufficient information to meet 10 CFR Part 50, Appendix A, GDC 2; 10 CFR Part 50, Appendix S; and 10 CFR 100.23.

- LNP COL 3.7-1 is acceptable because the applicant addressed the relevant information that meets the guidance in Section 3.7.2 of NUREG-0800. In conclusion, the applicant has provided sufficient information for satisfying 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 LNP COL FSAR, Revision 9, incorporates by reference, Section 3.7.3, “Seismic Subsystem Analysis,” of Revision 19 of the AP1000 DCD. In addition, in LNP COL FSAR Section 3.7, the applicant provided the following:

Departures

- LNP DEP 6.4-2

The applicant provided additional information in Table 3.7-207 of the LNP COL FSAR about LNP 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 LNP 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 LNP 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 the following:

- Comparison with RG 1.12, "Nuclear Power Plant Instrumentation for Earthquakes," Revision 2
- Location and description of instrumentation
- Control room operator notification
- Comparison of measured and predicted responses
- Tests and inspections

3.7.4.2 *Summary of Application*

Section 3.7 of the LNP COL FSAR, Revision 9, incorporates by reference Section 3.7 of the AP1000 DCD, Revision 19. Section 3.7 of the DCD includes Section 3.7.4.

In addition, in LNP COL FSAR Section 3.7.4, the applicant provided the following:

AP1000 COL Information Items

- STD COL 3.7-2 and LNP COL 3.7-2

The applicant provided additional information in STD COL 3.7-2 and LNP COL 3.7-2 in Section 3.7.4.4 to resolve COL Information Item 3.7-2 (COL Action Item 3.7.5-2) on post-earthquake procedures to compare measured and predicted ground motions. In LNP COL 3.7-2, the applicant also stated that post-earthquake operating procedures utilize the guidance of Electric Power Research Institute (EPRI) Reports NP-5930, TR-100082, and NP-6695, as modified and endorsed by the NRC in RG 1.166, "Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Postearthquake 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 a letter dated October 15, 2010, the LNP applicant identified a change to STD COL 3.7-2 in Section 3.7.4.4 of the LNP COL FSAR to address 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 walls.

- STD COL 3.7-5

The applicant provided additional information in STD COL 3.7-5 in Section 3.7.4.2.1 to resolve COL Information Item 3.7-5 (COL Action Item 3.7.5-4) on free field triaxial acceleration sensors. In STD COL 3.7-5, the applicant stated that a free-field sensor will be located and installed within the protected area 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.

Supplemental Information

- STD SUP 3.7-1

The applicant provided supplemental information in LNP 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 inservice during plant operation and shutdown.

- STD SUP 3.7-2

The applicant provided supplemental information in LNP COL FSAR Section 3.7.4.5 to address the test and inspection requirements for the acceleration sensors. In this section, the applicant stated that installation and acceptance testing of the triaxial 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, Items 3.3 and 3.12 refer to interfaces associated with DCD Section 3.7.4. The interface requirements for NRC review (associated with 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.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for seismic instrumentation are given in Section 3.7.4 of NUREG-0800.

The regulatory guidance documents for STD COL 3.7-2 and STD COL 3.7-5 are RG 1.166, RG 1.167, and RG 1.12, and Appendix S to 10 CFR Part 50 that provide for installation of free field triaxial acceleration sensors and establishment of post earthquake procedures to comparing measured and predicted responses.

3.7.4.4 *Technical Evaluation*

The NRC staff reviewed Section 3.7.4 of the LNP 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 related to seismic instrumentation. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting.

The staff has compared STD COL 3.7-2 and STD COL 3.7-5 in the LNP COL FSAR to STD COL 3.7-2, VEGP COL 3.7-2 and VEGP COL 3.7-5 in the VEGP COL FSAR, respectively.

The staff concludes that the information added to the applications for these COL items are sufficiently similar so that the evaluations performed in VEGP SER Section 3.7.4 for VEGP COL 3.7-2 and VEGP COL 3.7-5 are directly applicable to STD COL 3.7-2 and STD COL 3.7-5, respectively. The one notable difference between the VEGP and LNP 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. Also, instead of endorsing the October 15, 2010, VEGP letter regarding post-seismic event gaps in STD COL 3.7-2, the LNP applicant provided this information in its October 15, 2010, letter. In the LNP COL FSAR, the ground surface location at the site of the free-field sensor is not specified, but will be installed using NRC-approved methodology, and the staff concludes that this minor difference does not affect the conclusions reached by the staff.

The following portion of this technical evaluation section is reproduced from Section 3.7.4.4 of the VEGP SER. The review of LNP COL 3.7-2 corresponds to the review of VEGP COL 3.7-2, and is included in the standard content review of STD COL 3.7-2, below:

<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 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 seismic instrumentation, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP COL application is acceptable and meets the requirements of Appendix S to 10 CFR Part 50 and complies with the guidance provided in RGs 1.166, 1.167, and 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 RGs 1.166 and 1.167 and because the applicant has provided sufficient information for satisfying the requirements Appendix S to 10 CFR Part 50 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 has provided sufficient information for satisfying the requirement Appendix S to 10 CFR Part 50 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 in order to keep the maximum number of instruments in service during plant operation and shutdown.

- STD SUP 3.7-2 is acceptable because the applicant has provided sufficient information for satisfying the requirement of Appendix S to 10 CFR Part 50 by committing to complete installation and acceptance testing of the seismic instrumentation prior to initial startup.

3.8 <u>Design Of Category I Structures</u>

3.8.1 Concrete Containment

This section is not applicable to the LNP 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 LNP COL FSAR, Revision 9, incorporates by reference Section 3.8.2, "Steel Containment," of Revision 19 of the AP1000 DCD. In addition, in the LNP COL FSAR, the applicant provided the following:

Departures

- LNP DEP 6.3-1 and LNP DEP 3.2-1

The applicant provided additional information about LNP DEP 6.3-1 and LNP 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 LNP DEP 3.2-1 and LNP 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 LNP COL application are documented in NUREG-1793 and its supplements. Section 21.1 of this report evaluates the departures from the DCD provided in LNP DEP 6.3-1 and LNP DEP 3.2-1.

3.8.3 Concrete and Steel Internal Structures of Steel or Concrete Containment

Structures inside the containment are not part of the containment pressure boundary. They support the reactor coolant system components and related piping systems and equipment inside the containment. They also provide radiation shielding. The 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.

The containment internal structures are constructed of reinforced 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. These steel form modules (liners) consist of 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 back of the permanent steel form where surface attachments to the plate transfer loads into the concrete. Where these surface attachments are seismic Category I, the portion of the steel form module transferring the load into the concrete is classified as seismic Category I.

Section 3.8 of the LNP COL FSAR, Revision 9, incorporates by reference Section 3.8.3, "Concrete and Steel Internal Structures of Steel Containment," of Revision 19 of the AP1000 DCD. In addition, in the LNP COL FSAR Section 3.8.3.7, the applicant provided the following:

AP1000 COL Information Items

- STD COL 3.8-5

The applicant provided additional information related to in-service testing and inspection requirements. This information is reviewed in Section 3.8.5 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 LNP COL application are documented in NUREG-1793 and its supplements.

3.8.4 Other Seismic Category I Structures

The AP1000 DCD defines other seismic Category I structures as the shield building, the auxiliary building, the containment air baffle, Category I cable tray supports, and 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

Section 3.8 of the LNP COL FSAR, Revision 9, incorporates by reference Section 3.8.4, "Other Category I Structures," of Revision 19 of the AP1000 DCD. In addition, in the LNP COL FSAR Section 3.8.4.7, the applicant provided the following:

AP1000 COL Information Items

- STD COL 3.8-5

The applicant provided additional information related to testing and in-service inspection requirements. This information is reviewed in Section 3.8.5 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 evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

3.8.5 Foundations

3.8.5.1 *Introduction*

The NI structures consists of the containment building, the shield building, and the auxiliary building, on a common 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 2-inch gap at and below grade. A 4-inch 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*

Section 3.8 of the LNP COL FSAR, Revision 9, incorporates by reference Section 3.8 of the AP1000 DCD, Revision 19. Section 3.8 of the DCD includes Section 3.8.5.

In addition, in LNP COL FSAR Section 3.8.5, the applicant provided the following:

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 Section 2.5.4.

AP1000 COL Information Items

- LNP COL 2.5-17

In a letter dated September 23, 2010, the applicant proposed identifying, as LNP COL 2.5-17, the information in Section 14.3.3.2 addressing the type of waterproofing system to be used for the below grade exterior walls exposed to flood, and groundwater under seismic Category I structures.

- STD COL 3.8-5

In a letter dated April 19, 2011, the applicant endorsed the August 17, 2010, letter from the VEGP applicant that proposed STD COL 3.8-5, adding new Section 3.8.3.7, 3.8.4.7, and 3.8.5.7 to the FSAR. 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 a letter dated April 19, 2011, the applicant endorsed the October 1, 2010, letter from the VEGP applicant that proposed STD COL 3.8-6, adding a new Section 3.8.6.6 to the FSAR. The applicant provided information in STD COL 3.8-6, addressing the construction procedure program related to safety-related Category I structures.

License Condition

- Part 10, License Condition 6

In its letter dated April 19, 2011 the applicant endorsed the October 1, 2010, letter from the VEGP applicant that 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.

- Part 10, License Condition 4

In its letter dated May 27, 2011, the applicant provided information regarding the Strength Verification and Constructability Testing in accordance with criteria outlined in FSAR Subsection 3.8.5.11.3.

ITAAC

In Part 10, Appendix B, of the LNP COL application, the applicant proposed ITAAC requiring that the 35 foot thick Roller Compacted Concrete (RCC) Bridging mat is seismic Category I and is designed and constructed to bridge over the design basis karst feature when subjected to design basis loads as specified in the Design Description in FSAR 2.5.4.5.4 without loss of structural integrity and the safety related functions. In a letter dated August 19, 2011, the applicant provided revisions to clarify the RCC ITAAC.

In Part 10, Appendix B, of the LNP COL application, the applicant proposed ITAAC requiring that the Drilled Shaft Foundations for the Turbine, Radwaste, and Annex Buildings will preclude movement in excess of the separation provided between the structural elements of the Turbine, Radwaste, and Annex Buildings and the NI structures. In a letter dated August 19, 2011, the applicant provided revisions to clarify the Drilled Shaft Foundation ITAAC.

In Part 10, Appendix B, of the LNP COL application, the applicant proposed ITAAC requiring that the mudmat-waterproofing-RCC interface beneath the NI basemat has a coefficient of friction to resist sliding of ≥ 0.55.

3.8.5.3 *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 (GDC 1, GDC 2, GDC 4, and GDC 5, “Sharing of Structures, Systems, and Components” of Appendix A to 10 CFR Part 50; 10 CFR 50.55(a) and Appendix B, to 10 CFR Part 50) for the foundations are given in Section 3.8.5 of NUREG-0800.

3.8.5.4 *Technical Evaluation*

The NRC staff reviewed Section 3.8.5 of the LNP 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 foundations. The results of the NRC staff’s evaluation of the information incorporated by reference in the LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting.

The staff reviewed the information in the LNP COL FSAR:

Supplemental Information

- STD SUP 3.8-1

In LNP FSAR Section 3.8.5.1, "Description of the Foundations," the applicant referenced Subsection 2.5.4, "Stability of Subsurface Materials and Foundations," which presents the depth of overburden and depth of embedment of the LNP 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 loads of the superstructure to the underlying soils or rocks.

FSAR Section 2.5.4 stated that, below the NI basemat, a 10.7 m (35 ft) thick RCC bridging mat will be used to transmit the NI loads under static and dynamic conditions to the karst foundation.

In its review of the standard supplemental information in LNP COL FSAR Section 2.5.4, the staff determined that the applicant did not provide enough information for the design of the RCC bridging mat. As a result, the staff issued RAI 03.08.05-1, requesting that the applicant provide additional information that details the transfer of the NI loads to the karst foundation through the RCC bridging mat and justifies the use of the RCC bridging mat between the NI basemat and the karst foundation. In addition, the applicant was requested to provide a description of the material properties, installation, and compaction for the RCC bridging mat along with the analysis and design methods used for the bridging mat.

In its response to RAI 03.08.05-1 dated November 20, 2008, the applicant provided a brief description of the methods used to transmit the static and dynamic loads of the NI through the bridging mat and the use of the RCC bridging mat. The applicant stated that the RCC bridging mat is a block of mass concrete that transmits the static and dynamic NI loads to the underlying Avon Park Formation. The applicant also stated that the RCC bridging mat will be installed below the waterproofing membrane where the RCC bridging mat serves as the 'lower mud mat.' Additionally, the applicant provided a description of the material properties, installation, and compaction for the RCC bridging mat, along with the analysis and design methods for the bridging mat. The applicant provided additional information regarding the RCC mix design program. The staff's evaluation of the mix design program is discussed later in this section of this SER.

The staff reviewed the applicant response and concluded that additional information was needed related the design and construction of the RCC bridging mat. As a result, the staff issued RAI 03.08.05-2, requesting the applicant to describe the type of joints to be used at lift boundaries; identify methods to be used to determine tensile and shear strengths and stiffnesses at lift joints, together with their variability, number and types of tests to be used to verify properties; and identify the required shear strength at lift joints, the assumed shear strength at the lift joints and the technical basis for the assumed strength.

In its November 17, 2009, response to RAI 03.08.05-2, the applicant addressed the design and construction of the RCC bridging mat by stating that two types of lift joints may be formed at the bridging mat lift boundaries. The first type of lift joint is bonded with bedding material which acts as a bonding layer. The second type of lift joint that may be used does not include a layer of bedding mix. The applicant described the basis for the shear strength at lift joints, the expected seismic demand and the assumptions used in developing the design strength.

The applicant also stated that there are two testing programs associated with the RCC bridging mat: Production testing that will be conducted during placement of the RCC bridging mat; and an RCC test program that will be conducted prior to construction. The applicant provided a description of the tests that will be performed to assess shear strength for both the base material and for the lift joints including identification of the testing methods to be used. The staff's evaluation of these programs is discussed later in this section of this SER.

In addition, the applicant stated that for the assumed values recommended by the United States Army Core of Engineers (USACE) and ACI-318 for tensile and shear strength are based on empirical historical data, and have been used in the conceptual design phase. The allowable values for tensile and shear strength correspond to the recommended design values with an applied factor of safety. The applicant provided a number of quality control measures that assure that the RCC material is of good quality and to determine the compressive strength and density of the as-place material.

The staff reviewed the applicant's method for designing and constructing the RCC bridging mat and noted that the applicant did not adequately address the number of RCC tests to be performed and how the variability of RCC properties will be assessed; the desired level of performance of the bridging mat; and the transfer of shear or tension between the as-placed material and the bedding joints. As a result, the staff issued RAIs 03.08.05-4, 03.08.05-5, and 03.08.05-6 respectively.

In RAI 03.08.05-4, the staff requested that the applicant provide a detailed description as to how the proposed RCC construction for the Levy plant is similar to the construction for which the shear strength to compressive strength correlations provided by the USACE is appropriate and a detailed description about the methodology for testing of the production bridging mat. In addition, the staff also requested a detailed description as to how the RCC nominal strength capacities will be established and information about the test program that identifies the expected variability of material properties, methods used to quantify the variability, how this variability is incorporated into developing an appropriate factor of safety for design and how the tests that will be performed during production will ensure that the design strengths will be achieved.

In the May 27, 2011, supplemental response to RAI 03.08.05-4 the applicant addressed the shear strength to compressive strength correlation between the proposed RCC for the Levy plant and to similar construction provided by the USACE. The applicant stated that the RCC construction at the Levy plant will follow standard RCC guidance and construction practice, as described in the USACE Engineering Manual EM 1110-2-2006, "Roller-Compacted Concrete," with additional enhancements related to nuclear safety grade Quality Assurance; and that USACE correlations will be used for preliminary conceptual design. Furthermore, the applicant stated that laboratory testing would be used to verify that these relationships are appropriate. Additionally, the applicant intends to perform direct shear testing to evaluate the shear strength along lift surfaces.

The applicant also provided a description of the quality control and inspection plan that will be used during production that will ensure that the placement of production for the RCC is within project specifications. A report, "Post-COL Roller Compacted Concrete Test Plan," submitted by the applicant detailed the design, testing, and construction methods used for large commercial RCC construction projects and how the applicant intends to relate the experience gained on these projects to the Levy Nuclear Plant RCC Bridging Mat.

In the May 27, 2011, supplemental response to RAI 03.08.05-4 the applicant also addressed how the production mat will be sampled to provide assurance that the strength of 'as placed' material exceeds the design requirements. The applicant stated that confirmatory testing of the RCC production mat will be performed using Non-Destructive Testing Methods to ensure that the construction of the RCC and Bedding Joints is in accordance with the RCC construction specification. The applicant provided reports that detail the RCC Test Program. These documents discuss the tests that will be performed during the conceptual design phase and during construction to evaluate variability of material properties and ensure that design strengths will be achieved. Included in the test requirements are a number of direct shear tests that will be performed to verify that the design shear strength is achievable. As a result of the RCC testing program, the ITAAC entry in LNP COLA Part 10 Table 3.8-3 was revised to address the RCC to require consistency of the production LNP Bridging Mat placement and constituents with the design requirements. Additionally, the applicant provided details that describe how the RCC nominal strength capacities will be established. The applicant stated that the nominal capacities are established during the conceptual design phase using standard concrete codes equations, ACI 349-01 and ACI 318-99 and USACE Engineering Manual 1110-2-2006 guidance. These capacities include ACI 318-99 strength reduction factors and load factors of DCD Table 3.8.4-2, consistent with ACI 349-01. On this basis, the applicant concludes that the RCC failure probability is consistent with industry codes. A FEM of the RCC Bridging Mat was used to confirm that these capacities are adequate for the anticipated loading conditions.

Also, in RAI 03.08.05-4, the staff requested a written description of the applicant's expanded test program in order for the staff to complete its evaluation of the acceptability of the final test program. The staff requested that the test program identify the expected variability of material properties, methods used to quantify the variability, how this variability is incorporated into developing an appropriate factor of safety for design and how the tests that will be performed during production will ensure that the design strengths will be achieved.

In its May 27, 2011, supplemental response to RAI 03.08.05-4, the applicant addressed the variability of the RCC materials and stated that the variability of the RCC materials is accounted for in the mix design process. The applicant also stated that based on previous commercial RCC experience, the expected coefficient of variation on the compressive strength of the RCC is approximately 14 percent with the strict quality control measures that will be in place. Additionally, the applicant stated that the targeted RCC mix design strength accounts for forecasted variability.

The staff reviewed the applicant's responses along with associated reports, calculations, and applicable codes and standards provided by the applicant related to the design of the RCC bridging mat and concludes that the applicant's design methodology and results, construction methods, testing and inspection requirements are acceptable. The applicant design methodology and approach demonstrate that the stresses in the bridging mat will remain within concrete code allowable limits and is therefore assured of performing its required function. Because the applicant has complied with the regulatory requirements in 10 CFR 50.55(a) and GDC 1 by providing the details requested in response to RAI 03.08.05-4, the staff considers the RAI to be resolved. The incorporation of changes in a future revision to the LNP COL FSAR is being tracked as **Confirmatory Item 3.8-1**. The staff conclusion relies heavily on the successful placement of the large scale RCC test pad to be completed prior to the construction of the bridging mat. As a result, the staff has reviewed ITAAC 3.8.3, which is discussed later in this section of this SER.

Resolution of Confirmatory Item 3.8-1

Confirmatory item 3.8-1 is an applicant commitment to revise the LNP COL FSAR to address RCC mat construction, testing, and associated ITAAC. The staff verified that the LNP COLA was appropriately revised. As a result, Confirmatory Item 3.8-1 is now closed.

In its August 18, 2010, response to RAI 03.08.05-5, the applicant addressed the desired level of performance of the RCC Bridging Mat. The applicant stated that the performance is assured by the method of analysis using the load factors and strength reduction factors from ACI 349-01 in conjunction with the equations and methodology for plain concrete from ACI 318. The applicant further stated that a FEM with solid elements under service loading conditions was used to evaluate the demand on the bridging mat and includes an evaluation of 10-foot diameter voids and a 10-foot wide strip cavity beneath the RCC Bridging Mat. The applicant states that the calculated shear stresses across the lift joint do not exceed the allowable shear stress.

The staff has reviewed the load factors and strength reduction factors for the design of the RCC bridging mat and concludes that the strength reduction factor to estimate a target factor of safety forms an adequate basis for assuring that the desired level of performance for the RCC mat supports the NI structure. Because the applicant has provided the details requested in response to RAI 03.08.05-5 the staff considers the RAI to be resolved. The incorporation of changes in a future revision to the LNP COL FSAR is being tracked as **Confirmatory Item 3.8-2.**

Resolution of Confirmatory Item 3.8-2

Confirmatory Item 3.8-2 is an applicant commitment to update the LNP COL FSAR to provide requested details regarding the assumptions made to justify the expected performance of the RCC mat. The staff verified that the LNP COL FSAR was appropriately revised. As a result, Confirmatory Item 3.8-2 is now closed.

In FSAR Section 2.5.4.8 the applicant presented the results of its liquefaction analysis. Because both the Avon Park limestone and the RCC bridging mat are not prone to liquefaction, the applicant stated that liquefaction cannot occur below the NI. The section further states that liquefaction will not affect the NI and that the drilled shafts will be designed in a manner that precludes soil liquefaction effects from having an impact on the surrounding structures such that they might unfavorably interact with the NI. In reviewing Section 2.5.4.8 of the LNP FSAR, the staff observed that liquefaction has not been considered in any of the seismic interaction analyses for the Annex, Radwaste, and Turbine Buildings. As a result, the staff issued RAI 03.08.05-3 requesting that the applicant provide an explanation of how the AP1000 DCD seismic interaction analysis for the Annex Building bounds the Levy site given that liquefaction was not considered in the DCD analysis.

In its January 25, 2011, revised response to RAI 03.08.05-3, the applicant stated that the Turbine, Annex, and Radwaste Building foundation mats displacements calculated in its response to RAI 03.08.05-3 were superseded by those calculated in the supplemental response to RAI 03.08.05-7. The applicant stated in the response to RAI 03.08.05-07 that remediation measures for pockets of potential liquefaction will be taken through installation of vertical and horizontal drains to prevent buildup of excess pore pressure that is required for liquefaction to occur. The staff has reviewed the response and associated calculations and concludes that the proposed remediation is adequate to mitigate the effects of potential liquefaction on the seismic interaction between adjacent structures. The staff's evaluation of the supplemental response is documented in Section 2.5.1.1.1 of this SER.

Additionally, in RAI 03.08.05-7, the staff requested that the applicant provide further clarification on how the estimate of relative displacements between adjacent structures was calculated for seismic loads. One displacement source that did not appear to be considered was displacement that may develop from deformation of the soils along the sides of the RCC mat, including the engineered fill. Second, the staff requested that the applicant describe the procedure(s) that will be used to assess the significance of the interaction effects between the drilled shafts in the final design. A third question by the staff was related to the ground motion used to assess liquefaction potential and global displacement of structures. The applicant computed displacements that were associated with the GMRS and the related PBSRS. The staff requested that the applicant clarify why displacement and liquefaction are not evaluated to the higher desired performance goal level (1×10^{-05}). The final question by the staff was related to the design and installation of the drilled shaft foundations for the seismic category II and nonsafety-related adjacent buildings (Turbine Building, Annex Building, and Radwaste Building).

In its response to RAI 03.08.05-07 the applicant addressed how the estimate of relative displacements between adjacent structures were calculated for seismic loads by stating that the displacements have been computed which consider the deformation of soil adjacent to the RCC and drilled shaft-to-shaft interaction. In addition, the displacements associated with the performance goal level were evaluated.

The staff reviewed the applicants RAI responses and associated calculations and concludes that the seismically induced displacements are significantly smaller than the seismic gap provided in the DCD. As a result of the detailed information presented by the applicant, the ITAAC entry in LNP COLA Part 10 Table 3.8-4 was revised to address the drilled shaft foundations for the Turbine, Radwaste, and Annex Buildings to preclude movement so as not to exceed the separation provided between these buildings and the NI structures. The details of the conceptual drilled shaft design and installation procedure are described in Section 3.7.1 of this SER. Thus, the staff considers RAI 03.08.05-7 to be resolved. The incorporation of changes in a future revision to the LNP COL FSAR is being tracked as
Confirmatory Item 3.8-3.

<u>Resolution of Confirmatory Item 3.8-3</u>

Confirmatory Item 3.8-3 is an applicant commitment to update its FSAR to include details regarding its calculation of displacements between adjacent structures for seismic loads. The staff verified that the FSAR was appropriately revised. As a result, Confirmatory Item 3.8-3 is now closed.

In the applicant response to RAI 03.08.05-2, the applicant described a number of quality control measures that will provide information needed to ensure that the RCC material is of good quality and to determine the compressive strength and density of the as-placed material. In reviewing the applicant response, the staff determined that none of the quality control measures appeared to address the capability of the as-placed material to transfer shear or tension across the as-constructed bedding joints. Thus, the staff issued RAI 03.08.05-6, requesting the applicant to provide additional information that describes the transfer of shear or tension between the as-placed material and the bedding joints.

In its revised response to RAI 03.08.05-6, the applicant stated that the revised response incorporates pre-COL RCC testing results and the revised post-COL RCC test as discussed at the April 27-28, 2011 meeting that the staff participated in with the applicant in Tucson, Arizona to witness the RCC Specialty Tests. As part of that meeting, the staff requested the following information be provided:

I. Summary report of commercial RCC experience and test data.

II. Description of materials, processes, and equipment types/sizes from commercial projects and a commitment that those used for LNP will be similar.

III. Identify the specific RCC mix design for the LNP project, and confirm the acceptability of this mix to provide the characteristics required for the foundation design.

IV. Submit the 90-day specialty test report verifying RCC strength characteristics.

V. Submit post-COL RCC Strength Verification and Constructability Testing plan.

VI. Add a new FSAR Subsection 3.8.5.11 summarizing information on commercial test results, RCC mix design, pre-COL 90-day testing, and commitments for the post-COL testing and the use of equipment and process validated by the post-COL testing in production construction of the RCC Bridging mat.

VII. Add a new License Condition for post-COL testing stating that the licensee will complete 180-days prior to construction, the 90-day test report for the Strength Verification and Constructability Testing in accordance with the criteria outlined in FSAR Section 3.8.5.11.3.

VIII. Revise COLA Part 10 “Table 3.8-3: LNP COLA RCC ITAAC”

The applicant stated that the LNP RCC construction will follow industry standard methods that have been successfully implemented on large commercial RCC projects. The applicant provided a detailed description of the methods that will be used. Additionally, the applicant summarized the RCC production and placement practices that were used for three large commercial RCC projects and concluded that the properties of the aggregates, cement, and fly ash planned for the LNP RCC bridging mat will meet or exceed the requirements used for these successful commercial projects. The applicant stated that the experience from the large-scale commercial RCC projects provides assurance that LNP RCC bridging mat can be successfully constructed and have the desired strength.

Additionally, the applicant provided a detailed test plan that describes the quality control and inspection to occur during production construction, and stated that the implementation of the plan will ensure that the mixing, placement, and compaction of production RCC complies with the LNP RCC construction specifications.

The applicant provided details that describe the RCC testing results from three large commercial RCC projects and concluded the following:

- The compressive strengths measured during production construction exceeded those that were measured during pre-construction mix design laboratory testing. Thus, laboratory testing during RCC mix design provides reasonable assurance that the desired RCC compressive strength will be achieved or exceeded during production construction.

- The measured modulus of elasticity from commercial testing correlates well with that computed using ACI 318-99 Section 8.5.1 method. Thus use of ACI 318-99 Section 8.5.1 for modulus of elasticity for RCC design is appropriate.

- The USACE EM 1110-2-2006 correlation of the direct tensile strength of RCC to approximately 75 percent of the split tensile strength trends close to the ACI 318-99 equation 22-2 for tensile strength. Thus the use of ACI 318-99 equation 22-2 for tensile strength in RCC design is appropriate.

- Shear tests performed on pre-cracked (at lift joints) block samples show that the friction angle when concrete bedding mix is used is greater than the 45 degrees design value provided in the USACE EM 1110-2-2006. Thus, the use of 45 degrees friction angle for shear capacity in RCC design across lift joints is appropriate.

Furthermore, the applicant stated that testing of the production RCC mat will provide confirmation that the construction of the RCC and bedding joints is in accordance with the RCC construction specifications. The applicant described the testing that will occur during construction of the RCC bridging mat, including quality control testing. In addition, the applicant stated that Post-COL RCC and bedding mix strength verification and constructability testing (RCC Test Program Phase IV) will be performed on a large test pad. This testing will be performed post-COL but prior to construction of the LNP bridging mat for the following reasons:

- Due to the limitation on mixing and compaction equipment sizes that can be used in a laboratory setting, the required compaction cannot be achieved in a laboratory setting. A larger scale test pad in an open field setting is required.

- Because RCC design strength is specified as the 365-day strength, it is not practical to perform destructive testing on the RCC bridging mat during construction on cored or block cut test specimens. The post-COL RCC strength verification and constructability testing will be performed post-COL at the LNP site. The test pad construction will use mixing, placement, and compaction procedures and equipment comparable to those that will be used during LNP RCC bridging mat construction. The constitutive materials for the RCC mix will be comparable to that used in the RCC mix design program. The post-COL strength verification and constructability test report with 90-day test results will be completed at least 180-days prior to start of LNP RCC bridging mat construction.

The applicant stated that the RCC construction specifications, non-destructive testing and quality controls during construction together with implementing procedures and equipment comparable to those used on past successful RCC projects, pre-COL RCC mix design testing, the pre-COL RCC testing, and planned post-COL RCC testing using a large test pad provides sufficient assurance that the LNP design compressive and tensile strengths, and shear strengths across lift joints will be achieved during the RCC bridging mat construction using the RCC and bedding mix, mixing and placement procedures and equipment, and the compaction equipment specified for construction.

The applicant provided details of the mix design program for the RCC and bedding mix and stated that the program demonstrates that design workability and strength requirements can be achieved with the trial mixes and constituent materials procured for the program. The applicant described the preliminary testing on cored cylinders from the test panels indicated that the concrete in the test panels did not attain the desired compressive or tensile strengths and indicated that this low strength is believed to be due to the constructability issues related to construction of the laboratory-scale test panels that required the use of small mixing and compaction equipment.

The applicant states that conducting the “Roller Compacted Concrete Strength Verification and Constructability Testing,” post-COL but prior to production construction is acceptable because of the following reasons:

- RCC Mix Design testing shows that the specified compressive and split tensile strength can be achieved with the trial RCC mixes.

- Laboratory cast cylinders from both the mix design program and the RCC specialty test program using the LNP selected RCC design mix exceed the compression and tensile strengths required for the project.

- Biaxial shear test results on block samples from the RCC specialty test panel yielded shear strengths at least 1.67 times the maximum design demand shear across lift joints, despite the fact that the test panels did not achieve the desired compressive strength.

- Post-COL RCC Strength Verification and Constructability Testing (RCC Test Program Phase IV) as described in Attachment 2 of the applicant letter dated May, 27, 2011 "Post-COL Roller Compacted Concrete Test Plan," Revision 3, will be conducted prior to RCC bridging mat construction to verify that the design specified compressive strength, ACI 318-99 specified tensile strength, and USACE EM 1110-2-2006 specified shear strengths across lift joints can be achieved. For this post-COL test program, the test report with 90-day test results will be completed at least 180-days prior to start of RCC bridging mat construction. For these tests, constructability issues experienced during pre-COL specialty testing in a laboratory setting will be avoided by the use of production construction scale mixing, placement, and compaction equipment. The test pad for the pre-construction tests will be constructed using mixing and placement procedures similar to those that will be used for the LNP RCC bridging mat construction.

- The proposed License Condition for post-COL RCC testing states: "The licensee will complete 180-days prior to construction, the 90-day test report for the Strength Verification and Constructability Testing in accordance with the criteria outlined in FSAR Subsection 3.8.5.11.3 and make it available to the NRC."

- Two other seismic demands were evaluated based on the SSI analyses results. The applicant demonstrated that the maximum bearing pressure on the RCC bridging mat beneath the NI basemat is 20.29 ksf, less than the AP1000 maximum bearing pressure of 35 ksf. In addition, the maximum base shear on the RCC bridging mat corresponds to base shear to vertical load ratio of 0.12 for the NI which is less than the AP1000 maximum ratio of 0.55.

NRC staff reviewed the information provided by the applicant in response to the information requested during the April 27-28, 2011 audit and concludes that the information provided by the applicant adequately considered the quality requirements for the material and placement of the RCC bridging mat will ensure that the bridging mat as built will perform its intended function. The staff finds this information adequate because it meets the requirements of 10 CFR 50.55(a) and GDC 1. The staff agreed with the applicant's assessment that conducting the "Roller Compacted Concrete Strength Verification and Constructability Testing," post-COL but prior to production construction is acceptable. Thus, the staff considers RAI 03.08.05-6 to be resolved. The incorporation of changes in a future revision to the LNP COL FSAR is being tracked as **Confirmatory Item 3.8-4.**

Resolution of Confirmatory Item 3.8-4

Confirmatory Item 3.8-4 is an applicant commitment to provide details regarding the testing of the RCC. The required information was provided by the applicant as part of its response to RAI 03.08.05-4. The staff found the information in the applicant's response to RAI 03.08.05-4 to be acceptable. The staff verified that the FSAR was appropriately revised. As a result, confirmatory Item 3.8-4 is now closed.

AP1000 COL Information Items

- LNP COL 2.5-17

In a letter dated September 23, 2010, the LNP applicant proposed identifying, as LNP COL 2.5-17, the information in Section 3.8.5.1 addressing the type of waterproofing system to be used for the below grade exterior walls exposed to flood, and groundwater under seismic Category I structures. The applicant provided a waterproofing material to be used for the below grade, exterior walls exposed to flood and groundwater under seismic Category I structures. The applicant stated that a sheet type waterproofing membrane will be used for both the horizontal and vertical surfaces under Seismic Category I structures. The applicant further stated the waterproofing material will be qualified by test, with commercial grade dedication and lab testing to achieve a minimum coefficient of friction of 0.55. The performance requirements to be met by the COL applicant for the waterproofing material are described in Section 3.4.1.1.1.1 of the AP1000 DCD. Thus, the NRC staff considers LNP COL 2.5-17 to be resolved.

The following portion of this technical evaluation section is reproduced from Section 3.8.5.4 of the VEGP SER:

- *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***

Resolution of Standard Content Confirmatory Item 3.8-2

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 2.8.5.7, 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***

Resolution of Standard Content Confirmatory Item 3.8-3

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.

The following portion of this technical evaluation section is reproduced from Section 3.8.5.4 of the VEGP SER:

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 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 conditions acceptable:

- The licensee shall perform and satisfy the RCC ITAAC in SER Table 3.8-1.

- The licensee shall perform and satisfy the Drilled Shaft Foundation ITAAC in SER Table 3.8-2.

- The licensee shall perform and satisfy the Waterproof Membrane ITAAC in SER Table 3.8-3.

- 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.

- License Condition (3-5) – The licensee shall complete and make available to the NRC 180-days prior to construction the 90-day test report for the Strength Verification and Constructability Testing in accordance with the criteria outlined in FSAR Subsection 3.8.5.11.3.

3.8.5.6 *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 foundations, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP COL FSAR Section 3.8.5 forms an adequate basis for the design and construction of the RCC bridging mat at the LNP site, and meets the requirements of 10 CFR 50.55(a), GDC 1, 2, 4, and 5 to 10 CFR Part 50, Appendix A, and 10 CFR 50, Appendix B. The staff based its conclusion on the following:

- STD SUP 3.8-1 is acceptable because the applicant addressed the relevant information that meets the guidance in Section 3.8.5 of NUREG-0800. In conclusion, the applicant has provided sufficient information for satisfying 10 CFR Part 50, Appendix A, GDC 1, 2, 4, and 5.

- COL 2.5-17 In a letter dated September 23, 2010, the LNP applicant proposed identifying, as LNP COL 2.5-17, the information in Section 3.8.5.1 addressing the type of waterproofing system to be used for the below grade exterior walls exposed to flood, and groundwater under seismic category I structures.

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 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.

Section 3.9 of the LNP COL FSAR, Revision 9, incorporates by reference, with no departures or supplements, Section 3.9.1, “Special Topics for Mechanical Components,” 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 LNP 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.

Section 3.9 of the LNP COL FSAR, Revision 9, incorporates by reference, with no departures or supplements, Section 3.9.2, "Dynamic Testing and Analysis of Systems, Structures and Components," 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 LNP 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.

The criteria for the SSC design include the following considerations:

- Loading combinations, design transients, and stress limits
- Pump and valve operability assurance
- Design and installation criteria of Class 1, 2, and 3 pressure-relieving devices
- Component and piping supports

3.9.3.2 *Summary of Application*

Section 3.9 of the LNP COL FSAR, Revision 9, incorporates by reference Section 3.9 of the AP1000 DCD, Revision 19. Section 3.9 of the DCD includes Section 3.9.3.

In addition, in LNP COL FSAR Section 3.9.3, the applicant provided the following:

Departures

- LNP DEP 6.4-2

The applicant provided additional information in Table 3.9-202 of the LNP COL FSAR about LNP 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 LNP 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 additional information in 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] subsection 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 SER.

- STD COL 3.9-3

The applicant provided additional information in STD COL 3.9-3 to address COL Information Item 3.9-3 (COL Action Item 3.9.8-1), 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 additional information in STD COL 3.9-5 to address COL Information Item 3.9-5 (COL Action Item 3.12.5.10-1), which addresses pressurizer surge line monitoring. Evaluation of this particular COL information item is provided in Section 3.12 of this SER.

- STD COL 3.9-7

In its letter dated June 21, 2011, the applicant endorsed the letter dated April 23, 2010, from the VEGP applicant, that 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 to complete the ITAAC added to verify the design.

Supplemental Information

- STD SUP 3.9-3

The applicant provided supplemental information in STD SUP 3.9-3 to describe snubber design and testing and snubber installation requirements. In a letter dated June 21, 2011, the applicant stated that a correction will be made to the left margin annotation (LMA) in a future revision to the FSAR. The current version of the LNP COL FSAR has the LMA as STD COL 3.9-3 instead of STD SUP 3.9-3.

3.9.3.3 *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 ASME Code Class 1, 2, and 3 components, component supports, and CS structures are given in Section 3.9.3 of NUREG-0800.

3.9.3.4 *Technical Evaluation*

The NRC staff reviewed Section 3.9.3 of the LNP 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 functional design of ASME Code Class 1, 2, and 3 components and component supports and CS structures. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 3.9.3.4 of the VEGP SER:

> *AP1000 COL Information Items*
>
> - *STD COL 3.9-3 and STD SUP 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.

Resolution of LMA

In a letter dated June 21, 2011, the applicant stated that a correction will be made to the LMA in a future revision to the FSAR. The current version of the LNP COL FSAR has the LMA as STD COL 3.9-3 instead of STD SUP 3.9-3. The incorporation of the planned changes to the LNP COL FSAR will be tracked as **Confirmatory Item 3.9-7.**

Resolution of Confirmatory Item 3.9-7

Confirmatory Item 3.9-7 is an applicant commitment to properly add LMA STD SUP 3.9-3 in FSAR Section 3.9.3.4.4. The staff verified that the desired change had been made. As a result, Confirmatory Item 3.9-7 is now closed.

3.9.3.5 *Post Combined License Activities*

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

3.9.3.6 *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 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 LNP COL FSAR related to this section.

The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP 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:

- LNP 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 and STD SUP 3.9-3 are acceptable because the applicant addressed the relevant information that meets the guidance in Section 3.9.3 of NUREG-0800. In conclusion, the applicant has provided sufficient information for satisfying 10 CFR Part 50, Appendix A, GDC 1 and 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. Because 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.

Section 3.9 of the LNP COL FSAR, Revision 9, incorporates by reference, with no departures or supplements, Section 3.9.4, "Control Rod Drive System (CRDS)," 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 LNP COL application 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.

Section 3.9 of the LNP COL FSAR, Revision 9, incorporates by reference, with no departures or supplements, Section 3.9.5, "Reactor Pressure Vessel Internals," 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 LNP COL application are documented in NUREG-1793 and its supplements.

3.9.6 Inservice Testing of Pumps and Valves (Related to RG 1.206, Section C.III.1, Chapter 3, C.I.3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints")

3.9.6.1 *Introduction*

In this section, the NRC staff describes its review of the functional design, qualification, and 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 LNP Units 1 and 2. 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 LNP 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 LNP Units 1 and 2.

3.9.6.2 *Summary of Application*

Section 3.9 of the LNP COL FSAR, Revision 9, incorporates by reference Section 3.9 of the AP1000 DCD, Revision 19. Section 3.9 of the DCD includes Section 3.9.6.

In addition, in LNP COL FSAR Section 3.9.6, the applicant provided the following:

<u>*Departures*</u>

- LNP DEP 6.4-2

The applicant provided additional information in Table 3.9-203 of the LNP COL FSAR about LNP 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 LNP 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 additional information in several sections of LNP 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 LNP Units 1 and 2. For example, the LNP COL FSAR supplements the provisions in the AP1000 DCD with respect to the Edition and Addenda of the ASME OM Code applicable to the description of the IST program for LNP Units 1 and 2, 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. Under STD COL 3.9-3, the applicant supplemented the AP1000 DCD provisions for design, installation, preservice examination and testing, and inservice examination and testing of dynamic restraints (snubbers) in LNP COL FSAR Section 3.9.3.4.4, "Inspection, Testing, Repair, and/or Replacement of Snubbers."

The AP1000 DCD addresses the functional design and qualification of mechanical equipment to be used at an AP1000 nuclear power plant in several DCD sections. For example, 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. 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 ISIs, and periodic inservice operations to be performed in situ to verify the functional capability of the valves. 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. 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." 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. Section 5.4.10, "Component Supports," includes provisions for design, testing, and inspection of component supports in the reactor coolant system. The LNP 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. 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 LNP 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) to be used in the AP1000 reactor 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." 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 Section 3.9.8.3, "Snubber Operability Testing." Section 3.9.8.3 states that COL applicants referencing the AP1000 design will develop a program to verify operability of essential snubbers. The LNP COL FSAR provides supplemental information for Section 3.9.3.4.4 regarding snubbers. For example, LNP 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 (B&PV Code); the ASME OM Code; and ASME Standard QME-1-2007. LNP 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. LNP COL FSAR Table 3.9-201, "Safety Related Snubbers," provides a list of safety-related snubbers to be installed at LNP, 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. Table 3.9-16, "Valve Inservice Test Requirements," in AP1000 DCD, 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. LNP 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 LNP Units 1 and 2 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 LNP COL FSAR, the applicant describes 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 indicates 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 LNP 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, Items G2 and G5

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.

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 NRC staff's review of the LNP 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 Appendix A to 10 CFR Part 50, 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. Paragraph (11) in 10 CFR 52.79(a) 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. Paragraph (29)(i) in 10 CFR 52.79(a) requires that a COL application provide plans for conduct of normal operations, including maintenance, surveillance, and periodic testing of SSCs. Paragraph (37) in 10 CFR 52.79(a) 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 the NRC regulations. For example, Section C.IV.4 in RG 1.206 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, dated February 22, 2006.

The NRC staff followed Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints," of NUREG-0800 in its review of the LNP COL application. The staff also compared the LNP COL FSAR information with the guidance provided in RG 1.206. Appendix 1AA, "Conformance with Regulatory Guides," indicates that the COL application conforms to RG 1.206 without exceptions related to the IST Program. In addition, Table 1.9-202, "Conformance with SRP Acceptance Criteria," in the LNP COL FSAR indicates that the COL application conforms to NUREG-0800, Section 3.9.6.

3.9.6.4 *Technical Evaluation*

The NRC staff reviewed Section 3.9.6 of the LNP 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 functional design, qualification and IST programs for pumps, valves, and dynamic restraints. The results of the NRC staff's evaluation of the design-related information incorporated by reference in the LNP 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 SER section.

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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. The 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 3.9.6.4 of the VEGP SER:

> *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****.*

Resolution of Standard Content Confirmatory Item 3.9-4

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****.*

Resolution of Standard Content Confirmatory Item 3.9-5

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.

Technical Specifications

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****.*

Resolution of Standard Content Confirmatory Item 3.9-6

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

License Conditions

- *Part 10, License Condition 3, Items G2 and G5*

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.

Squib Valves

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 LNP Units 1 and 2 squib valves.

The surveillance program is established to provide reasonable assurance that the LNP 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 (3-6) – Before initial fuel load, the licensee shall implement (1) the Preservice Testing Program and (2) the Motor-Operated Valve Testing Program.

- License Condition (3-7) – 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.

- License Condition (3-8) – 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 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 IST Program, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the design-related information incorporated by reference in the LNP 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 SER section.

In addition, the staff concludes that the relevant information presented in the LNP COL FSAR is acceptable and meets the guidance in Section 3.9.6 of NUREG-0800 and in RG 1.206. The staff based its conclusion on the following:

- LNP 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.

Section 3.9 of the LNP COL FSAR, Revision 9, incorporates by reference, with no departures or supplements, Section 3.9.7, "Integrated Head Package" 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 LNP COL application are documented in NUREG-1793 and its supplements.

3.10 <u>Seismic and Dynamic Qualification of Mechanical and Electrical Equipment</u>

3.10.1 Introduction

Seismic and dynamic qualification of seismic Category I equipment includes the following types:

- Safety-related active mechanical equipment that performs a mechanical motion while accomplishing a system safety-related function. Examples include pumps, valves, and valve operators.

- 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 in order to fulfill its design safety-related function.

- 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. This equipment includes devices associated with systems essential to safe shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal, or is otherwise essential in preventing significant release of radioactive material to the environment or in mitigating the consequences of accidents.

The criteria for the seismic and dynamic qualification include the following considerations:

- Adequacy of seismic and dynamic qualification input motions.
- Methods and procedures for qualifying electrical equipment, instrumentation, and mechanical components.
- Methods and procedures for qualifying supports of electrical equipment, instrumentation, and mechanical components.
- Documentation.

3.10.2 Summary of Application

Section 3.10 of the LNP COL FSAR, Revision 9, incorporates by reference Section 3.10 of the AP1000 DCD, Revision 19.

Section 3.10 of the LNP 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.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the seismic and dynamic qualification of mechanical and electrical equipment are given in Section 3.10 of NUREG-0800.

3.10.4 Technical Evaluation

The NRC staff reviewed Section 3.10 of the LNP 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 seismic and dynamic qualification program. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 3.10.4 of the VEGP SER:

> *<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 will be tracked as ***Confirmatory Item 3.10-1****.*

Resolution of Standard Content Confirmatory Item 3.10-1

Confirmatory Item 3.10-1 is an applicant commitment to revise its FSAR to address seismic qualification for Category I equipment. The staff verified that the VEGP COL FSAR was appropriately updated. As a result, Confirmatory Item 3.10-1 is now closed.

3.10.5 Post Combined License Activities

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

3.10.6 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 seismic and dynamic qualification program, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The staff compared the information in the application to the relevant NRC regulations, the acceptance criteria in Section 3.10 of NUREG-0800. 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 which specifies the replacement frequencies of affected safety-related equipment in harsh environments, and nonsafety-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. 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 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 LNP COL application with its incorporation by reference of the AP1000 DCD and supplemental information to fully describe the EQ and other related operational programs in support of the COL application for LNP Units 1 and 2.

3.11.2 Summary of Application

Section 3.11 of the LNP COL FSAR, Revision 9, incorporates by reference Section 3.11 of the AP1000 DCD, Revision 19. Section 3.11 of the AP1000 DCD describes the EQ Program for electrical and mechanical equipment to be used in the AP1000 certified design.

Departures

- LNP DEP 3.11-1

In a letter dated May 13, 2013, the applicant proposed departure LNP DEP 3.11-1 relating to the "Environmental Zone" for three spent fuel pool level instruments.

- LNP 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 LNP COL FSAR about LNP 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 LNP 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 LNP COL FSAR Section 3.11.5, "Combined License Information Item For Equipment Qualification File," the applicant provided additional information to address COL Information Item 3.11-1 (COL Action Item 3.11.2-1) regarding administrative control of the EQ Program for LNP Units 1 and 2.

License Conditions

- Part 10, License Condition 3, Item G1

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.

3.11.3 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 EQ of mechanical and electrical equipment are given in Section 3.11 of NUREG-0800.

The applicable regulatory requirements or guidance 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, Section C.IV.4 in RG 1.206 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 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, dated February 22, 2006.

3.11.4 Technical Evaluation

The NRC staff reviewed Section 3.11 of the LNP 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 EQ of mechanical and electrical equipment. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting.

Departure

- LNP DEP 3.11-1

In a letter dated May 13, 2013, the applicant proposed 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. The applicant stated that this change corrects inconsistency shown in the DCD. 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 has reviewed the proposed 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 Section 3.11.4 of the VEGP SER:

> <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.

Supplemental Review of Operational Aspects of the EQ Program

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.

License Conditions

- *Part 10, License Condition 3, Item G1*

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 (3-9) – Before initial fuel load, the licensee shall implement the Environmental Qualification Program.

- License Condition (3-10) – 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 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 EQ Program, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP COL FSAR is acceptable and meets the guidance in Section 3.11 of NUREG-0800 and in RG 1.206. The staff based its conclusion on the following:

- LNP 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.

- LNP 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 LNP Units 1 and 2, 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 (Related to RG 1.206, Section C.III.1, Chapter 3, C.I.3.12, "Piping Design Review")

3.12.1 Introduction

This section covers the design of the piping system and piping support for seismic Category I, Category II, and nonsafety systems. It also discusses the adequacy of the structural integrity, as well as the functional capability, of the safety-related piping system, 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 NS 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

Chapter 3 of the LNP COL FSAR, Revision 9, incorporates by reference Chapter 3 of the AP1000 DCD, Revision 19. Sections 3.7 and 3.9 of the AP1000 DCD address Section 3.12, "ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and their Associated Supports" of NUREG-0800.

In addition, in LNP COL FSAR Sections 3.7 and 3.9, the applicant provided the following:

AP1000 COL Information Item

- STD COL 3.9-2

The applicant provided additional 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 NRC's review and that reconciliation of these documents is completed after construction and prior to fuel load.

- STD COL 3.9-5

The applicant provided additional 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 its letter dated September 23, 2010, the applicant endorsed the letter dated April 23, 2010, from the VEGP applicant, that 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.

Supplemental Information

- LNP SUP 3.7-3

LNP SUP 3.7-3 adds new Sections 3.7.1.1.1 and 3.7.1.1.2 to provide the seismic response spectra design information for the LNP site.

License Condition

- Part 10, License Condition 2, Item 3.9-7

In its letter dated September 23, 2010, the applicant endorsed the letter dated April 23, 2010, from the VEGP applicant, that proposed a license condition addressing the as-designed piping analysis completion schedule.

ITAAC

In its letter dated September 23, 2010, the applicant endorsed the letter dated April 23, 2010, from the VEGP applicant, that proposed 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.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the pipe and support analysis are given in Section 3.12 of NUREG-0800.

3.12.4 Technical Evaluation

The NRC staff reviewed Section 3.9 of the LNP 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 piping design review. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 3.12.4 of the VEGP SER:

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.

AP1000 COL Information Items

- *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 "Pressurizer Surge Line Thermal Stratification." 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, and August 6, 2010, letters will be tracked as ***Confirmatory Item 3.12-1****.*

Resolution of 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.# [where # is the next sequential number] 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 [Section 14.3.3.2 for LNP] *for pipe analysis and add an ITAAC (Table 3.8-2)* [Table 3.8-3 for LNP] *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)* [Table 3.8-3 for LNP] *were appropriately updated. As a result, Confirmatory Item 3.12-2 is now closed.*

- LNP COL 3.7-3

Sections 3.7.1.1.1 and 3.7.1.1.2 of the LNP COL FSAR provide the seismic response spectra design information. The staff reviewed the seismic response input information and SSI analysis and documented its evaluation in Section 3.7.1 and 3.7.2 of this SER. The staff also concluded that the LNP site-specific ISRS are enveloped by the ISRS of the AP1000 CSDRS and HRHF during the review of section 3.7.1 and 3.7.2. On the basis of the LNP site-specific piping analysis seismic input ISRS are enveloped by the ISRS of the AP1000 design, the staff finds that the AP1000 standard piping analyses are acceptable for the LNP site. As discussed above, Confirmatory Item 3.12-2 for LNP is now resolved. In a September 23, 2010, letter, the applicant endorsed SNC's April 23, 2010 response that included planned changes to the FSAR. These changes were included in Revision 4 to the LNP COL FSAR. Therefore, this issue is resolved for the LNP COL application.

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 SER Table 3.12-1.

- License Condition (3-11) – 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 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 piping design, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP COL application is acceptable and meets the NRC regulations. The staff based its conclusion on the following:

- STD COL 3.9-2 is acceptable because it meets the general requirements of the ASME Code, as specified by 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 by 10 CFR 50.55a.

- LNP SUP 3.7-3 is acceptable because the LNP site-specific ISRS are enveloped by the ISRS of the AP1000 CSDRS and HRHF spectra and, therefore, the AP1000 DCD piping design analyses are acceptable for the LNP site.

Table 3.6-1. Pipe Rupture Hazards Analysis ITAAC

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Systems, structures, and components (SSCs), that are required to be functional during and following a design basis event shall be protected against or qualified to withstand the dynamic and environmental effects associated with analyses of postulated failures in high and moderate energy piping.	Inspection of the as-designed pipe rupture hazard analysis report will be conducted. The report documents the analyses to determine where protection features are necessary to mitigate the consequence of a pipe break. Pipe break events involving high-energy fluid systems are analyzed for the effects of pipe whip, jet impingement, flooding, room pressurization, and temperature effects. Pipe break events involving moderate-energy fluid systems are analyzed for wetting from spray, flooding, and other environmental effects, as appropriate.	An as-designed pipe rupture hazard analysis report exists and concludes that the analysis performed for high and moderate energy piping confirms the protection of systems, structures, and components required to be functional during and following a design basis event.

Table 3.8-1 Roller Compacted Concrete ITAAC

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
The RCC Bridging Mat is seismic Category I and is designed and constructed to bridge over the design basis karst feature when subjected to design basis loads as specified in the Design Description in FSAR Subsection 2.5.4.5.4 without loss of structural integrity and the safety related functions.	i) An inspection of the bridging mat placement will be performed. Deviations in the RCC Bridging Mat properties due to as-built conditions that fall outside the range considered in the design as described in FSAR Subsection 2.5.4.5.4 will be analyzed for the design basis karst feature when subjected to design basis loads. ii) An inspection of the RCC mix and bedding mix constituents will be performed in accordance with FSAR Subsection 3.8.5.11.4. Deviations from the design constituents will be evaluated against the range of properties established for these materials during the design phase. iii) An inspection of the as-built RCC thickness will be performed.	i) A report exists which reconciles deviations from design and placement process of the RCC during construction and concludes that the as-built RCC bridging mat conforms to the approved design and will bridge over a design basis karst feature when subjected to design basis loads specified in the Design Description without loss of structural integrity and the safety related functions. ii) A report exists which reconciles deviations in mix constituents used in construction and concludes that the as-built RCC conforms to the design requirements for these properties. iii) A document exists that verifies that the as-built thickness of the RCC bridging mat is at least as thick as the design requirement.

Table 3.8-2 Drilled Shaft Foundation ITAAC

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Drilled Shaft Foundations for the Turbine, Radwaste, and Annex Buildings will preclude movement of the building foundations in excess of the separation provided between the structural elements of the Turbine, Radwaste, and Annex Buildings and the nuclear island structures.	During construction, inspection of the physical properties of the rock socket for each drilled shaft will be performed in accordance with LNP FSAR Chapter 3 Subsection 3.8.5.9. Inspection of the as-built drilled shaft foundation physical arrangement will also be performed.	A report exists that reconciles the during construction physical properties of the rock socket for each drilled shaft and the as-built physical arrangement of the Turbine, Radwaste, and Annex Buildings' drilled shaft foundations with design specifications and drawings. The report concludes that the as-built drilled shaft foundation conforms to the design commitment.

Table 3.8-3 Waterproof Membrane ITAAC

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
The friction coefficient to resist sliding is ≥ 0.55.	Testing will be performed to confirm that the mudmat-waterproofing-RCC interface beneath the Nuclear Island basemat has a coefficient of friction to resist sliding of ≥ 0.55.	A report exists and documents that the as-built waterproof system mudmatwaterproofing-RCC interface) has a coefficient of friction of ≥ 0.55 as demonstrated through material qualification testing.

Table 3.12-1 Piping Design ITAAC

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
The ASME Code Section III piping is designed in accordance with the ASME Code Section III requirements.	Inspection of the ASME Code Design Reports (NCA-3550) and required documents will be conducted for the set of lines chosen to demonstrate compliance.	The ASME Code Design Report(s) (NCA-3550) (certified, when required by the ASME Code) exist and conclude that the design of the piping for lines chosen to demonstrate all aspects of the piping design complies with the requirements of the ASME Code Section III.

4.0 REACTOR

4.1 Introduction

This chapter describes the design 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, the thermal-hydraulic design, and the reactivity control systems functional 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 Levy Nuclear Plant (LNP) combined license (COL) Final Safety Analysis Report (FSAR), Revision 9, incorporates by reference Chapter 4 of the AP1000 Design Control Document (DCD), Revision 19.

In addition, in LNP 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 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 for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition).”

To resolve the confirmatory item, the U.S. 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 LNP 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 contained 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 and reactivity control systems functional design. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 4.4 of the VEGP SER:

[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).

<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 pending formal revision to the VEGP FSAR. The staff verified that the VEGP COL FSAR was appropriately revised. As a result, Confirmatory Item 4.4-1 is now closed.

The LNP applicant has endorsed RAI 1-7 and RAI 1-11 and has also endorsed the January 8, 2010, letter submitted by the VEGP applicant, but needed to revise Appendix 1AA of its FSAR to be consistent with the VEGP response. This issue was being tracked as LNP Confirmatory Item 4.4-1. The staff verified that the LNP COL FSAR was appropriately revised. As a result, LNP 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 proposes to include the following license condition related to instrumentation uncertainties:

- 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 application 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, the thermal-hydraulic design, and reactivity control systems functional design, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this chapter. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented within the LNP 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.

5.0 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, 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 (SGs) for plant power generation.

Section 5.1 of the Levy Nuclear Plant (LNP) combined license (COL) Final Safety Analysis Report (FSAR), Revision 9, 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 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 LNP 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* (CFR) 10 CFR 50.55a, "Codes and Standards," incorporates by reference the American Society of Mechanical Engineers (ASME) *Boiler & Pressure Vessel Code* (BPV Code) and 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 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.

[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).

5.2.1.1.2 Summary of Application

Section 5.2 of the LNP COL FSAR, Revision 9, incorporates by reference Section 5.2 of the AP1000 DCD, Revision 19. Section 5.2 of the DCD includes Section 5.2.1.1.

In addition, in LNP COL FSAR Section 5.2.1.1, the applicant provided the following:

AP1000 COL Information Item

- STD COL 5.2-1

The applicant provided additional information in Standard (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 Section 5.2.6.1, “ASME Code and Addenda,” of the AP1000 DCD. The portion of STD COL 5.2-1 evaluated here applies to ASME BPV Code reconciliation. The portion applicable to Code cases is reviewed in Section 5.2.1.2 of this safety evaluation report (SER).

In particular, Revision 9 to LNP COL FSAR in 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 DC 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.

The requirements for alternatives to codes and standards that previously appeared in the 10 CFR 50.55a(a)(3) regulation cited in the FSAR, now appear in 10 CFR 50.55a(z) following a 2014 rulemaking.

5.2.1.1.3 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 the Commission regulations for the ASME BPV Code reconciliation are given in Section 5.2.1 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 NRC 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, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," 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 NRC staff reviewed Section 5.2.1.1 of the LNP COL FSAR and 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 integrity of the RCPB. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 Section 5.2.1.1.4 of the VEGP SER:

> *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(z)] *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(z)]*. This meets the regulations and is, therefore, acceptable.*

Correction to the Standard Content Evaluation Text

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 LNP, BLN or VEGP evaluations.

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

In addition, the staff concludes that the relevant information presented in the LNP 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 BPV 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 and 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 BPV Code must be authorized by the NRC pursuant to 10 CFR 50.55a(z).

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 LNP. 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 BPV Standards Committee. ASME Code cases acceptable to the NRC 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 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

Section 5.2 of the LNP COL FSAR, Revision 9, incorporates by reference Section 5.2 of the AP1000 DCD, Revision 19. Section 5.2 of the DCD includes Section 5.2.1.2.

LNP COL FSAR Section 5.2 does not include supplemental information in the incorporation by reference of Section 5.2.1.2 of the AP1000 DCD. However, LNP COL FSAR Section 5.2 specifies supplementary information in STD COL 5.2-1 that relates to applicable Code cases.

In addition, in LNP COL FSAR Section 5.2.1.1, the applicant provided the following:

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 Section 5.2.6.1, "ASME Code and Addenda," of the AP1000 DCD. The portion of STD COL 5.2-1 evaluated here applies to applicable Code cases.

5.2.1.2.3 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 the Commission regulations for the applicable Code cases are given in Section 5.2.1.2 of NUREG-0800.

The applicable regulatory requirements for the NRC staff's review of the LNP COL application are as follows.

GDC 1 in Appendix A to 10 CFR Part 50 and 10 CFR 50.55a, as related to the establishment of the minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of nuclear power plant components, require conformance with appropriate editions of published industry codes and standards.

As one means of meeting the applicable NRC regulations, RG 1.84 lists ASME BPV 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 BPV 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 NRC staff reviewed Section 5.2 of the LNP COL FSAR and 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 applicable Code cases. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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 Table 5.2-3 of Revision 19 to the AP1000 DCD.

The NRC 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 LNP COL FSAR Section 5.2.1.2 for compliance with the NRC regulations.

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 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 5.2.1.2.4 of the VEGP SER:

AP1000 COL Information Item

- *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(z)], *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(z)]*.*

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.

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

In addition, the staff concludes that the relevant information presented in the LNP 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 NRC staff has determined that LNP 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 LNP 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).

Section 5.2 of the LNP COL FSAR, Revision 9, incorporates by reference, with no departures or supplements, Section 5.2.1.3, “Alternate Classification,” 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 LNP 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).

Section 5.2 of the LNP COL FSAR, Revision 9, incorporates by reference, with no departures or supplements, Section 5.2.2, “Overpressure Protection,” 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 LNP 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*

Section 5.2 of the LNP COL FSAR, Revision 9, incorporates by reference Section 5.2 of the AP1000 DCD, Revision 19. Section 5.2 of the DCD includes Section 5.2.3.

In addition, in LNP COL FSAR Section 5.2.3.2.1, the applicant provided the following:

Supplemental Information

- STD SUP 5.2-1

The applicant provided supplemental (SUP) 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 the Commission regulations for the RCPB materials are given in Section 5.2.3 of NUREG-0800.

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 NRC staff reviewed Section 5.2 of the LNP 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 RCPB materials. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 5.2.3.4 of the VEGP SER:

> <u>*Supplemental Information*</u>
>
> - *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.*
>
> *Chemistry procedures will provide guidance for the sampling and monitoring of primary coolant properties.*

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.

5.2.3.5 *Post Combined License Activities*

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

5.2.3.6 *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 RCPB materials, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP 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 Section 9.3.4 of NUREG-0800 with respect to developing a water chemistry program consistent with the latest EPRI guidelines and is acceptable. Conformance with these guidelines provides an acceptable basis for satisfying, 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 leak tight 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 BPV Code, meet the applicable inspection requirements set forth in Section XI of the ASME Code, "Rules for Inservice Inspection of Nuclear Power Plant Components."

5.2.4.2 *Summary of Application*

Section 5.2 of the LNP COL FSAR, Revision 9, incorporates by reference Section 5.2 of the AP1000 DCD, Revision 19. Section 5.2 of the DCD includes Section 5.2.4.

In addition, in LNP COL FSAR Section 5.2.4, the applicant provided the following:

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

In a letter dated August 27, 2010, the VEGP applicant proposed a new STD COL 5.3-7 to address AP1000 DCD COL Information Item 5.3-7 included in a Westinghouse letter dated August 3, 2010. The new information states that the COL holder will augment the plant-specific ISI program in VEGP COL FSAR Section 5.2.4.1, related to the Quickloc weld buildup on the RV vessel head. In its letter dated March 1, 2011, the LNP applicant endorsed that VEGP letter as standard, thereby adopting STD COL 5.3-7 for the LNP COL application. The LNP COL FSAR was appropriately revised.

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

- License Condition 6, regarding PSI/ISI program details

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 the Commission regulations for ISI are given in Section 5.2.4 of NUREG-0800.

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 Appendix A to 10 CFR Part 50, 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 Items 5.2-2 and 5.3-7, 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," dated October 28, 2005.

5.2.4.4 *Technical Evaluation*

The NRC staff reviewed Section 5.2.4 of the LNP COL FSAR and 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 RCPB ISI and testing. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In Section 5.2.4 of NUREG-1793, 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 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 remains 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 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 (VEGP Units 3 and 4) were equally applicable to the LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 5.2.4.4 of the VEGP SER:

> *<u>AP1000 COL Information Item</u>*
>
> *The following portion of this technical evaluation section is reproduced from Section 5.2.4.4 of the BLN SER:*
>
> - *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 incore 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:

> *<u>5.2.4.8 Relief Requests</u>*
>
> *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(z)] *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:

> *5.2.4.9 Preservice Inspection of Class 1 Components*
>
> *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*

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:

> *5.2.4.10 Program Implementation*
>
> *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.

<u>Resolution of Standard Content Confirmatory Item 5.2-1</u>

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 VEGPCOL FSAR was appropriately updated to incorporate reference to10 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(z)]*. 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**.*

<u>Resolution of Standard Content Confirmatory Item 5.2-2</u>

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:

<u>License Condition</u>

- *License Condition 6, regarding PSI/ISI program details*

The BLN COL FSAR addresses implementation milestones for the PSI/ISI programs in Part 10, or 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.*

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

The staff concludes that the relevant information presented in the LNP COL FSAR meets the relevant acceptance criteria provided in Section 5.2.4 of NUREG-0800, 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, relating 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 proposed resolution to the COL information items is 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.

- STD SUP 5.2-2, relating 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 Generic Letter (GL) 88-05.

- STD COL 5.3-7, relating to the ISI program augmentation to include 100 percent volumetric examination of the weld build-up on the RV head for the Quickloc penetrations ensures that the integrity of the RCPB boundary weld will be maintained. Therefore, the staff finds that the applicant's proposed resolution as stated in their letter, dated March 1, 2011, to COL Information Item 5.3-7 is acceptable on the basis that it meets GDC 32 of Appendix A to 10 CFR Part 50, 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*

Section 5.2 of the LNP COL FSAR, Revision 9, incorporates by reference Section 5.2.5 of Revision 19 of the AP1000 DCD.

In addition, the applicant proposed the following:

AP1000 COL Information Item

- STD COL 5.2-3

In a letter, dated August 5, 2010, the applicant for the reference COL VEGP Units 3 and 4 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 COL Information Item 5.2-3. The VEGP applicant provided additional information regarding an issue concerning low-level RCS leakage. In its letter dated March 1, 2011, the LNP applicant endorsed that VEGP letter as standard thereby adopting STD COL 5.2-3 for the LNP COL application.

5.2.5.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference 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."

5.2.5.4 *Technical Evaluation*

Section 5.2 of the LNP COL FSAR, Revision 9, incorporates by reference, with no departures or supplements, Section 5.2.5 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, with one exception. That exception is discussed in the standard content material below. 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.

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 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 5.2.5.4 of the VEGP SER:

> *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, Revision 18. 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.

<u>AP1000 COL Information Item</u>

- *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****.*

Resolution of Standard Content Confirmatory Item 5.2-3

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.

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 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 RCPB leakage detection, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The staff concludes that the relevant information presented in the LNP 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 with these guidelines provides an acceptable basis for satisfying 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.

Section 5.3 of the LNP COL FSAR, Revision 9, incorporates by reference, with no departures or supplements, Section 5.3.1 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 LNP COL application are documented in NUREG-1793 and its supplements.

5.3.2 Reactor Vessel Materials

5.3.2.1 *Introduction*

This section 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*

Section 5.3 of the LNP COL FSAR, Revision 9, incorporates by reference Section 5.3 of the AP1000 DCD, Revision 19. Section 5.3 of the DCD includes Section 5.3.2.

In addition, in LNP COL FSAR Section 5.3.2.6, the applicant provided the following:

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 Appendix F of NUREG-1793. The additional information discusses the RV material surveillance program.

License Conditions

- Part 10, License Condition 3.J.1, Reactor Vessel Material Surveillance

The licensee shall implement this operational program prior to initial criticality.

- Part 10, License Condition 6

The licensee shall provide an operational program schedule to support NRC inspections.

5.3.2.3 *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 RV materials are given in Section 5.3.1 of NUREG-0800.

The applicable regulatory requirements and guidance for acceptance of the COL information item are as follows:

1. GDC 32 found in Appendix A to 10 CFR Part 50, as it relates to the RVSP;
2. 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," as it relates to compliance with the requirements of 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements";
3. 10 CFR Part 50, Appendix G, as it relates to materials testing and acceptance criteria for fracture toughness;
4. 10 CFR 50.55a, as it relates to the requirements for testing and inspecting Code Class 1 components of the RCPB as specified in Section XI of the ASME Code;
5. SECY-05-0197, as it relates to fully describing an operational program; and
6. 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," as it relates to the RVSP.

5.3.2.4 *Technical Evaluation*

The NRC staff reviewed Section 5.3.2 of the LNP COL FSAR and 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 relevant information related to RV materials. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 5.3.2.4 of the VEGP SER:

> *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

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

The staff concludes that the relevant information presented in the LNP COL FSAR is acceptable and meets the relevant regulatory guidance provided in Section 5.3.1 of NUREG-0800 and RG 1.206, the policy established in SECY-05-0197, and the requirements of Appendices G and H to 10 CFR Part 50. 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 Section 5.3.1 of NUREG-0800 and in RG 1.206, Section C.III.1, Chapter 5, C.I.5.3.1. Conformance with these guidelines provides an acceptable basis for satisfying, in part, the requirements of Appendices G and H to 10 CFR Part 50.

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 <u>Introduction</u>

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 Appendix G of Section XI of the ASME Code 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*

Section 5.3 of the LNP COL FSAR, Revision 9, incorporates by reference Section 5.3 of the AP1000 DCD, Revision 19. Section 5.3 of the AP1000 DCD includes Section 5.3.3.

In addition, in LNP COL FSAR Section 5.3.6.1, the applicant provided the following:

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.

License Condition

- 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 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 P-T limits are given in Section 5.3.2 of NUREG-0800.

5.3.3.4 *Technical Evaluation*

The NRC staff reviewed Section 5.3.3 of the LNP COL FSAR and 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 P-T limits. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 5.3.3.4 of the VEGP SER:

> <u>*AP1000 COL Information Item*</u>
>
> - *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.

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 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 NRC staff reviewed the application and the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to P-T limits, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP COL FSAR is acceptable and meets the regulatory basis addressed in NUREG-1793. Specifically, the relevant regulatory basis includes Section 5.3.2 of NUREG-0800; GL 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits"; and Appendix G to 10 CFR Part 50. 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 Section 5.3.2 of NUREG-0800. Conformance with these guidelines provides an acceptable basis for satisfying in part, the requirements of Appendix G to 10 CFR Part 50.

- 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*

Section 5.3.4 of the AP1000 DCD describes the RV integrity. The RV is the RCPB used to support and enclose the reactor core. It 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, this section 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*

Section 5.3 of the LNP COL FSAR, Revision 9, incorporates by reference Section 5.3 of the AP1000 DCD, Revision 19. Section 5.3 of the DCD includes Section 5.3.4.

In addition, in LNP COL FSAR Section 5.3.6, the applicant provided the following:

AP1000 COL Information Item

- 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 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

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 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 RV integrity are given in Section 5.3.3 of NUREG-0800.

In addressing the COL information item, PWRs are required, in part, to have the pressurized thermal shock reference temperature, 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 NRC staff reviewed Section 5.3.4 of the LNP 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 RV integrity. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 5.3.4.3 of the VEGP SER:

> *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.

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 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 RV integrity, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP COL FSAR meets the relevant acceptance criteria provided in Section 5.3.3 of NUREG-0800, and the requirements of Appendix B to 10 CFR Part 50 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 Section 5.3.3 of NUREG-0800 and thus provides an acceptable basis for satisfying, in part, the requirements of Appendix B to 10 CFR Part 50 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.

Section 5.3 of the LNP COL FSAR, Revision 9, incorporates by reference, with no departures or supplements, Section 5.3.5 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 LNP 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:

- RCPs
- SGs, including materials and ISI
- RCS piping and valves
- Main steam line flow restriction
- Pressurizer and pressurizer relief discharge
- Automatic depressurization system valves
- RNS
- 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 DCD is incorporated by reference in the 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.

5.4.2 Summary of Application

Section 5.4 of the LNP COL FSAR, Revision 9, incorporates by reference Section 5.4 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Section 5.4, the applicant provided the following:

Departures

- LNP DEP 3.2-1 and LNP DEP 6.3-1

The applicant provided additional information in Section 5.4 of the LNP COL FSAR about LNP DEP 3.2-1 and LNP 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 LNP DEP 3.2-1 and LNP DEP 6.3-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.1 of this SER.

<u>AP1000 COL Information Item</u>

- 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 Section 5.4.15 of the AP1000 DCD. The information in STD COL 5.4-1 provides the SG program description, references the applicable ASME BPV 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 LNP 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 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 component and subsystem design are given in Section 5.4.2 of NUREG-0800.

The applicable regulatory requirements for acceptance of the COL information item are 10 CFR 50.55a, as it relates to periodic inspection and testing of the RCPB as detailed in Section XI of the ASME Code, and 10 CFR Part 50, Appendix A, GDC 32, 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 Article IWB-2000 of the ASME Code, Section XI, then the SG tube inspection requirements are governed by the TS.

5.4.4 Technical Evaluation

The NRC staff reviewed Section 5.4 of the LNP COL FSAR and 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 RCS component and subsystem design. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 5.4.4 of the VEGP SER:

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.

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

In addition, the staff concludes that the relevant information presented in the LNP COL FSAR is acceptable and meets the relevant regulatory requirements provided in Appendix A to 10 CFR Part 50, GDC 32 and 10 CFR 50.55a, and the regulatory guidance addressed in RG 1.206 and RG 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes." The staff based its conclusion on the following:

- LNP DEP 3.2-1 and LNP 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 Appendix A to 10 CFR Part 50, GDC 32, and 10 CFR 50.55a including the specific modification provided in 10 CFR 50.55a(b)(2)(iii).

6.0 ENGINEERED SAFETY FEATURES

6.0 Engineered Safety Features

Engineered safety features (ESF) protect the public in the event of an accidental release of radioactive fission products from the reactor coolant system (RCS). The ESF function is to localize, control, mitigate, and terminate such accidents, and to maintain radiation exposure levels to the public below applicable limits and guidelines.

Section 6.0 of the Levy Nuclear Plant (LNP) combined license (COL) Final Safety Analysis Report (FSAR), Revision 9, incorporates by reference, with no departures or supplements, Section 6.0, "Engineered Safety Features," 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 LNP 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 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 LNP COL FSAR Section 6.1 corresponds to two sections of the DCD, Section 6.1.1, "Metallic Materials"; and Section 6.1.2, "Organic Materials." The NRC staff's evaluation of these two FSAR sections is provided below.

6.1.1 Metallic Materials

6.1.1.1 *Introduction*

In this section, the NRC staff reviews metallic materials used in ESF components to ensure that they are compatible with one another and with ESF fluids. The compatibility of 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.

[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).

6.1.1.2 *Summary of Application*

Section 6.1 of the LNP COL FSAR, Revision 9, incorporates by reference Section 6.1 of the AP1000 DCD, Revision 19. Section 6.1 of the AP1000 DCD includes Section 6.1.1.

In addition, in LNP COL FSAR Section 6.1.1, the applicant provided the following:

AP1000 COL Information Item

- STD COL 6.1-1

The applicant provided additional information in Standard (STD) COL 6.1-1 to resolve COL Information Item 6.1-1. STD COL 6.1-1 describes quality assurance measures for special processes in fabricating austenitic stainless steels. In a letter dated April 7, 2010, the DCD applicant, Westinghouse, proposed to revise Appendix 1A of the AP1000 DCD to remove stated exceptions to conformance with Regulatory Guide (RG) 1.44, "Control of the Use of Sensitized Steel," Revision 0. The NRC staff's review of STD COL 6.1-1 includes the information in the Westinghouse letter. The COL applicant did not submit additional information in response to this proposed revision.

6.1.1.3 *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 the Commission regulations for the metallic materials are given in Section 6.1.1 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)."

The regulatory basis for the 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 for the COL information item is described in RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," Revision 3, and RG 1.44.

6.1.1.4 *Technical Evaluation*

The NRC staff reviewed Section 6.1.1 of the LNP 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 metallic materials. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 6.1.1.4 of the VEGP SER:

> *AP1000 COL Information Item*
>
> - *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.

The staff confirmed the AP1000 DCD incorporated by reference by the applicant was appropriately modified as described above.

6.1.1.5 *Post Combined License Activities*

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

6.1.1.6 *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 metallic materials used in the ESF, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP COL FSAR is acceptable because it meets the requirements of 10 CFR Part 50, Appendix B, and the guidance provided in RGs 1.31 and 1.44. The staff based its conclusion on the following:

- 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 RGs 1.31 and 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 considerations for protective coatings differ for these four areas and the coatings selection process accounts for these differing considerations. 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*

Section 6.1 of the LNP COL FSAR, Revision 9, incorporates by reference Section 6.1 of the AP1000 DCD, Revision 19. Section 6.1 of the AP1000 DCD includes Section 6.1.2.

In addition, in LNP COL FSAR Section 6.1.2, the applicant provided the following:

<u>*AP1000 COL Information Item*</u>

- STD COL 6.1-2

The applicant provided additional information in STD COL 6.1-2 to resolve 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. In a letter dated March 31, 2010, the DCD applicant, Westinghouse, proposed revisions to COL Information Item 6.1-2 in Section 6.1.3.2 of the AP1000 DCD to address Service Level II coatings. In letters dated September 23, 2010, and March 7, 2011, the applicant endorsed the VEGP letters dated July 2 and August 13, 2010, respectively, that proposed revising the FSAR to address the updated 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.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for protective coatings are given in Section 6.1.2 of NUREG-0800.

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 NRC staff reviewed Section 6.1.2 of the LNP 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 protective coatings and other organic materials inside containment. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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. Although the staff concluded that the evaluation performed for the standard content is directly applicable to the LNP 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 Section 6.1.2.4 of the VEGP SER:

> *AP1000 COL Information Item*
>
> - *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 finds the COL information related to control of coatings during the design and construction phase acceptable.

Subsequently, the staff finds the FSAR revisions proposed above 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 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 protective coatings and other organic materials inside containment, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP COL FSAR is acceptable and meets the requirements of 10 CFR Part 50, Appendix B, and the guidance provided in RG 1.54. The staff based its conclusion on the following:

- STD COL 6.1-2, relating to the coatings program, is acceptable because the Appendix B quality assurance program, with the additional guidance provided in RG 1.54, 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

Section 6.2 of the LNP COL FSAR, Revision 9, incorporates by reference Section 6.2 of the AP1000 DCD, Revision 19. Section 6.2 of the DCD 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.” DCD Section 6.2.5 is evaluated by the NRC staff in Section 6.2.6 of NUREG-1793. NUREG-1793 also includes the staff’s evaluation of the following issues:

- Fracture prevention of the containment pressure boundary in accordance with NUREG-0800, Section 6.2.7

- In-containment refueling water storage tank (IRWST) hydrodynamic loads

There are no COL information items associated with the review of either of these issues. The staff’s evaluation of the sections that address fracture prevention of the containment pressure boundary is found in Section 3.8 of this SER. With respect to the hydrodynamic loads, the staff’s evaluation may be found in Section 6.2.8 of NUREG-1793.

The staff’s evaluation of the containment cleanliness program associated with Generic Safety Issue (GSI)-191, “Assessment of Debris Accumulation on PWR Sump Performance,” is evaluated in Section 6.3 of this SER.

In addition, in LNP COL FSAR Section 6.2, and in Part 10 of the LNP COL application, the applicant provided the following:

Departures

- LNP DEP 6.2-1

The applicant provided additional information in Section 6.2.4.5.1 of the LNP COL FSAR about LNP 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 LNP 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 additional information in STD COL 6.2-1 to address COL Information Item 6.2-1 and COL Action Item 6.2.6-1, which addresses the containment leak rate test program. In addition, LNP 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 LNP COL FSAR Section 6.2.5.1, STD COL 6.2-1.

License Conditions

- Part 10, License Condition 3, Item G.8

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 LNP COL FSAR Table 13.4-201, "Operational Programs Required by NRC Regulations."

- Part 10, License Condition 6

This proposed license condition states that the COL holder shall provide an operational program implementation schedule to support NRC inspections.

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 the Commission regulations for containment functional design are given in Section 6.2.1.1A of NUREG-0800. The regulatory requirements related to this section are 10 CFR Part 50, Appendix A, General

Design Criteria (GDC) 16, “Containment Design”; GDC 38, “Containment Heat Removal”; and GDC 50, “Containment Design Basis.”

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for containment leak rate testing are given in Section 6.2.6 of NUREG-0800. 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.”

Conformance with the requirements of Option A of Appendix J, or the requirements of Option B of Appendix J 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 found in Nuclear Energy Institute (NEI) 94-01, “Industry Guideline for Implementing the Performance-Based Option of 10 CFR Part 50, Appendix J,” 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,” dated April 4, 2007, to review the operational program, Containment Leakage Rate Testing Program.

6.2.4 Technical Evaluation

The NRC staff reviewed Section 6.2 of the LNP 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 CSs. The results of the NRC staff’s evaluation of the information incorporated by reference in the LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 LNP COL FSAR.

The following portion of this technical evaluation section is reproduced from Section 6.2.4 of the VEGP SER:

> *<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.8*
> - *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 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 CSs, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP COL FSAR is acceptable and complies with the guidance in Sections 6.2.1 and 6.2.6 of NUREG-0800. The staff based its conclusion on the following:

- LNP 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 NRC staff has determined that the requirements of 10 CFR Part 50, Appendix J, have been met.

6.3 Passive Core Cooling System (Related to RG 1.206, Section C.III.1, Chapter 6, C.I.6.3, "Emergency 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. LNP 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

Section 6.3 of the LNP COL FSAR, Revision 9, incorporates by reference Section 6.3 of the AP1000 DCD, Revision 19. Section 6.3 of the DCD includes Section 6.3.2.2.7, "IRWST and Containment Recirculation Screens"; Section 6.3.8.1, "Containment Cleanliness Program"; and

Section 6.3.8.2, "Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA."

In addition, in LNP COL FSAR Section 6.3.8.1, the applicant provided the following:

Departures

- LNP DEP 3.2-1 and LNP DEP 6.3-1

The applicant provided additional information in Section 6.3 of the LNP COL FSAR about LNP DEP 3.2-1 and LNP 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 LNP DEP 3.2-1 and LNP 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 additional information in STD COL 6.3-1 to address 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 requires the applicant to develop a containment cleanliness program to limit the amount of debris that might be left in the containment following refueling and maintenance outages.

Section 1.9 of the LNP COL FSAR incorporates by reference Section 1.9, "Compliance with Regulatory Criteria," of the AP1000 DCD. Section 1.9 of the DCD includes Section 1.9.4.2.3, "New Generic Issues," and Section 1.9.5.5, "Operational Experience."

In addition, in LNP COL FSAR Section 1.9, the applicant provided the following information related to the effect of debris accumulation on long-term cooling:

- STD COL 1.9-3

The applicant provided additional information in STD COL 1.9-3 to address the review of GSI-191.

- STD COL 1.9-2

The applicant provided additional information in STD COL 1.9-2 to address the review of BL 03-01 and GL 04-02.

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 NRC 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 NEI 04-07, “Pressurized Water Reactor Sump Performance Evaluation Methodology,” Revision 0, Volume 1, and in the “Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02,” in NEI 04-07, Revision 0, Volume 2.

6.3.4 Technical Evaluation

The NRC staff reviewed Section 6.3 of the LNP 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 passive core cooling system. The results of the NRC staff’s evaluation of the information incorporated by reference in the LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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.

AP1000 COL Information Items

The following portion of this technical evaluation section is reproduced from Section 6.3.4 of the VEGP SER:

> *AP1000 COL Information Items*
>
> - *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 6.2.2-1, 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 6.2.2-1 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.

Resolution of Standard Content Confirmatory Item 6.3-1

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 6.2.2-1 (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 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 passive containment cleanliness program, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP COL FSAR is acceptable and meets the regulatory requirements and guidance discussed in Section 6.3.3 of this SER. The staff based its conclusion on the following:

- LNP DEP 3.2-1 and LNP 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 1.9-3, 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 SER Section 6.1.2, and the containment cleanliness program is evaluated under STD COL 6.3-1.

- 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 SER Section 6.1.2, and the containment cleanliness program is evaluated under STD COL 6.3-1.

6.4 Habitability Systems

6.4.1 Introduction

The design and operation of a set of systems provide habitability functions for the AP1000 design. These systems include the nuclear island non-radioactive ventilation system, the main control room (MCR) emergency habitability system (VES), the radiation monitoring system, the plant lighting system, and the fire protection system.

6.4.2 Summary of Application

Section 6.4 of the LNP COL FSAR, Revision 9, incorporates by reference Section 6.4 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Section 6.4, the applicant provided the following:

Departures

- LNP DEP 6.4-1

The applicant provided information about LNP 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 LNP DEP 6.4-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.2 of this report.

- LNP DEP 6.4-2

The applicant provided information about LNP 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 LNP 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 and LNP COL 9.4-1b

The applicant provided a list of onsite chemicals in LNP COL FSAR Table 6.4-201 to supplement the list of chemicals identified in Table 6.4-1 of the AP1000 DCD. The chemicals in Table 6.4-201 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 letter dated September 23, 2010, the applicant endorsed the June 17, 2010, letter from VEGP regarding the storage of standard chemicals described under STD COL 6.4-1. In a letter dated March 7, 2011, the LNP applicant endorsed letters from the VEGP applicant dated July 30, 2010, and September 3, 2010, that proposed modifications to the COL FSAR Table 6.4-201 related to the size and stated location of the liquid hydrogen storage tank and related to the impact evaluation notes.

- STD COL 6.4-2

The applicant provided additional information in STD COL 6.4-2 to address 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.”

- LNP COL 6.4-1

The applicant provided LNP COL 6.4-1 to address COL Information Item 6.4-1. The applicant provided information in the FSAR regarding the storage of plant-specific hazardous chemicals.

Supplemental Information

- STD SUP 6.4-1

The applicant provided supplemental information in STD SUP 6.4-1 to address CR doses for accident analyses in the downwind unit of a dual unit site.

- LNP SUP 6.4-1

The applicant provided supplemental information in LNP SUP 6.4-1 related to the use of gaseous and liquid dispersants.

6.4.3 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 habitability systems are given in Section 6.4 of NUREG-0800.

MCR habitability is addressed in the following regulations and guidance:

- 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, and the NRC's regulations.

- NUREG-0737, “Clarification of TMI Action Plan Requirements,” TMI Action Plan, Item III.D.3.4, “Control Room Habitability.”

- RG 1.78, “Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release,” Revision 1.

- RG 1.52, “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,” Revision 3, June 2001.

- RG 1.196, “Control Room Habitability at Light Water Nuclear Power Reactors,” May 2003.

6.4.4 Technical Evaluation

The NRC staff reviewed Section 6.4 of the LNP 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 habitability systems. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR, except for the evaluation of STD SUP 6.4-2 and STD COL 6.4-1. For these two items, the staff compared the BLN COL FSAR, Revision 2 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 LNP COL FSAR.

AP1000 COL Information Items

The following portion of this technical evaluation section is reproduced from Section 6.4.4 of the VEGP SER:

- *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.

<u>AP1000 COL Information Items</u>

- *STD COL 6.4-1* [and LNP COL 9.4-1b]

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****.*

<u>Resolution of Standard Content Open Item 6.4-1</u>

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****.*

<u>Resolution of Standard Content Confirmatory Item 6.4-1</u>

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****.*

<u>Resolution of Standard Content Confirmatory Item 6.4-2</u>

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****.*

Resolution of Standard Content Confirmatory Item 6.4-3

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.

- LNP COL 6.4-1

The NRC staff reviewed LNP COL 6.4-1, related to COL Information Item 6.4-1, included under Section 6.4.4 of the LNP COL FSAR. LNP COL 6.4-1, including Table 6.4-201, indicated that there were no site-specific sources of hazardous chemicals that could impact the MCR. This is acceptable because it results in no impact to MCR habitability.

Supplemental Information

The following portion of this technical evaluation section is reproduced from Section 6.4.4 of the VEGP SER:

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 determined that the omitted portion was not relevant to LNP.

- LNP SUP 6.4-1

The NRC staff reviewed LNP SUP 6.4-1, related to COL Information Item 6.4 1, included under Section 6.4.4 of the LNP COL FSAR. The applicant indicated that the site does not plan to use any gas dispersants. The applicant plans to use liquid dispersants, but did not propose any for onsite use and storage. This is acceptable because, since no dispersants are proposed for onsite use and storage, there is no impact on MCR habitability. If dispersants are identified for use in the future, the applicant would evaluate potential impacts on MCR habitability at that time.

6.4.5 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:

- 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

6.4.6 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 MCR habitability, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the information presented in the LNP COL FSAR is acceptable because it meets the acceptance criteria associated with the relevant requirements of the Commission regulations for habitability systems given in Section 6.4 of NUREG-0800. The staff based its conclusions on the following:

- LNP DEP 6.4-1 and LNP COL 9.4-1b, 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.

- LNP 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 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.

- LNP COL 6.4-1 is acceptable because there are no plant-specific chemicals having the potential to 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 Section 6.4 of NUREG-0800.

- LNP SUP 6.4-1 is acceptable because no dispersants are proposed for onsite use and storage. There is no impact on MCR habitability. If dispersants are identified for use in the future, the applicant would evaluate potential impacts on MCR habitability at that time.

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.

Section 6.5 of the LNP COL FSAR, Revision 9, incorporates by reference, with no departures or supplements, Section 6.5 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 LNP COL application are documented in NUREG-1793 and its supplements.

6.6 Inservice Inspection of Class 2, 3, and MC Components (Related to RG 1.206, Section C.III.1, Chapter 6, C.I.6.6, "Inservice Inspection of Class 2 and 3 Components")

6.6.1 Introduction

Inservice inspection (ISI) programs must meet requirements of 10 CFR 50.55a, "Codes and Standards," in which Section XI of the ASME Boiler and Pressure Vessel Code (ASME Code) 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 Subsections IWC and IWD of Section XI of the ASME Code, "Rules for Inservice Inspection (ISI) of Nuclear Power Plant Components." 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

Section 6.6 of the LNP COL FSAR, Revision 9, incorporates by reference Section 6.6 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Section 6.6, the applicant provided the following:

AP1000 COL Information Items

- STD COL 6.6-1

The applicant provided additional information in STD COL 6.6-1 to address COL Information Item 6.6-1. The information relates to plant-specific preservice inspection (PSI) and ISI programs.

- STD COL 6.6-2

The applicant provided additional information in STD COL 6.6-2 to address COL Information Item 6.6-2. The information relates to preservation of component accessibility design considerations during the construction phase.

Supplemental Information

- STD SUP 6.6-1

The applicant provided supplemental information related to the design stage consideration of component accessibility to enable the performance of ISI examinations.

License Condition

- Part 10, License Condition 6

This proposed license condition states that the COL holder shall provide an operational (PSI/ISI) program schedule to support NRC inspections.

6.6.3 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 ISI of Class 2 and 3 components are given in Section 6.6 of NUREG-0800.

The applicable regulatory requirements for acceptance of the resolution of COL information items and supplementary information on ISI and testing of Class 2 and 3 components are established in GDC 45, "Inspection of Cooling Water System" found 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 found in SECY-05-0197.

6.6.4 Technical Evaluation

The NRC staff reviewed Section 6.6 of the LNP 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 ISI of Class 2 and 3 components. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 6.6.4 of the VEGP SER:

AP1000 COL Information Items

- *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 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.

License Condition

- *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 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 ISI of ASME Code Class 2 and 3 components, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP 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 MC (metal containment) 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).

7.0 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 <u>Introduction</u>

7.1.1 Introduction

Westinghouse Electric Company (Westinghouse) proposed to revise the AP1000 Design Control Document (DCD) to address final setpoint calculations for protective functions. These proposed changes to the DCD impact the AP1000 combined license (COL) applications.

7.1.2 Summary of Application

Section 7.1 of the Levy Nuclear Plant (LNP) COL Final Safety Analysis Report (FSAR) Revision 9, incorporates by reference Section 7.1 of the AP1000 DCD, Revision 19.

In addition, the applicant proposed the following:

<u>AP1000 COL Information Item</u>

- STD COL 7.1-1

In a letter dated March 8, 2010, Westinghouse proposed to revise the AP1000 DCD by adding COL Information Item 7.1-1 to address final setpoint calculations. In a letter dated September 23, 2010, the applicant proposed a revision to the LNP COL FSAR by adding Standard (STD) COL 7.1-1, “Setpoint Calculations for Protective Functions” to reflect the above.

7.1.3 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 Instrumentation and Controls are in Section 7.1 of NUREG-0800, “Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants.”

The applicable regulatory requirements for the information being reviewed in this section are:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36, “Technical specifications”
- 10 CFR 52.79(a)(30)

7.1.4 Technical Evaluation

The Nuclear Regulatory Commission (NRC) staff reviewed Section 7.1 of the LNP 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 safety-related display information. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 Section 7.1.4 of the VEGP SER:

> *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 16.1-1. 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 is being tracked as* ***Confirmatory Item 16.1-1****. The closure of this Confirmatory Item is provided in SER Section 16.1*

[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).

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 FSAR Table 1.8-202 and Section 7.1 to address COL Information Item STD COL 7.1-1. The staff verified that the VEGP COL FSAR was appropriately revised. 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 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 setpoint calculations for protective functions, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff has compared the application to the relevant NRC regulations and other NRC regulatory guides and concludes that, the applicant is in compliance with the NRC regulations. The staff based its conclusion on the following:

- 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 <u>Reactor Trip</u>

Section 7.2 of the LNP COL FSAR, Revision 9, incorporates by reference, Section 7.2, "Reactor Trip," of Revision 19 of the AP1000 DCD. In addition, in the LNP COL FSAR, the applicant provided the following:

<u>Departures</u>

- LNP DEP 6.4-2

The applicant provided additional information in Section 7.2 of the LNP COL FSAR about LNP 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 LNP DEP 6.4-2 information appearing in other chapters of the FSAR, is reviewed in Section 21.3 of this SER.

- LNP DEP 7.3-1

The applicant provided additional information in Table 7.2-201 of the LNP COL FSAR about LNP 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 LNP 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 LNP 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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

7.3 Engineered Safety Features

Section 7.3 of the LNP COL FSAR, Revision 9, incorporates by reference, Section 7.3, "Engineered Safety Features," of Revision 19 of the AP1000 DCD. In addition, in the LNP COL FSAR, the applicant provided the following:

Departures

- LNP DEP 6.4-1

The applicant provided additional information in Section 7.3.1.2.17 of the LNP COL FSAR about LNP 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 LNP DEP 6.4-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.2 of this SER.

- LNP DEP 6.4-2

The applicant provided additional information in Section 7.3.1.2.17 of the LNP COL FSAR about LNP 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 LNP DEP 6.4-2 information appearing in other chapters of the FSAR, is reviewed in Section 21.3 of this SER.

- LNP DEP 7.3-1

The applicant provided additional information in Section 7.3.1.2.14 of the LNP COL FSAR about LNP 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 LNP 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 LNP 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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In RAI 1-4, issued to the applicant for the BLN, Units 3 and 4, the staff questioned how the applicant would verify that the as-built I&C system configuration conformed to schematics. In its response to RAI 1-4, 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 1-4 was endorsed as standard for LNP by the applicant in its letter dated December 15, 2008.

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 1-4 and the applicant endorsement of that response acceptable.

7.4 Systems Required for Safe Shutdown

Section 7.4 of the LNP COL FSAR, Revision 9, incorporates by reference Section 7.4, "Systems Required for Safe Shutdown," of Revision 19 of the AP1000 DCD. In addition, in the LNP COL FSAR, the applicant provided the following:

Departures

- LNP DEP 6.3-1 and LNP DEP 3.2-1

The applicant provided additional information about LNP DEP 6.3-1 and LNP DEP 3.2-1 in Section 7.4.1.1 of the FSAR 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 LNP DEP 6.3-1 and LNP 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 7.4.1.1 of the LNP 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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.
Section 21.1 of this report evaluates the departures from the DCD provided in LNP DEP 6.3-1 and LNP DEP 3.2-1.

7.5 Safety-Related Display Information (Related to RG 1.206, Section C.III.1, Chapter 7, C.1.7.5, "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.

7.5.2 Summary of Application

Section 7.5 of the LNP COL FSAR, Revision 9, incorporates by reference Section 7.5 of the AP1000 DCD, Revision 19. In a letter dated May 26, 2010, in response to DCD Open Item OI-SRP7.5-ICE-01, Westinghouse proposed to revise the AP1000 DCD adding COL Information Item 7.5-1 for site-specific post accident monitoring variables. Westinghouse created a new COL Information Item (COL 7.5-1), which was incorporated into Revision 18 of the DCD.

In addition, in LNP COL FSAR Section 7.5, the applicant provided the following:

Departure

- LNP DEP 6.4-2

The applicant provided additional information in Section 7.5 of the LNP COL FSAR about LNP 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 LNP DEP 6.4-2 information appearing in other chapters of the FSAR, is reviewed in Section 21.3 of this SER.

AP1000 Information Items

- LNP COL 7.5-1 and STD COL 7.5-1

The applicant provided additional information in LNP COL FSAR Section 7.5, “Safety-Related Display Information,” describing the FSAR Table 7.5-201 supplement to DCD Table 7.5-1 and providing variable data shown in the DCD table as “site specific.”

The applicant also provided additional information in LNP COL FSAR Section 7.5, describing the FSAR Table 7.5-202 supplement to DCD Table 7.5-8 and providing variable data shown in DCD Table 7.5-8 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 Section 7.5 of NUREG-0800.

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 NRC staff reviewed Section 7.5 of the LNP 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 safety-related display information. The results of the NRC staff’s evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the LNP COL FSAR:

AP1000 Information Items

- LNP 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.

Appendix 1AA of the LNP COL FSAR took exception to Revision 4 of RG 1.97. The applicant, instead, stated conformance to Revision 3 of RG 1.97. 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 staff discusses the acceptability of Revision 3 of RG 1.97 in Section 12.1 of this SER.

Revision 3 of RG 1.97 states that the variable and range information should be provided for environs radiation and radioactivity, and meteorological instrumentation.

The staff issued RAI 7.5-1 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 applicant provided a supplemental response to RAI 7.5-1 with sufficient meteorological range and accuracy information for wind direction, wind speed, and differential temperature. In addition, the revised LNP COL FSAR Table 7.5-201 included the boundary environs radiation variable and the required range information for the post-accident monitoring system. The supplemental information conforms to the guidance of Revision 3 of RG 1.97. The staff confirmed that the LNP COL FSAR, Revision 5, incorporates the instrumentation supplemental information. The staff finds the response acceptable and considers RAI 7.5-1 closed.

In a letter dated May 26, 2010, WEC 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 LNP as part of LNP SUP 7.5-1 and incorporated in the LNP COL FSAR, the identification of COL Information Item 7.5-1 in the DCD required that the applicant address this information with a COL identifier rather than as supplemental information. Accordingly, the applicant's letter dated September 23, 2010, proposes to replace LNP SUP 7.5-1 with STD COL 7.5-1 (for standard information) and LNP COL 7.5-1 (for LNP-specific information). This change of identifiers does not impact the staff's conclusion regarding the instrumentation information added to the LNP COL FSAR. The incorporation of the changed identifiers into the LNP COL FSAR is **Confirmatory Item 7.5-1**.

Resolution for Confirmatory Item 7.5-1

Confirmatory Item 7.5-1 is an applicant commitment to replace LNP SUP 7.5-1 with STD COL 7.5-1 (for standard information) and LNP COL 7.5-1 (for LNP-specific information). The staff verified that the LNP COL FSAR was appropriately revised. As a result, Confirmatory Item 7.5-1 is now closed.

7.5.5 Post Combined License Activities

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

7.5.6 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 safety-related display information, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff has compared the application to the relevant NRC regulations and other NRC RGs and concludes that the applicant is in compliance with the NRC regulations. The applicant has satisfactorily addressed the guidance of Revision 3 of RG 1.97 through the response to RAI 7.5-1. The staff based its conclusion on the following:

- LNP 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.

- LNP COL 7.5-1 and STD COL 7.5-1 provided sufficient information regarding the safety-related display information, and is, therefore, 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

Section 7.6 of the LNP COL FSAR, Revision 9, incorporates by reference, with no departures or supplements, Section 7.6, "Interlock Systems Important to Safety," 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 LNP 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")

Section 7.7 of the LNP COL FSAR, Revision 9, incorporates by reference, with no departures or supplements, Section 7.7, "Control and Instrumentation 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 LNP COL application are documented in NUREG-1793 and its supplements.

8.0 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 at the Levy County Nuclear Plant (LNP). This chapter provides information on the functional adequacy of the offsite electric 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 NRC staff (the staff) of the LNP Combined License (COL) Final Safety Analysis Report (FSAR), Part 2 of the COL application (COLA), submitted by Progress Energy Florida (PEF), the COL applicant (the applicant).[1]

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 AP1000 design certification (DC).

8.1.2 Summary of Application

Section 8.1 of the LNP COL FSAR, Revision 9, incorporates by reference Section 8.1 of the AP1000 Design Control Document (DCD), Revision 19.

In addition, in LNP COL FSAR Section 8.1, the applicant provided the following:

Supplemental Information

- LNP SUP 8.1-1

The applicant provided supplemental (SUP) information in LNP COL FSAR Section 8.1, "Introduction," describing LNP's connections to PEF electrical grid and the connection interfaces with neighboring utilities via the LNP, Units 1 and 2, 500-kilovolt (kV)/230-kV switchyard at the LNP site.

---

[1] The applicant, Duke Energy Florida, LLC, was formerly identified as Duke Energy Florida, Inc., and Progress Energy Florida, Inc. In a letter dated April 15, 2013, Progress Energy Florida notified the NRC that its name was changing to Duke Energy Florida, Inc., effective April 29, 2013. The name changes and a 2012 corporate merger between Duke Energy and Progress Energy are described in Chapter 1 of the SER. Because a portion of the review described in this chapter was completed prior to the name change, the NRC staff did not change references to "Progress Energy Florida" or "PEF" to "Duke Energy Florida" or "DEF" in this chapter.

- LNP SUP 8.1-2

The applicant provided supplemental information in LNP COL FSAR Section 8.1 describing the function and connection of the reserve auxiliary transformers (RATs) A and B for LNP Units 1 and 2.

- LNP SUP 8.1-3

The applicant provided supplemental information in LNP 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 LNP COL FSAR Table 8.1-201.

8.1.3 Regulatory Basis

The regulatory basis for 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 the introduction to the electric power systems are given in Section 8.1 of NUREG-0800, “Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR [light-water reactor] Edition).”

The applicable regulatory requirements, guidelines, and related acceptance criteria for the supplemental information items are as follows:

- Section 50.63, “Loss of All Alternating Current Power,” of Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.63)

- RG 1.155, “Station Blackout”

- RG 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition)”

8.1.4 Technical Evaluation

The NRC staff reviewed Section 8.1 of the LNP 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.[2] The NRC 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 NRC staff’s evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

[2] 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 reviewed the following information in the LNP COL FSAR:

Supplemental Information

- LNP SUP 8.1-1

The staff reviewed the supplemental information related to the PEF utility grid and its connection to neighboring utilities included under LNP SUP 8.1-1. The applicant's supplement to Section 8.1.1 is summarized as follows:

The PEF electrical grid consists of nuclear and fossil fuel generating facilities and an extensive 500-kV/230-kV bulk power transmission system. PEF maintains multiple direct interconnections with neighboring utilities. These interconnections serve to increase the reliability of the PEF electrical grid.

LNP Units 1 and 2 are connected to a new common switchyard having dual voltages 500-kV and 230-kV. The switchyard also serves as units' preferred and maintenance source. The switchyard has both breaker-and-a-half and double breaker schemes. There are four 500-kV transmission lines that connect the switchyard to the grid.

The NRC staff finds that the applicant has adequately described the LNP Units 1 and 2 connection to the utility grid and that the information provided is in accordance with the recommendations of RG 1.206 and the guidance in Section 8.1 of NUREG-0800.

- LNP SUP 8.1-2

The NRC staff reviewed the supplemental information related to the PEF onsite power system included under LNP SUP 8.1-2. The applicant's supplement to Section 8.1.1 is summarized as follows:

The LNP Units 1 and 2 reserve auxiliary transformers also serve as sources of maintenance power. They are supplied from the 500-kV/230-kV step-down transformers located in the switchyard.

The NRC staff finds that the applicant's description of the LNP Units 1 and 2 onsite power system is in accordance with the recommendations of RG 1.206 and the guidance in Section 8.1 of NUREG-0800.

- LNP SUP 8.1-3

The NRC staff also reviewed supplemental information included in LNP SUP 8.1-3 related to regulatory guidelines and industry standards and found it to be consistent with Section 8.1 of NUREG-0800 with the exception of the information discussed below.

LNP COL FSAR Table 8.1-201, Item 1b indicated that RG 1.155 is not applicable to LNP. This item was deemed standard among COL applications being discussed in Bellefonte's (BLN) response to Request for Additional Information (RAI) 8.1-2. In a letter dated

December 15, 2008, the applicant stated that the standard response to RAI 8.1-2 applies to the LNP COL application.

The standard response submitted by BLN in a letter dated June 24, 2008, 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 NRC staff found the above response to be 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, Positions C.2 and C.3.4.

Several discussions were held between the NRC staff and the applicant regarding this issue. Subsequently, in a letter dated April 15, 2009, the BLN applicant stated that the training and procedures to support mitigation of an SBO event would be implemented in accordance with BLN COL 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. This response was found acceptable by the NRC staff.

In a letter dated December 7, 2009, the LNP applicant endorsed BLN’s revised response.

The NRC staff has verified that LNP has updated Sections 1.9.5.1.5 and 1.9.6 of the LNP COL FSAR to include the above-mentioned items including the implementation of training and procedures to support mitigation of an SBO event. This satisfies RG 1.155, Positions C.2 and C.3.4. Based on the above, the NRC 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 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 introduction to the electric power systems, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff’s technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff has 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 concludes that the applicant is in compliance with the NRC regulations. The staff based its conclusion on the following:

- LNP SUP 8.1-1 is acceptable because the applicant provided sufficient information regarding the PEF transmission system and its connection to neighboring utilities in accordance with the recommendations of RG 1.206.

- LNP SUP 8.1-2 is acceptable because the applicant's description of the LNP 1 and 2 onsite power system is in accordance with the recommendations of RG 1.206 and the guidance in Section 8.1 of NUREG-0800.

- LNP SUP 8.1-3 is acceptable because the applicant addressed COL-specific regulatory guidelines and industry standards and additional new regulatory guidelines, are adequately addressed by the applicant. In conclusion, the applicant has 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 RGs 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 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 LNP 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

Section 8.2 of the LNP COL FSAR, Revision 9, incorporates by reference Section 8.2 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Section 8.2, the applicant provided the following:

AP1000 COL Information Items

- LNP COL 8.2-1

The applicant provided additional information in LNP 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 500-kV/230-kV switchyard and the transmission lines connecting Units 1 and 2 to the switchyard and the 500-kV switchyard to various substations throughout the transmission grid; (2) the connections of the generator step-up (GSU) transformers and the 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 designs of the GSU transformers, unit auxiliary transformers (UATs), and RATs; (6) the design of the control building in the plant site 500-kV/230-kV switchyard; (7) the administrative control of the 500-kV/230-kV switchyard and transmission line circuit breakers, (8) the switchyard and transmission line testing and inspection plan, and (9) voltage operating range, frequency decay rate, and preservation of grid connection. LNP COL 8.2-1 is addressed in FSAR Sections 8.2.1, 8.2.1.1, 8.2.1.2, 8.2.1.3, and 8.2.1.4.

- LNP COL 8.2-2

The applicant provided additional information in LNP 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 switchyard arrangement and design of the protective relaying scheme; and (2) a transmission system study performed 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. LNP COL 8.2-2 is addressed in LNP COL FSAR Sections 8.2.1.2.1 and 8.2.2.

Site-Specific Information Replacing Conceptual Design Information (CDI)

- LNP CDI

The applicant provided site-specific information 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

- LNP SUP 8.2-1

The applicant provided supplemental information describing details of a failure modes and effects analysis (FMEA) performed for the offsite power distribution system, plant site switchyard, and the transmission system.

- LNP SUP 8.2-2

The applicant provided supplemental information describing the formal agreement between LNP and PEF's Transmission Operations and Planning organization, which is the transmission system operator (TSO). The applicant provided supplemental information describing PEF's responsibility for assuring that adequate voltage is available to LNP Units 1 and 2; 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 addition, describing the interfaces between LNP and PEF's Transmission Operations that protocols are in place for LNP to remain cognizant of grid vulnerabilities in order to make informed decisions regarding maintenance activities critical to the electric system.

- LNP SUP 8.2-3

The applicant provided supplemental information describing the reliability of the 500-kV transmission lines that feeds the LNP site for the period from August 2003 to January 2008.

- LNP SUP 8.2-4

The applicant provided supplemental information describing the protective devices controlling the switchyard breakers, stating that their settings are determined with consideration given to preserving the plant grid connection following a turbine trip.

- LNP SUP 8.2-5

In a letter dated March 21, 2014, the applicant provided a supplemental response to RAI Letter No. 114 that proposed to revise the FSAR with a new Section 8.2.1.2.2 in order to address Bulletin 2012-01, "Design Vulnerability in Electric Power System."

Interface Requirements

The plant interfaces for the standard design of the AP1000 are discussed in AP1000 DCD Tier 2, Section 8.2.5, and in Items 8.1, 8.2, and 8.3 of AP1000 DCD Tier 2, Table 1.8-1, where they are identified as "non-nuclear safety (NNS)" interfaces.

Inspections, Tests, Analyses and Acceptance Criteria

In a letter dated March 21, 2014, the applicant provided a supplemental response to RAI Letter No. 114 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 the Commission regulations for the offsite power system are given in Sections 8.1 and 8.2 of NUREG-0800.

The regulatory bases for acceptance of the COL information and supplementary information items are established in:

- 10 CFR Part 50, "Domestic licensing of production and utilization facilities," Appendix A, "General Design Criteria for Nuclear Power Plants" (GDC) Criterion 17 "Electric power systems";

- 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)"; and

- 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 NRC staff reviewed Section 8.2 of the LNP 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.[2] The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to the offsite power system. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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 [VEGP] Units 3 and 4) were equally applicable to the LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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. Confirmatory items that are first identified in this SER section have a LNP designation (e.g., Confirmatory Item LNP 8.2-1).

The staff reviewed the information in the LNP COL FSAR:

AP1000 COL Information Item

- LNP COL 8.2-1

The applicant provided additional information in LNP 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 Appendix F of NUREG-1793, 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 provide the design of the ac power transmission system and its testing and inspection plan (COL Action Item 8.2.3.3-1).

The NRC staff reviewed the resolution to COL information item, LNP COL 8.2-1, related to the transmission system design, testing, and inspection addressed in Section 8.2 of the LNP COL FSAR. The NRC staff's evaluation is described below.

LNP Units 1 and 2 receive offsite ac power from a common 500-kV/230-kV switchyard which is connected to the PEF transmission network. The applicant described the connection of the RATs to the 500-kV to 230-kV transformers in the switchyard. The normal power supply to the main ac power system is provided from the main generator through the unit auxiliary transformers (UATs). The 500-kV line is the preferred power supply and is the recognized GDC 17 offsite power source for LNP Units 1 and 2. When either the normal power or the preferred power supply is available, the RATs serve as a source of maintenance power. Thus, when in use, the 230-kV line becomes the recognized GDC 17 offsite power source. The NRC staff reviewed the resolution to the supplemental information LNP COL 8.2-1 related to the description of the offsite power system. The staff determined that additional information was needed to complete the technical evaluation of this item.

FSAR Section 8.2.1.1.1 describes the ratings for the 500-kV and 230-kV circuit breakers associated with the LNP 1 and LNP 2 and states that they are rated at 3000A, with interrupting capability of 50,000 amperes (amps) root-mean-square (RMS). This section further describes the rating for the disconnect switches. Since no basis is provided for the specified ratings, in RAI 8.2-1, the staff requested the applicant to explain why the ratings for circuit breakers and disconnect switches in the switchyard are adequate for the application. In particular, the staff asked the applicant to identify the maximum fault available from the system and confirm that the breaker interrupting ratings, both symmetrical and asymmetrical, are consistent with the available fault. In a letter, dated June 23, 2009, the applicant stated that it had used steady state power flow simulations to determine the required current capability of transmission facilities, such as circuit breakers and disconnect switches. The facility ratings were determined for all line-in and line-out conditions. The applicant determined that none of the 500-kV and 230-kV circuit breakers and disconnects switches showed a loading condition above 3000 amps and were, therefore, adequate. The applicant also stated that they had used short circuit simulations to determine the required maximum interrupting capability of the circuit breakers. The analysis assumed that all generating sources relevant to the new facility were in service. Under this assumption, the short circuit levels at the Levy substation were below 28kA. The applicant concluded that the interrupting capability of 50,000 amps for the circuit breakers was adequate. The staff finds the applicant's response acceptable because the design of the offsite system components meets the requirements of GDC 17. Therefore, the NRC staff finds the issues in RAI 8.2-1 are resolved.

With regard to switchyard and transmission lines testing and inspections, described in FSAR Section 8.2.1.4, in RAI 8.2-2 the staff requested the applicant to indicate the extent to which maintenance and modifications to the switchyard and substation will be reviewed, controlled, and approved through the LNP process. In a letter dated June 23, 2009, the applicant stated that PEF utilizes procedure NGGM-IA-0003, "Transmission Interface Agreement for Operations, Maintenance, and Engineering Activities at Nuclear Plants," for testing and inspections. Accordingly, an individual is assigned from the LNP engineering organization to serve as the Switchyard System Engineer (SSE) and another individual is assigned from LNP maintenance organization to serve as the Plant Transmission Activities Coordinator (PTAC). The PTAC serves as the point of contact for transmission maintenance activities impacting the nuclear plant, while the SSE serves as the point of contact for coordinating all transmission engineering and power system operation activities requiring pre-planning and scheduling among various nuclear and non-nuclear organizations. The PTAC is also responsible for ensuring that transmission equipment within the scope of the Maintenance Rule is maintained in compliance with NRC regulations and that design changes produced by Transmission Engineering are properly reviewed for impact by the Plant and Transmission Engineering. The staff review of the applicant's response observed that the list of PTAC's responsibilities does not include communication to the grid operator of risk-sensitive plant maintenance activities. Therefore, in RAI 8.2-8, the staff asked the applicant to indicate whether: (a) it coordinates Nuclear Power Plant maintenance activities that can have an impact on the transmission system with the PTAC and TSO; and (b) it has contacts with the TSO to determine current and anticipated grid conditions as part of the grid reliability evaluation performed before conducting grid-risk-sensitive maintenance activities. In a letter dated February 5, 2010, the applicant stated that the

Interface Agreement and associated communication protocols will be in accordance with the requirements of NERC Reliability Standard NUC-001. In particular, the applicant stated that the Interface Agreement requires that: (a) Nuclear Plant Operations notify the TSO of any plant activity that has the potential to impact the generation capability of the plant or to create perturbations on the grid; and (b) Nuclear Plant Operations and the TSO hold a pre-job briefing for field work activities, including maintenance. Part of the pre-job briefing is a discussion of the risk assessment that has been performed. The risk involved is determinant in the decision as to when and how to proceed with the activity. The staff finds the applicant's response to be acceptable because adequate communication is being established between the PTAC and the TSO, thus ensuring that grid-risk-sensitive activities are adequately addressed to ensure the reliability of the offsite system in conformance with the requirements of GDC 17. Therefore, the NRC staff finds the issues in RAI 8.2-2 and RAI 8.2-8 are resolved.

In RAI 8.2-3 the staff asked that the applicant to indicate how the information from the PTAC will be shared among the LNP units. In a letter dated June 23, 2009, the applicant stated that the PTAC is a member of the Nuclear Plant Engineering organization and provided examples of information requiring to be shared by the PTAC. These included, briefing management of any concerns related to maintenance backlogs, known deficiencies and maintenance test results; entering degradation trends, line, or component failures or transients into the site Corrective Action Program; advising plant management of design ratings of lines, structures, and insulators for wind speeds; and maintaining system health report and asset management plan. The staff finds the applicant's response acceptable because it is consistent with the requirements of GDC 18 and the guidelines of RG 1.206. Therefore, the NRC staff finds the issue in RAI 8.2-3 resolved.

Additionally, the applicant provided the site-specific voltage and frequency variations expected at the LNP Units 1 and 2 switchyard during transient and steady state operating conditions and the site-specific frequency decay rate to satisfy LNP COL 8.2-1.

- LNP COL 8.2-2

The applicant provided additional information in LNP COL 8.2-2 to resolve COL Information Item 8.2-2, which states:

> The Combined License applicant will address the technical interfaces 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 NRC staff's evaluation of the technical interfaces is addressed under "Interface Requirements" in this section of the SER.

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 Appendix F of NUREG-1793, 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 NRC staff reviewed the resolution to COL information item, LNP COL 8.2-2, related to the transmission system stability analysis and switchyard circuit breaker protective device settings included under Section 8.2 of the LNP COL FSAR. The NRC staff's evaluation follows.

LNP COL 8.2-2 was provided by the applicant describing details of: 1) the switchyards arrangement and design of the protective relaying scheme; and 2) a transmission system study performed to verify grid stability, switchyard voltage, and frequency to confirm the transmission system capability to maintain RCP operation for three seconds following a turbine trip as specified in AP1000 DCD Section 8.2.2. LNP COL 8.2-2 is addressed in LNP COL FSAR Sections 8.2.1.2.1 and 8.2.2.

The 500-kV and 230-kV switchyards are locally interconnected and each designed with two (2) full-capacity main buses and composite breaker-and-a-half/double-breaker arrangement for reliability and maintainability. This arrangement allows for isolation of components and buses, while preserving the plant's connection to the grid. The transmission line protection consists of three different high speed schemes for 500-kV and two high speed schemes for 230-kV lines. Each scheme has impedance backup non-pilot schemes and directional comparison blocking schemes with (as necessary) permissive over reach trip schemes used for bus fault protection. For both 500-kV and 230-kV systems, breaker failure protection schemes are also used. Transformer protection consists of two different high speed schemes.

The NRC 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 LNP's connection to the grid following a turbine trip. This satisfies COL Action Item 8.2.3.1-2.

With regard to grid stability, the applicant stated that LNP had completed a transmission system study of the offsite power system for the addition of LNP 1 and LNP 2. This study evaluated, overloads and voltage impact on the transmission system; transient and dynamic stability of LNP 1 and LNP 2; voltage and frequency response during a turbine trip followed by a generator trip; and frequency decay rate for large, regional generation/load mismatches. The applicant determined that the transmission system, with the planned transmission system changes, will accommodate the addition of LNP 1 and LNP 2; the transient and dynamic stability performance of LNP 1 and LNP 2 is within acceptable limits for the proposed configuration; the results of turbine trip simulations demonstrate that the voltage and frequency of the 26-kV generator buses and 500-kV switchyard buses will remain within the required limits for at least 3 seconds following the turbine trip of either LNP 1 or LNP 2; and the simulations performed as part of joint studies within Florida Reliability Coordinating Council (FRCC) demonstrate that the rate of frequency decay for large generation/load mismatches is well within acceptable limits.

Therefore, the applicant concluded that the interface requirements for steady state load, nominal voltage, allowable voltage regulation, nominal frequency, allowable frequency fluctuation, maximum frequency decay rate, and the limiting under frequency value for the RCP are met. Therefore, the grid stability analysis confirmed that 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 DCD Section 8.2.2 (COL Action Items 8.2.3.1-1 and 8.2.3.1-3).

FSAR Section 8.2.2 states that, "in order to maintain Reactor Coolant Pump operation for three seconds following a turbine trip …, the grid voltage at the high side of the main step-up and reserve auxiliary transformers cannot drop from the pre-trip steady-state value by more than 15 percent of the rated voltage." Therefore, in RAI 8.2-6, the staff requested the applicant to indicate the estimated minimum pre-trip steady-state voltage at the transformers, whether this voltage was used in the analysis, and whether a system disturbance would meet the 15 percent requirement. In a letter dated June 23, 2009, the applicant stated that the estimated pre-trip steady-state voltage at the high side of the main step-up and reserve auxiliary transformers is between 0.95 and 1.05 per unit and that the high side voltage used in these analyses was 1.025 per unit for the main step-up transformers and 0.955 per unit for the reserve auxiliary transformers. The applicant also stated that computer simulations of a turbine trip with this alignment of the RCPs were performed using a pre-trip steady state generator bus voltage of 0.98 per unit. These simulations demonstrated that the generator bus voltage drop would be approximately 3 percent, significantly less than the maximum allowable drop of 15 percent. 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 NRC staff finds the issues in RAI 8.2-6 resolved.

The staff observed that LNP COL FSAR did not specifically discuss how power and control cables are routed from the switchyard to the plant. In RAI 8.2-5, the staff asked the applicant to describe whether routing of these cables is underground and to describe the cables design features and the monitoring program that will be implemented to avoid or arrest the degradation of cable insulation from the effects of moisture. In its response dated June 23, 2009, the applicant stated that high voltage connections between the AP1000 power block and the switchyard are routed overhead. The applicant also stated that, the power, control and instrumentation cables that are routed underground from the AP1000 power block to the switchyard will have moisture/water resistant jackets and manholes for duct bank access that are below the ground water level will have sump pumps. The staff found the response to be inadequate because it was not consistent with Generic Letter (GL) 2007-01's description of inspection, testing and monitoring programs to detect the degradation of inaccessible or underground power cables that support equipment and other systems that are within the scope of 10 CFR 50.65 (the Maintenance Rule). Therefore, in RAI 8.2-9, the staff requested the applicant: to indicate whether they had made any plans to implement a testing and inspection program for inaccessible or underground power cables; indicate the frequency for such testing and inspection; or provide justification for not developing such a program.

In its response dated February 5, 2010, the applicant reiterated that the Levy County site does not include any high voltage cables that are routed underground or any medium voltage cables that are routed between the AP1000 power block and the switchyard. Regarding low voltage cables that are routed between the AP1000 power block and the switchyard, the applicant stated that, by definition, they are not exposed to significant voltage and that, due to the sump pumps, they will not be exposed to significant moisture as described in NUREG-1801, XI.E3, "Generic Aging Lessons Learned (GALL) Report." Therefore, the applicant concluded that the low voltage cables will not require periodic testing, beyond post installation testing and initial functional testing.

The staff did not agree with the applicant conclusions. While it is true that cable insulation degradation and negative effects increase with the voltage to which the cables are exposed, the low voltage cable insulation is not exempt from degradation due to moisture or submergence.

NUREG-1801, XI.E1, for instance, states, in part, "in a limited number of localized areas [of a nuclear power plant], the actual environments may be more severe than the plant design environment for those areas. Conductor insulation materials used in cables and connections may degrade more rapidly than expected in these adverse localized environments. An adverse localized environment is a condition in a limited plant area that is significantly more severe than the specified service environment for the cable. An adverse variation in environment is significant if it could appreciably increase the rate of aging of a component or have an immediate adverse effect on operability. The purpose of the aging management program described herein is to provide reasonable assurance that the intended functions of electrical cables and connections that are not subject to the environmental qualification requirements of 10 CFR 50.49 and are exposed to adverse localized environments caused by heat, radiation, or moisture will be maintained consistent with the current licensing basis through the period of extended operation." This statement does not exclude low voltage cables. Furthermore, as described in GL 2007-01, operating experience indicates the occurrence of failures of buried medium-voltage [as well as] ac and direct current (dc) low voltage cables from insulation failure. The concern is that exposure to 100 percent Relative Humidity and/or intermittent submergence may result in cable insulation degradation and multiple grounds that may go unnoticed until the cables are submerged again and, thus, prevent the affected components from performing their intended function. However, the NRC staff finds the issues in RAIs 8.2-5 and 8.2-9 resolved as follows:

The following portion of this technical evaluation section is reproduced from Section 8.2.4 of the VEGP SER:

> <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.

> *The 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

- LNP SUP 8.2-1

LNP SUP 8.2-1 was provided by the applicant describing details of a FMEA performed for the offsite power distribution system, plant site switchyard, and the PEF transmission system. The NRC staff has reviewed the FMEA of the LNP switchyard and confirmed that the applicant has identified no single initiating event, such as a breaker not operating during a fault condition; a fault on a switchyard bus; a spurious relay trip; and a loss of control power supply which would cause failure of more than one single offsite transmission line, or a loss of offsite power to either LNP1 or LNP2 via the GSU. The staff also finds that the applicant's analysis is in conformance with the guidance of RG 1.206. Therefore, LNP SUP 8.2-1 is acceptable.

- LNP SUP 8.2-2

With regard to LNP SUP 8.2-2 the applicant provided, in part, the following information:

> The interfaces between LNP and PEF's Transmission Operations and Planning Department are managed via a formal Interface Agreement. PEF conducts transmission system operations under a vertically integrated utility business model. Under [this] model, the System Operators (Grid Operators) are the TSOs, and operate both the transmission and generation systems (nuclear and non-nuclear) and work in the same company that will hold the license to operate LNP. LNP off-site power reliability is jointly managed by the system operators, transmission personnel, and licensed nuclear plant personnel through communications and actions governed by the formal Interface Agreement.
>
> The Interface Agreement specifies the responsibilities and lines of communication for the various organizations responsible for the operation, maintenance, and engineering of facilities associated with LNP. LNP operators are directed to notify the TSO of any plant activity that may impact generation capability. The TSO is required to monitor system conditions to ensure adequate voltage is maintained to support LNP, and promptly notify the LNP operators of existing, or anticipated conditions, which would result in inadequate voltage support.
>
> The TSO and LNP plant operators coordinate operations to maintain the switchyard voltage such that the steady state voltage on the 26-kV isophase bus is within 0.95 – 1.05 per unit (pu) of its nominal value.
>
> LNP procedures address the criteria used to determine when the main control room (MCR) is required to contact the TSO. The procedures used by the TSOs direct them to promptly notify the LNP operators of conditions for which there would not be adequate switchyard voltage, including predicted post LNP trip conditions. These procedures include separate steps that address both current and anticipated conditions. The intent of these separate steps is to provide, to the extent possible, early warning to the LNP operators of problem conditions.
>
> The TSO uses procedures based on enveloping transmission planning analyses to operate the grid. As long as the grid configuration is within that allowed by the procedure under various system loading conditions, adequate plant voltage support is assured. Specific case studies are also used to support planned grid configurations when not clearly bounded by existing analyses. In addition to the transmission system analysis-based procedures, the TSO also uses computer programs that can predict LNP switchyard voltages expected to occur upon realization of any one of a number of possible losses to the grid, including a trip of the LNP generator, a trip of another large generator, or the loss of an important transmission line. This program tool operates based on raw data from transducers across the system, which is processed through a state estimator to generate a current state of the system snapshot. The output is then processed through a contingency analysis program that generates a set of new results with various single elements of the system out of service. These results are then screened against a predetermined set of acceptance limits. Postulated scenarios which then do not meet the acceptance limits, are listed for review by the TSO. The predictive

> analysis computer program updates approximately every 10 minutes. Also, the grid operating procedures that are based on enveloping transmission system analyses are updated when transmission system or plant changes require it.
>
> Procedural guidance is provided regarding a target switchyard voltage schedule and operation of the main generator voltage regulator. Operation of the main generator within the plant voltage schedule ensures that a trip of the generator does not result in an unacceptable voltage drop in the switchyard. The TSO procedure defines the TSO's actions and requirements during high load conditions. These actions are based on transmission system enveloping analyses wherein the worst-case loss of a generating station (including LNP) on the PEF system is considered relative to LNP voltage support. In the event system conditions are outside the guidelines of the analysis-based procedure, the TSO will alert the LNP operators to that effect.

The NRC staff reviewed the information provided by the applicant on the functions of the TSO that establishes a voltage schedule for the LNP 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 its nominal value. Based on the information provided by the applicant on the functions of TSO, the NRC staff finds that the applicant has demonstrated that protocols are in place for LNP to remain cognizant of grid vulnerabilities in order to make informed decisions regarding maintenance activities critical to the electric system. This is consistent with Generic Letter (GL) 2006-2 of which one of the provisions is to reduce the likelihood of losing offsite power. The NRC finds that the information provided is also consistent with the guidelines of RG 1.206. Therefore, LNP SUP 8.2-2 is acceptable.

- LNP SUP 8.2-3

With regard to LNP SUP 8.2-2 the applicant provided, in part, the following information:

> From August 2003 to January 2008, the average grid availability for the existing PEF 500-kV transmission lines within the system is approximately 99.9 percent, with eleven (11) forced outages. The average frequency of forced line outages since 2003 is approximately 2.44 per year for the involved lines, with the majority due to public interference, animal or lightning strikes causing the outages. Leading causes of forced outages of significant duration that were recorded are public interference.

The NRC staff review of the supplemental information provided regarding the grid availability historical data finds that the supplemental information is consistent with the guidelines of RG 1.206. Therefore, LNP SUP 8.2-3 is acceptable.

- LNP SUP 8.2-4

With regard to LNP SUP 8.2-4 the applicant stated that the protective devices controlling the switchyard breakers are set with consideration given to preserving the plant grid connection following a turbine trip. The staff concludes that the switchyard breaker arrangement, the protection of lines by independent high speed relay schemes, and the breaker failure scheme would preserve the LNP's connection to the grid following a turbine trip. On this basis, LNP SUP 8.2-4 is satisfied.

- LNP SUP 8.2-5

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," (Agencywide Documents Access and Management System (ADAMS) Accession Number ML12074A115) on July 27, 2012, to all holders of operating and combined licenses (COL) 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). 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 08-1 (ADAMS Accession Number ML12228A611), dated August 15, 2012, to Duke Energy Florida (DEF) for LNP Units 1 and 2, to address the matters described in Bulletin 2012-01 and to ensure that the LNP design meets GDC 17.

In response to RAI 08-1, "Single-Phase Open Circuit Condition," DEF provided its supplemental response in a letter dated June 4, 2013 (ADAMS Accession Number ML13157A025), for LNP 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 staff's review of this response, staff could not determine whether the LNP 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. Therefore, the staff requested DEF, in an RAI dated August 14, 2013 (ADAMS Accession Number ML13226A124), to clarify or provide supporting information for several statements from its June 4, 2013, RAI response to determine whether the LNP Units 1 and 2 design meets the GDC 17 requirements.

On November 1, 2013, the NRC conducted a public meeting (ADAMS Accession Number ML13309B117) with representatives from the Nuclear Energy Institute and industry 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. The industry informed NRC 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 Bulletin open phase issue.

GDC 17 requires, in part, that "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 that (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 is 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. Further, the ac power systems do supply power to equipment that is important to safety since that equipment serves defense-in-depth functions, as follows: 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 LNP should address the design vulnerability identified in Bulletin 2012-01.

To address the electric power system vulnerability related to Bulletin 2012-01, it is the staff's position that an acceptable approach for passive designs includes the following four elements: 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; an alarm in the main control room for operators to take manual actions if the standby diesel generators are not automatically connected to the ES-1 and ES-2 buses; 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 FSAR; and procedures and training for the operating and maintenance staff. This approach ensures the required offsite AC power source with adequate capacity and capability is available to important to safety equipment including safety related battery chargers to meet their intended safety function in accordance with GDC 17 requirements.

In a letter dated March 21, 2014 (ADAMS Accession Number ML14010A421), the applicant provided a supplemental response to the Staff's RAI. In this response, the applicant added new features, described below, that address the staff's concerns. To make its conclusion on the acceptability of LNP SUP 8.2-5, the staff relied on information, detailed below, related to the loss-of-phase detection system installed on the credited GDC 17 offsite power circuit as provided in the applicant's March 21, 2014, supplemental RAI response, including the RAI response, proposed FSAR changes, and a proposed ITAAC. Because this information in the March 21, 2014, supplemental RAI response addresses this issue, the staff's analysis and finding does not rely on information including the RAI response, proposed FSAR language, and proposed ITAAC related to other design features in that or previous responses (dated January 9, 2014, October 24 and June 4, 2013, or September 14, 2012), including undervoltage protective relays, potential transformers on the medium voltage buses, negative sequence motor trips or other running load trips, and battery charger undervoltage detection. The staff evaluation does not address the capability of these other design features to detect a loss-of-phase condition.

As part of the March 21, 2014 supplemental response, the applicant provided text that will be added to the next revision of the FSAR. Some of the proposed text addressed the original design features and is not included below. The additional text that directly addresses the staff's position is as follows:

> Associated LNP COL Application Revisions:
>
> 1) Add the following subsection to FSAR Chapter 8 following Subsection 8.2.1.2.1 with a LMA of LNP SUP 8.2-5:
>
> 8.2.1.2.2 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.
>
> . . .
>
> 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.
>
> 2) Add the following subsection to FSAR Chapter 8:
>
> 8.2.6 References
>
> Add the following information at the end of DCD Subsection 8.2.6.
>
> 201. NRC Bulletin 2012-01, "Design Vulnerability in Electric Power System," July 27, 2012.
>
> . . .

4) In LNP COLA Part 10, Appendix B. Inspections, Tests, Analyses and Acceptance Criteria, add the following information as a new line item 7 in Table 2.6.12-1:

Design Commitment	**Inspections, Tests, 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.

These proposed additions to the FSAR and the ITAAC acceptably address the staff position as to what is necessary to protect a passive plant with regard to an open phase condition as described in Bulletin 2012-01, and that the LNP design meets GDC 17. Therefore, the staff finds this issue to be resolved and RAI 08-1 closed pending the staff's confirmation that the revisions to the FSAR noted above are incorporated in the LNP Units 1 and 2 COL application. The staff is tracking these revisions as **LNP Confirmatory Item 8.2-1**.

<u>Resolution of LNP Confirmatory Item 8.2-1</u>

LNP Confirmatory Item 8.2-1 is an applicant commitment to update its FSAR and ITAAC to include details necessary to protect a passive plant with regard to an open phase condition, described in Bulletin 2012-01. The staff verified that the FSAR and ITAAC were appropriately updated and that the LNP design meets GDC 17. As a result, LNP Confirmatory Item 8.2-1 is now closed.

- LNP CDI

The CDI information provided by the applicant regarding the transformer area located next to each unit's turbine building is consistent with the AP1000 DCD and satisfies the applicable requirements of GDC 17.

Interface Requirements

The plant interfaces for the standard design of the AP1000 are discussed in DCD Tier 2, Section 8.2.5, and in Items 8.1, 8.2, and 8.3 of DCD Tier 2, Table 1.8-1, where they are identified as 'non-nuclear safety (NNS)' interfaces.

The applicant incorporated by reference Section 1.8 of the AP1000 DCD. 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).[3] As required by 10 CFR 52.79(d)(2), the COL application must demonstrate how these interface items have been met.

In order to satisfy plant Interface Item 8.1 in AP1000 DCD Tier 2, Table 1.8-1, the applicant provided the design criteria, RGs, and IEEE standards in Section 8.1.4.3 of the LNP COL FSAR. The NRC staff finds the information to be consistent with Section 8.1 of NUREG-0800 and acceptable. Therefore, this interface item for offsite power system has been met.

With regard to plant Interface Item 8.2 in AP1000 DCD Tier 2 Table 1.8-1, the staff observed that in FSAR Subsection 8.2.2 the applicant states that the "transmission study has confirmed that the interface requirements for steady state load, nominal voltage, allowable voltage regulation, nominal frequency, allowable frequency fluctuation, maximum frequency decay rate, and the limiting under frequency value for the RCP have been met." In RAI 8.2-4 the staff asked the applicant to provide the summary of the grid stability analysis results, the assumptions made, and the acceptance criteria for each case analyzed. Additionally, the applicant was requested to 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 letter dated June 23, 2009, the applicant provided a table comparing the required parameter values (acceptance criteria) and the associated analysis results. Additionally, the applicant stated that the LNP COL FSAR would be revised to include such table. The staff has verified that Revision 2 to the LNP FSAR contains the foregoing change. Therefore, the staff finds that the analysis results meet the AP1000 design requirements, the requirements of GDC 17 and the guidelines of RG 1.206. Therefore, this issue is resolved and Interface Item 8.2 in AP1000 DCD Tier 2 Table 1.8-1 is satisfied.

Regarding plant Interface Item 8.3 in AP1000 DCD Tier 2, Table 1.8-1, the applicant did not provide a statement affirming that "the protective devices controlling the switchyard breakers are set with consideration given to preserving the plant grid connection following a turbine trip." In RAI 8.2-7, the staff asked the applicant to provide a reference to where this issue is discussed in the LNP application, or to provide a proposed revision to the application to address the issue.

---

[3] 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 its response dated August 6, 2009, the applicant identified a proposed revision to LNP COL FSAR Section 8.2.1.2.1 to add LNP SUP 8.2.4 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 NRC staff verified that the LNP COL FSAR was updated to include this change and concludes that the switchyard arrangement, the protection of lines by independent high speed relaying, and breaker failure would preserve the LNP connection to the grid following a turbine trip satisfying the requirements of GDC 17. Therefore, the NRC staff finds this interface has been met and the issue in RAI 8.2-7 resolved. On this basis, COL Information Item 8.2-3.1-2 is also resolved.

The NRC staff has reviewed the information supplied by the applicant and concludes that the applicant has adequately addressed Interface Items 8.1, 8.2, and 8.3 of AP1000 DCD Tier 2, Table 1.8-1.

Inspections, Tests, Analyses and Acceptance Criteria

In a letter dated March 21, 2014, the applicant proposed to revise Part 10 of the COL application to include the following two site-specific ITAAC.

The applicant proposed the following site-specific ITAAC for the Main AC Power System (ECS) to be added to DCD Tier 1 Section 2.6.1 as new item 4.g in Table 2.6.1-4. This ITAAC was not necessary for the staff to reach its conclusions regarding LNP SUP 8.2-5. The staff did not evaluate it, and does not intend to include it in the license.

Design Commitment	Inspections, Tests, and Analyses	Acceptance Criteria
4.g.) The ECS provides an alarm in the MCR 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 as new line item 7 in Table 2.6.12-1 in LNP COL application Part 10, Appendix B.

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 No. 7 is presented above in the evaluation of LNP SUP 8.2-5.

8.2.5 Post Combined License Activities

For the reasons discussed in the technical evaluation section above, the staff finds acceptable ITAAC No. 7 as defined in SER Table 8.2A-1, “ITAAC for Offsite Power System.”

8.2.6 Conclusion

The NRC reviewed the application and checked the referenced DCD. The NRC 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 LNP COL FSAR related to this section. The results of the NRC staff’s technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented within the LNP COL FSAR is acceptable and meets the requirements of GDC 17 and GDC 18. The staff based its conclusion on the following:

- LNP 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, and its testing and inspection plan in accordance with the guidelines of RG 1.206.

- LNP 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 with the guidelines of RG 1.206. In addition, the NRC 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 LNP's connection to the grid following a turbine trip.

- LNP CDI in Section 8.2.1 of the LNP COL FSAR is acceptable because the applicant provided sufficient information involving the transformer area being located next to each unit's turbine building in accordance with the guidelines of RG 1.206.

- LNP SUP 8.2-1 is acceptable because the applicant provided sufficient information describing details of a failure analysis performed for the offsite power distribution system, and plant site switchyard in accordance with the guidelines of RG 1.206.

- LNP SUP 8.2-2 is acceptable because the applicant provided sufficient information to describe PEF's responsibility for maintaining area bulk transmission system reliability. The applicant also provided sufficient information to demonstrate that protocols are in place for LNP to remain cognizant of grid vulnerabilities in order to make informed decisions regarding maintenance activities critical to the electric power system in accordance with the guidelines of RG 1.206 and GL 2006-2.

- LNP SUP 8.2-3 is acceptable because the applicant provided sufficient information regarding causes of outages of the transmission line over the past 5 years in accordance with the guidelines of RG 1.206.

- LNP SUP 8.2-4 is acceptable because the applicant provided sufficient information to satisfy the interface requirement regarding the setting of protective devices controlling the switchyard to preserve the LNP connection to the grid following a turbine trip satisfying the requirements of GDC 17.

- LNP SUP 8.2-5 and proposed ITAAC No. 7 are acceptable, pending closure of LNP Confirmatory Item 8.2-1, because the applicant provided sufficient information to address the loss-of-phase condition vulnerability described in Bulletin 2012-01 and to comply with GDC 17.

- The applicant provided sufficient information regarding the interfaces for standard design from the generic DCD Table 1.8-1, Items 8.1, 8.2, and 8.3.

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 is 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, and NRC regulations.

8.2.A.2 Summary of Application

Section 14.3 of the LNP COL FSAR, Revision 9, incorporates by reference Section 14.3 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Section 14.3, the applicant provided the following:

Supplemental Information

- STD SUP 14.3-1

The applicant provided supplemental information related to the offsite power system in STD Supplement (SUP) 14.3-1 in LNP COL FSAR Section 14.3.2.3.

ITAAC

Part 10 of the COL application includes six SS-ITAAC in Table 2.6.12-1 addressing the offsite power system.

In a letter dated March 21, 2014, the applicant proposed an additional SS-ITAAC related to detection and alarm of a loss-of-phase condition. The staff's evaluation of this ITAAC appears in the preceding Section 8.2.

8.2.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 the Commission regulations for ITAAC are given in Section 14.3 of NUREG-0800.

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 NRC staff reviewed Section 14.3 of the LNP 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.[2] The NRC 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 NRC staff's evaluation of the information incorporated by reference in the LNP 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 [VEGP] Units 3 and 4) were equally applicable to the LNP Units 1and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 8.2.A.4 of the VEGP SER:

> *<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 BNL 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 the technical evaluation section above, the staff proposes to include the following ITAAC related to the Offsite Power System:

- The licensee shall perform and satisfy the ITAAC defined in Table 8.2A-1, "Offsite Power System."

8.2.A.6 Conclusion

The staff concludes that the relevant information presented within the LNP COL FSAR is acceptable and meets the requirements of GDC 17 and GDC 18.

8.3 Onsite Power Systems

8.3.1 Alternating Current 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 LNP, 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 systems.

8.3.1.2 *Summary of Application*

Section 8.3 of the LNP COL FSAR, Revision 9, incorporates by reference Section 8.3 of the AP1000 DCD, Revision 19. Section 8.3 of the AP1000 includes Section 8.3.1. In addition, in LNP COL FSAR Section 8.3.1, the applicant provides the following:

AP1000 COL Information Items

- LNP COL 8.3-1

LNP COL 8.3-1 describes: 1) the grounding grid system design within the plant boundary; and 2) a lightning protection risk assessment for the buildings comprising LNP Units 1 and 2.

- LNP STD COL 8.3-2

STD COL 8.3-2 describes the details of: 1) the bases of the recommendations in operation, inspection, and maintenance procedures for the onsite standby diesel generators and 2) the procedures for the periodic testing of penetration overcurrent protective devices.

Supplemental Information

- LNP SUP 8.3-1

LNP SUP 8.3-1 describes the site conditions provided in Section 2.1 and Section 2.3 of the FSAR that are bounded by the standard site conditions used to rate the diesel engine and the associated generator in DCD Section 8.3.1.1.2.3.

- LNP SUP 8.3-2

LNP SUP 8.3-2 provides supplemental information describing the site-specific switchyard and power transformer voltage.

- LNP STD SUP 8.3-4

STD SUP 8.3-4 provides supplemental information regarding periodic verification of the onsite ac power system's capability to transfer between the preferred power supply and the maintenance power supply.

8.3.1.3 *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 ac power systems are given in Section 8.3.1 of NUREG-0800.

The regulatory bases for acceptance of LNP 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 Std. 80, "Guide for Safety in AC Substation Grounding"
- IEEE Std. 665, "Guide for Generating Station Grounding"

The bases for acceptance of the part of STD COL 8.3-2, addressing the recommendations in operation, inspection, and maintenance procedures for the onsite standby diesel generators, are the guidelines of industry standards.

The regulatory bases for acceptance of the part of STD COL 8.3-2, addressing procedures for penetration protective device testing, are the guidelines of:

- RG 1.63, "Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants"

8.3.1.4 *Technical Evaluation*

The NRC staff reviewed Section 8.3.1 of the LNP COL FSAR and checked the reference 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.[2] The NRC staff's review confirmed that the information contained in the application and incorporated by reference in the LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 staff reviewed the information contained in the LNP COL FSAR:

<u>*AP1000 COL Information Items*</u>

- LNP COL 8.3-1

The NRC staff reviewed LNP COL 8.3-1 related to COL Information Item 8.3-1. COL Information Item 8.3-1 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 Appendix F of the NRC staff's FSER for the AP1000 DCD (NUREG-1793), which states:

> The COL applicant will provide the design of the site-specific grounding and lightning protection.

The NRC staff reviewed the resolution to COL information item, LNP COL 8.3-1, related to the ground grid system and lightning protection included under Section 8.3 of the LNP COL FSAR. The NRC staff's evaluation is described below.

The applicant states that a grounding grid system design within the plant boundary includes a determination of step and touch potentials near equipment and ensuring that they are within the acceptable limit for personnel safety. Actual resistivity measurements from soil samples taken at the plant site were analyzed to create a soil model. The ground grid conductor size was then determined using the methodology outlined in IEEE Std. 80, "IEEE Guide for Safety in AC Substation Grounding," and a grid configuration for the site was created. The grid configuration was modeled in conjunction with the soil model.

The NRC staff review of the grounding grid system design description observed that Table 8.1-201 of the LNP FSAR includes RG 1.204 which endorses IEEE Std. 665 for generation station grounding. The staff also observed that the same subsection of the DCD indicates compliance with IEEE Std. 665. Therefore, in RAI 08.03-01 the staff asked the applicant to discuss the extent to which the LNP ground grid design complies with IEEE Std. 665 and confirm that their use of IEEE Std. 80 did not invalidate the LNP conformance with the guidelines of RG 1.204. In a letter, dated July 13, 2009, the applicant stated that IEEE Std. 80 methodology was used in the determination of ground grid conductor size and that this methodology did not invalidate their conformance with the guidance of RG 1.204. The applicant also clarified that Appendix 1AA of the LNP COL FSAR includes RG 1.204, Revision 0, with no exceptions taken. The staff finds the applicant's response acceptable because it is consistent with the guidelines of RG 1.206. Therefore, the NRC staff finds the issues in RAI 8.03-01 resolved.

With regard to lightning protection, the applicant stated that, at LNP 1 and LNP 2, lightning protection is provided in accordance with the guidelines in RG 1.204. Specifically, the applicant stated that the zone of protection is based on 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. Lightning protection grounding is interconnected with the station/switchyard grounding system. The staff review of the applicant's description of the LNP lightning protection system design observed that in Table 8.1-201 of the LNP COL FSAR it is stated that RG 1.204 is implemented via IEEE Std. 665. Since the Regulatory Guide also endorses IEEE Std. 666-1991, "IEEE design Guide for Electric Power Service Systems for Generating Systems," IEEE Std. 1050-1996, "IEEE Guide for Instrumentation and Control Grounding in Generating Stations," and IEEE Std. C62.23-1995, "IEEE Application Guide for Surge Protection of Electric Generating Plants," in RAI 08.03-02 the staff requested that the applicant discuss the applicability of these other standards. On July 13, 2009, the applicant clarified that Appendix 1AA of the LNP COL FSAR includes RG 1.204, Revision 0, with no exceptions taken. Therefore, the applicant stated that they would also comply with the other standards in accordance with RG 1.204. Additionally, they stated that Table 8.1-201 of the LNP COL FSAR will be revised to remove the note: "Implemented via IEEE-665, IEEE Guide for Generating Station Grounding, (DCD Section 8.3, and Reference 201)," under the "Remarks" column for RG 1.204. The staff finds the applicant's response acceptable because it is consistent with the guidelines of RG 1.206. The staff also verified that the LNP FSAR has been revised to remove the note; therefore, the NRC staff finds the issues in RAI 8.03-02 resolved.

Based on the above, the staff concludes that IEEE Std. 665 provides an acceptable method for lightning protection; therefore, the supplemental information provided by the applicant on lightning protection is acceptable.

- LNP STD COL 8.3-2

The NRC staff reviewed LNP STD COL 8.3-2 related to STD COL 8.3-2 as follows.

The following portion of this technical evaluation section is reproduced from Section 8.3.1.4 of the VEGP SER:

> - *STD COL 8.3-2*
>
> *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:*
> >
> > – *Periodic testing of penetration protective devices*
> >
> > – *Diesel generator operation, inspection and maintenance in accordance with manufacturer's recommendations*
>
> *The commitment was also captured as COL Action Items 8.3.1.2-1 and 8.4.1-1 in Appendix F of the NRC staff's FSER for the AP1000 DCD (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 LNP 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.

Resolution of Standard Content Confirmatory Item 8.3.1-1

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.

Supplemental Information

- LNP SUP 8.3-1

The applicant stated in LNP SUP 8.3-1 that their site conditions are bounded by the standard site conditions in DCD Section 8.3.1.1.2.3 used to rate the diesel generators. The staff agrees that the LNP site conditions are bounded by the standard site conditions used to determine the rating.

- LNP SUP 8.3-2

The applicant provided information in LNP SUP 8.3-2 describing the site-specific switchyard and power transformer voltage. The staff found this statement of fact acceptable; no evaluation is required

- .LNP STD SUP 8.3-4

The applicant provided information in LNP STD SUP 8.3-4 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 above satisfies the requirements of GDC 18 and is, therefore, 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 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 ac power systems, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff has compared the COL information items, the supplemental information, the interfaces for standard design, and the proposed design changes and corrections within 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 the NRC regulations pending resolution of the confirmatory item discussed above. The staff based its conclusion on the following:

- LNP 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 RGs 1.206 and 1.204.

- LNP STD COL 8.3-2 is acceptable because the applicant provided sufficient information related to preoperational testing of the diesel generators and periodic testing of the penetration overcurrent protective devices consistent with industry standards and the recommendations of RG 1.63.

- LNP SUP 8.3-1 is acceptable because the applicant demonstrated its site-specific conditions are bounded by the standard site conditions in the AP1000 DCD for rating the diesel generator.

- LNP SUP 8.3-2 is acceptable because the applicant adequately addressed the site-specific switchyard and transformer voltage.

- LNP STD SUP 8.3-4 is acceptable because the applicant will implement procedures for periodic verification of offsite power system capability for automatic and manual transfer from the preferred power supply to the maintenance power supply and vice versa to satisfy the requirements of GDC 18.

8.3.2 Direct Current Power Systems

8.3.2.5 *Introduction*

The dc power systems include those dc power sources and their 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 inverters convert dc from the dc distribution system to ac instrumentation and control power, as required. These three components, when combined, provide an Uninterruptible Power Supply (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*

Section 8.3 of the LNP COL FSAR, Revision 9, incorporates by reference Section 8.3 of the AP1000 DCD, Revision 19. Section 8.3 of the AP1000 DCD includes Section 8.3.2. The advanced safety evaluation (ASE) with confirmatory items for Section 8.3.2 was based on the LNP COL FSAR, Revision 2 and DCD, Revision 17. After submitting DCD Revision 17 to the NRC, Westinghouse revised the COL information Item (COL 8.3-2) and the applicant took a departure (STD DEP 8.3-1) to address the revised COL information item. This COL information item has been incorporated into Revision 18 of the DCD; however, the discussion of the COL information item below did not change.

In addition, in LNP COL FSAR Section 8.3.2, the applicant provided the following:

<u>*Tier 2 Departure*</u>

- STD DEP 8.3-2

In a letter dated October 20, 2010, the applicant endorsed a Southern Nuclear letter dated October 15, 2010, for the VEGP application that proposed the following Tier 2 standard departure related to a proposed revision to AP1000 DCD Section 8.3.2.2. In the October 15, 2010, Southern Nuclear letter, Southern stated that the Class 1E battery chargers are designed to limit the input (ac) current to an acceptable value under faulted conditions on the output side; however, the voltage regulating transformers do not have active components to limit current; therefore, the Class 1E voltage regulating transformer maximum current is determined by the impedance of the transformer. 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. Since AP1000 DCD Section 8.3.2.2 states that the Class 1E voltage regulating transformers are designed to limit the input (ac) current to an acceptable value under faulted conditions on the output side, the use of the breakers/fuses for the regulating transformers for isolation function, in lieu of current limiting characteristics as presented in the AP1000 DCD, is a departure for VEGP. Because the issue is identified as a standard item it is also a departure for LNP.

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; and 2) the clearing of ground faults on the Class 1E dc power system. In a letter dated March 1, 2011, the applicant endorsed a Southern Nuclear letter dated October 15, 2010, for the VEGP application that proposed to revise STD COL 8.3-2 by adding information related to periodic testing for the battery chargers and voltage regulating transformers.

Supplemental Information

- STD SUP 8.3-3

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 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 dc power systems are given in Section 8.3.2 of NUREG-0800.

The regulatory basis for acceptance of COL information item, STD COL 8.3-2 and STD SUP 8.3-3, is established in:

- GDC 17

- GDC 18

- RG 1.206

- RG 1.129, “Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants”

- IEEE Std. 450, “Recommended Practice for the Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications”

- RG 1.75, “Physical Independence of Electrical Systems,” Revision 3

8.3.2.4 *Technical Evaluation*

The NRC staff reviewed Section 8.3.2 of the LNP COL FSAR 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.[2] The NRC staff’s review confirmed that the information contained in the application and incorporated by reference addresses in the LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 8.3.2.4 of the VEGP SER:

> *<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.

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-SRP 8.3.2-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, AP1000 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****.*

<u>Resolution of Standard Content Confirmatory Item 8.3.2-2</u>

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.

The following portion of this technical evaluation section is reproduced from Section 8.3.2.4 of the BLN SER:

<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 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 dc power systems, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented within the LNP 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 the applicant is in compliance with the NRC regulations. The staff based its conclusion on the following:

- STD COL 8.3-2 is acceptable because the applicant provided sufficient information involving the inspection, maintenance, and testing of Class 1E batteries and clearing of ground faults on the Class 1E dc system, and periodic testing of the battery chargers and voltage regulating transformers.

- STD SUP 8.3-3 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.

- STD 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.

Table 8.2A-1. ITAAC for Offsite Power System

Design Commitment	Inspections, Tests, and Analyses	Acceptance Criteria
1. A minimum of one offsite circuit supplies electric power from the transmission network to the interface with the onsite ac power system.	Inspections of the as-built offsite circuit will be performed.	At least one offsite circuit is provided from the transmission switchyard interface to the interface with the onsite ac power system.
2. Each offsite power circuit interfacing with the onsite ac power system is adequately rated to supply assumed loads during normal, abnormal and accident conditions.	Analyses of the offsite power system will be performed to evaluate the as-built ratings of each offsite circuit interfacing with the onsite ac power system against the load assumptions.	A report exists and concludes that each as-built offsite circuit is rated to supply the load assumptions during normal, abnormal and accident conditions.
3. During steady state operation, each offsite power source 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.	Analyses of the as-built offsite circuit will be performed to evaluate the capability of each offsite circuit to supply the voltage requirements at the interface with the onsite ac power system.	A report exists and concludes that during steady state operation each as-built offsite circuit is capable of supplying the voltage at 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.	Analyses of the as-built offsite circuit will be performed to evaluate the capability of each offsite circuit to supply the frequency requirements at the interface with the onsite ac power system.	A report exists and concludes that during steady state operation each as-built offsite circuit is capable of supplying the frequency at the interface with 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 circuit is compatible with the interrupting capability of the onsite short circuit interrupting devices.	Analyses of the as-built offsite circuit will be performed to evaluate the fault current contribution of each offsite circuit at the interface with the onsite ac power system.	A report exists and concludes the short circuit contribution of each as-built offsite circuit at the interface with the onsite ac power system is compatible with the interrupting capability of the onsite 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.	Analyses of the as-built offsite power system will be performed to confirm that power will be available to the reactor coolant pumps for a minimum of 3 seconds following a turbine trip when the buses powering the reactor coolant pumps are aligned to either the UATs or the RATs.	A report exists and concludes that voltage at the high-side of the GSU, and the RATs, does not drop more than 0.15 pu from the pre-trip steady-state voltage for a minimum of 3 seconds following a turbine trip when the buses powering the reactor coolant pumps are aligned to either the UATs or the RATs.

Table 8.2A-1. ITAAC for Offsite Power System

Design Commitment	Inspections, Tests, and Analyses	Acceptance Criteria
7) The credited GDC 17 offsite 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 offsite 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 offsite 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 offsite 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.

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 Levy Nuclear Plant (LNP) combined license (COL) Final Safety Analysis Report (FSAR), Revision 9, 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 LNP COL application are documented in NUREG-1793, "Final Safety Evaluation Report 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 LNP COL FSAR, Revision 9, 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 LNP COL FSAR Section 9.1.6, the applicant[2] provided the following:

AP1000 COL Information Item

- 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.

License Condition

- 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 LNP Part 10, Revision 2, a license condition to provide a schedule to support the NRC's inspection of operational programs and proposes to add 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 (LWR Edition)."

The regulatory basis for acceptance of the COL information and supplementary information items are established in:

---

[2] The applicant, Duke Energy Florida, LLC, was formerly identified as Duke Energy Florida, Inc., and Progress Energy Florida, Inc. In a letter dated April 15, 2013, Progress Energy Florida notified the NRC that its name was changing to Duke Energy Florida, Inc., effective April 29, 2013. The name changes and a 2012 corporate merger between Duke Energy and Progress Energy are described in Chapter 1 of the SER. For the review described in this chapter completed prior to the name change, the NRC staff did not change references to "Progress Energy," "Progress Energy Florida," or "PEF," to "Duke Energy," "Duke Energy Florida," or "DEF."

- 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 LNP 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 LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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. Confirmatory items that are first identified in this SER section have an LNP designation (e.g., **Confirmatory Item LNP 9.1-1**).

The following portion of this technical evaluation section is reproduced from Section 9.1.2.4 of the VEGP SER:

<u>*AP1000 COL Information Item*</u>

- *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 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.

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.

<u>Metamic Monitoring Acceptance Criteria</u>:

- *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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP 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 LNP COL FSAR, Revision 9, 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 in Section 9.1.3.7 of the FSAR and a proposed license condition. Section 20.3 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 other 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 LNP 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 LNP COL FSAR, Revision 9, 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 LNP 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 LNP 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 LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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.

Resolution of Standard Content Open Item 9.1-2

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****.*

Resolution of Standard Content Confirmatory Item 9.1-2

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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP 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 LNP 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 LNP 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 LNP COL FSAR, Revision 9, 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 LNP 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 LNP 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 LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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:

> *<u>Supplemental Information</u>*
>
> - *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****.*

<u>AP1000 COL Information Items</u>

- *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****.*

Resolution of Standard Content Confirmatory Item 9.1-3

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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP 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 LNP 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 LNP 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*

Section 9.2 of the LNP COL FSAR, Revision 9, incorporates by reference Section 9.2 of the AP1000 DCD, Revision 19. Section 9.2 of the DCD includes Section 9.2.1.

In addition, in LNP COL FSAR Section 9.2.1, the applicant provided the following:

Supplemental Information

- LNP 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 LNP 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 LNP 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 LNP COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the LNP COL FSAR:

Supplemental Information

- LNP SUP 9.2-2

The applicant provided additional information in LNP 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 mechanical draft cooling towers for the LNP units can be adversely affected by interactions that exist between the SWS two mechanical draft cooling towers between units. In addition, interactions between cooling towers circulating water system (CWS) verses service water system) may adversely affect the cooling capacity of the SWS. Since LNP is utilizing mechanical induced-draft towers for the CWS verses natural draft cooling towers as submitted by other COL applicants, interactions on the SWS cooling towers is now more likely due to the lower in height of the discharge plume. Adverse interactions can occur due to localized atmospheric influences caused by siting considerations, the locations of major structures, the locations of the mechanical draft cooling towers, mechanical draft cooling tower fan speed, and wind effects. Because the certified AP1000 design is for only a single unit site and utilizes only one SWS mechanical draft cooling tower interaction effects between the mechanical draft cooling towers of multi-unit sites was not evaluated by the staff for the AP1000 DCD. Therefore, the staff requested in RAI Letter #50, Question 9.2.1-1 that the applicant revise FSAR Section 9.2.1 to address potential adverse interactions between the LNP mechanical draft SWS cooling towers and the mechanical draft CWS cooling towers for the two LNP units. Based on the applicant's response of July 6, 2009, the applicant indicated that approximately 900 feet of separation will exist between the SWS cooling towers of adjacent units and that the large turbine building structure is located between these two cooling towers. The applicant also indicated that greater than 1,200 feet of separation will exist between the units SWS cooling towers and the two mechanical induced-draft cooling towers for the CWS. The potential for adverse impacts on the SWS tower is further limited by site meteorological conditions. The SWS cooling towers are located so that the inclined directional wind vector would direct a tower plume away from the adjacent unit. Should site wind conditions exist that could direct the plume along the line of sight between the SWS cooling towers, the plume would still be required to navigate the interposing turbine building that separates the tower and large distance for an interface condition to occur. On this basis, the applicant concluded that there is minimal probability that a SWS cooling tower plume could travel to the vicinity of a SWS cooling tower on an adjacent unit. Also, there is a minimum probability that the CWS cooling tower plume would interact with the SWS cooling towers such that a significant degradation in performance would occur. In addition, the applicant stated that the FSAR will be revised to state that SWS cooling tower was evaluated for potential impacts from interference and air restriction effects due to yard equipment layout and tower operation on an adjacent unit and no adverse impacts were determined. Based on the information that was provided in the FSAR markup, the staff considers the licensee's response of this issue to be acceptable since the interactions between the cooling towers will be minimal and will not adversely affect the cooling capacity of the SWS. Therefore, RAI Letter #50 Question 9.2.1-1 is resolved and was incorporated into Revision 2 of the LNP COL FSAR.

9.2.1.5 *Post Combined License Activities*

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

9.2.1.6 *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 SWS, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP COL FSAR is acceptable and meets the guidelines given in Sections 9.2.1 and 9.2.5 of NUREG-0800. The staff based its conclusion on the following:

- LNP SUP 9.2-2 is acceptable because the design of the SWS cooling towers meets the guidance in Sections 9.2.1 and 9.2.5 of NUREG-0800, regarding adverse interactions between the SWS cooling towers on the LNP site.

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")

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 LNP COL FSAR, Revision 9, incorporates by reference, with no departures or supplements, Section 9.2.2, "Component Cooling Water System (CCS)," 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 LNP 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 LNP COL FSAR, Revision 9, 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 LNP 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 LNP COL FSAR, Revision 9, 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 LNP 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 LNP COL FSAR, Revision 9, 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 LNP COL FSAR Section 9.2.5, the applicant provided the following:

<u>*AP1000 COL Information Item*</u>

- LNP COL 9.2-1

The applicant provided additional information in LNP COL 9.2-1 to address COL Information Item 9.2-1 in LNP COL FSAR Sections 9.2.5.2.1, "General Description," and 9.2.5.3, "System Operation," 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 LNP 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 LNP COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the LNP COL FSAR:

AP1000 COL Information Item

- LNP COL 9.2-1

The applicant provided additional information in LNP 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 PWS included under Sections 9.2.5.2.1, 9.2.5.2.2, 9.2.5.3, 9.2.5.6 and 9.2.12.1 of the LNP COL FSAR. In these sections, the applicant proposes to use filtered water from the site well water subsystem of the RWS as the source of potable water. The PWS meets or exceeds the pressure, capacity, and quality requirements of the AP1000 DCD. The staff finds this an acceptable resolution of COL Information Item 9.2-1 because the applicant has ensured the potable water supply source and the pressure requirements from the AP1000 DCD are met. The AP1000 DCD 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 LNP COL 9.2-1 is outside the power block and not potentially contaminated by radioactive water. Therefore, the staff finds that GDC 60 is satisfied with respect to preventing contamination by radioactive water.

The staff's evaluation of control room habitability is addressed in Section 6.4 of this SER.

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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP COL FSAR is acceptable and meets the guidance in Section 9.2.4 of NUREG-0800. The staff based its conclusion on the following:

- LNP 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 LNP COL FSAR, Revision 9, 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 LNP COL FSAR Section 9.2.6, the applicant provided the following:

Departures

- LNP DEP 6.4-1

The applicant provided additional information in Section 9.2.6 of the LNP COL FSAR about LNP 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 LNP DEP 6.4-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.2 of this SER.

Supplemental Information

- LNP 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 sanitary waste, once treated, is combined with other plant discharge streams.

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 LNP 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 LNP 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 LNP COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the LNP COL FSAR:

Supplemental Information

- LNP 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 LNP COL FSAR. In Section 9.2.6.2.1 of the LNP COL FSAR, the applicant proposes to treat sanitary waste onsite. It is stated that the sewage treatment plant has sufficient capacity to treat waste from LNP Units 1 and 2. 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 LNP COL

FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP 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:

- LNP 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.

- LNP 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 LNP COL FSAR, Revision 9, 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.[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 LNP 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 the CWS. 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. TCS system piping is made of carbon steel, except that nonmetallic piping may be used.

9.2.8.2 *Summary of Application*

Section 9.2 of the LNP COL FSAR, Revision 9, 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 LNP COL FSAR Section 9.2.8, the applicant provided the following:

Site-Specific Information Replacing Conceptual Design Information

- LNP CDI

The applicant provided additional information to replace conceptual design information (CDI) in the AP1000 DCD with information identifying the source of cooling water for the LNP 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 LNP 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 LNP COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the LNP COL FSAR:

Site-Specific Information Replacing Conceptual Design Information

- LNP 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 LNP design specifies the use of only the circulating water for this purpose and raw water is not utilized for the TCS. This arrangement was reviewed and approved by the NRC during its evaluation of the AP1000 DCD. Consequently, the LNP 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 LNP TCS is acceptable.

LNP COL FSAR Section 9.2.8.2.2, “Component Description – Piping,” states that the TCS system piping is made of carbon steel and that piping and connections are welded, except where flange connections are used for accessibility and maintenance of components. Nonmetallic piping may be used. Since ASME B31.1, “Power Piping,” Appendix III states that nonmetallic piping is limited to 140 °Fahrenheit (F) (60 °Celsius (C)) and 150 pounds per square inch (psi) (1000 kilopascals (kPa)) in the water service application, the staff generated RAI Letter #54 Question 9.2.2-1 to ask if nonmetallic piping can be used based on the service conditions of the TCS.

The applicant responded to Question 9.2.2-1 on June 23, 2009, and stated that the TCS was reviewed during the AP1000 certification and the application of nonmetallic piping is under the design authority of Westinghouse. In addition, the applicant stated that Westinghouse Technical Report TR-103 (APP-GW-GLN-019), “Fluid System Changes” provides the following information on page 21 of 154, which address this RAI:

> As far as application of AP1000 systems, HDPE [High Density Polyethylene] may be used for systems and system areas of low pressure and low temperature. Based on manufacturer’s recommendations, HDPE will be used in systems with pressure up to 150 psi (1000 kPa) and temperature up to 140 °F (60 °C) for water service. Pressure and temperature limits for other services shall be based on the hazards involved, but in no application they shall exceed 150 psi (1000 kPa) and 140 °F (60 °C).

The applicant’s response addressed the staff’s concerns regarding the use of nonmetallic piping in the TCS service. The staff finds the response acceptable since nonmetallic material is limited up to 150 psi (1000 kPa) and temperatures up to 140 °F (60 °C); therefore, RAI Letter #54 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 LNP COL FSAR related to this section. The results of the NRC staff’s technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP 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:

- LNP 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) collects and processes the waste water from the equipment and floor drains in the nonradioactive building areas during plant operations and outages. The 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 wastewater that collects in the retention basins is routed to the Crystal River Energy Complex (CREC) discharge canal through the CWS blowdown.

9.2.9.2 *Summary of Application*

Section 9.2 of the LNP COL FSAR, Revision 9, 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 LNP COL FSAR Section 9.2, the applicant provided the following:

AP1000 COL Information Item

- LNP COL 9.2-2

The applicant provided additional information in LNP 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 LNP 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 LNP COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the LNP COL FSAR:

AP1000 COL Information Item

- LNP COL 9.2-2

The applicant provided additional information in LNP 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 LNP 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 LNP COL FSAR. To address LNP COL 9.2-2, details were provided for the location of the WWRB and routing configuration.

The wastewater from the retention basin is routed to the CREC discharge canal through the CWS blowdown. The staff performed an initial review of Section 9.2.9 of the LNP COL FSAR and determined that the description of wastewater routing and components was insufficient. To address the COL items, additional information was needed before for the staff could review the adequacy of the site-specific wastewater retention basin and associated components.

In order to meet GDC 60, the applicant was asked to demonstrate suitable control of the release of radioactive materials in liquid effluent. The staff requested the applicant in Letter #51 related to RAI 9.3.3-1, to describe how the potentially radioactive effluents draining into the water basin will be monitored and justify the absence of water level instrumentation and radiation monitoring in the WWRB. The staff also requested the applicant provide additional details on the routing of water and a description of the associated components (i.e., transfer pumps, size of basin, etc.) as requested in COL Information Item 9.2-2.

The applicant responded to the staff's request in a letter dated June 23, 2009. The response provided additional information on radiation monitoring, level instrumentation and components for the WWRB.

The WWRBs are located southwest of LNP 1 and 2 near the sewage treatment plant. One basin is provided per unit. For redundancy, each unit is provided to intake the maximum possible flow from two units if one basin is out of service. The basins are constructed of reinforced concrete walls and continuously poured base mats with no construction joints in the mats or any exterior walls (except a construction joint with a waterstop may be used at the exterior wall/mat junction) and waterstops at all construction joints to minimize seepage. The size of the basins provides retention time for settling of solids larger than 10 microns that may be suspended in the wastewater stream.

Two 100 percent pumps for each retention basin are provided to transfer water from the WWRB to the CWS blowdown. For each retention basin, only one of the pumps will operate at any given time. The pumps will have separate feeds from the 480 volts alternating current (VAC) distribution system. In the event of a loss of offsite power (LOOP), power will not be supplied to the WWRB transfer pumps. The basin transfer pumps are designed to discharge a maximum of 850 gallons per minute (gpm) to the CWS blowdown.

The applicant confirmed that fluids discharging into the retention basin are either monitored with radiation monitoring instrumentation or preclude interconnection with systems containing radioactive fluids. The applicant further clarified that a radiation monitor will be installed on the common discharge of the basin transfer pumps and will provide an alarm and trip the basin transfer pumps upon detecting radioactivity in the waste water.

To protect against flooding, a level indicator and level transmitter are provided for each WWRB to automatically control flow out of the WWRB. High alarms will indicate basin level where operator action is required.

The applicant's response to RAI 9.3.3-1 above and subsequent incorporation into Section 9.2.9 of the LNP COL FSAR is acceptable to the staff. The LNP COL FSAR adequately addresses COL Information Item 9.2-2. Therefore, RAI 9.3.3-1 is resolved.

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 inadvertent release of radioactive materials by preventing the inadvertent transfer of contaminated fluids to system portions for noncontaminated drainage that could result in radioactive release to the environment.

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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information

incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP 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:

- LNP COL 9.2-2 is acceptable because the staff finds that the relevant information in the LNP 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 LNP COL FSAR, Revision 9, 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 LNP COL application are documented in NUREG-1793 and its supplements.

9.2.11 Raw Water System

9.2.11.1 *Introduction*

The RWS is a nonsafety-related system that consists of two subsystems; the RWS freshwater and saltwater subsystems. The RWS freshwater subsystem pumps water from ground water wells and the saltwater subsystem supplies water from the Cross Florida Barge Canal (CFBC), for use by the LNP units.

The RWS saltwater subsystem supplies raw (unprocessed) water for make-up to the CWS mechanical draft cooling tower basins. In addition, the unprocessed water is used for water for the make-up strainer backwash and for the screen wash pump suction source. The RWS saltwater subsystem supply pumps can also be used to provide alternate dilution flow for the liquid waste discharge when cooling tower blowdown is not available for the discharge path. Only the RWS saltwater subsystem is shared by the two LNP units through cross ties.

The RWS freshwater subsystem provides water from the ground water wells for make-up to the SWS cooling tower basins, the demineralizer water treatment system (DTS), PWS, and the fire protection system (FPS) fire water storage tanks. The RWS freshwater subsystem also provides the water for the strainer backwash and the media filter backwashes and an alternate make-up for the SWS via the secondary fire water storage tank clearwell to the cooling tower basin. The SWS cooling tower basins rely upon make-up from the RWS freshwater subsystem in order to achieve and maintain cold shutdown conditions.

9.2.11.2 *Summary of Application*

Section 9.2.11 of the LNP COL FSAR, Revision 9, provides information concerning the RWS design basis, system description, system operation, safety evaluation, tests and inspections, and instrumentation. The RWS was referred to in the AP1000 DCD in relation to the CWS, SWS, DTS, and FPS, but an 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 LNP units and consequently, the applicant has provided a complete description of this system in the LNP COL FSAR for the LNP units.

In LNP COL FSAR Section 9.2.11, the applicant provided the following:

Interface Requirements

The plant interfaces for the RWS are identified in Table 1.8-203 of the LNP COL FSAR as Item 9.4, “Plant makeup water quality limits,” and Item 9.5, “Requirements for location and arrangement of raw and sanitary water systems.” These items are identified as “non-nuclear safety (NNS)” interfaces.

Supplemental Information

- LNP SUP 9.2-1

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*

Because the RWS was not considered within the scope of the AP1000 DCD, a regulatory basis for this system was not established for the standard plant design. The regulatory basis of the RWS for the LNP units is provided in this section.

The acceptance criteria that pertain to CWS and RWS evaluations are given in NUREG-0800, Sections 10.4.5, “Circulating Water System”; 9.2.1, “Station Service Water System”; 9.2.5, “Ultimate Heat Sink”; 3.4.1, "Flood Protection"; and 3.5, "Barrier Design for Missile Protection."

The regulatory bases and guidance for acceptance of the supplemental information and interface items are established in:

- GDC 2, “Design Basis for Protection Against Natural Phenomena”
- GDC 4

- 10 CFR 20.1406, "Minimization of Contamination"
- RG 1.29, "Seismic Design Classification," Position C2

9.2.11.4 *Technical Evaluation*

The staff reviewed the information provided in Section 9.2.11 of the LNP COL FSAR that describes the RWS for the LNP units, including the information provided by Figure 9.2-201, "Raw Water System Flow Diagram." 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 results of the NRC staff's evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The remainder of this SER section evaluates both LNP SUP 9.2-1 and Interface Items 9.4 and 9.5.

A. GDC 2, GDC 4, and RG 1.29

The staff's review of the information in LNP COL FSAR Section 9.2.11 is 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," states 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 is needed to adequately describe the consequences of RWS failures and to explain why safety-related SSCs are not affected. Likewise, additional information is needed to explain why a failure of the RWS will not adversely affect RTNSS systems and components or impact the control room occupants. Because the applicant did not identify and address these considerations, the staff is unable to confirm compliance with GDC 2, GDC 4, conformance with the guidance in RG 1.29, Position C.2, and passive plant policy considerations and passive plant policy considerations. The staff requested in RAI Letter #52 Question 9.2.1-2 that the applicant revise Section 9.2.11 to address the impact of RWS failures accordingly, including development of plant-specific ITAAC and test program specifications as appropriate.

In its response dated July 22, 2009, the applicant provided a detailed response to the GDC 2, GDC 4, ITAAC and testing questions. A summary of the applicant's response is described below.

> The applicant stated that failure of the RWS piping located in the yard and inside the turbine building were considered.
>
> The LNP RWS consists of two subsystems, a freshwater subsystem that supplies groundwater for make-up to the DTS, PWS, FPS fire water storage tanks, and SWS cooling tower basins; and a saltwater subsystem that supplies water from the CFBC for make-up to the CWS mechanical draft cooling tower basins. The potential failures of the two RWS subsystems and the corresponding impact on SSCs that are safety-related or AP1000 equipment Class D are described below.

For failure of RWS piping in yard areas, the saltwater subsystem of RWS does not directly interface with any safety-related system, but only interfaces with CWS. The piping is routed underground from the intake structure on the CFBC to the CWS cooling tower basin. The only above ground portions of the RWS saltwater subsystem are at the intake structure and at the CWS cooling tower basin. This piping is not routed in close, proximity to any safety-related SSCs. DCD Section 3.4.1.1.1 indicates that a failure of the CWS cooling tower, the SWS piping, or the CWS piping could result in a potential flood source. However, these potential sources are located far from safety-related structures and the consequences of a failure in the yard would be enveloped by the analysis described in DCD Section 10.4.5 for failure of the CWS. Site grading will carry water away from safety-related or AP1000 Class D SSCs.

For failure of RWS piping in yard areas, the freshwater subsystem of RWS interfaces with DTS, FPS, PWS, and SWS, none of which are safety-related systems. The piping for the freshwater subsystem is routed underground from the wells to the well water storage tanks and from the media filters to the points of interface with the other systems. This piping is not routed in close proximity to any safety-related SSCs. The only RTNSS system in close proximity to this subsystem is the SWS. Because of the significant difference in system capacities, a resultant flood from a break in the RWS freshwater subsystem piping is bounded by the analysis for a break in the CWS piping.

For failure of RWS piping inside the turbine building, the RWS freshwater subsystem piping is routed outside in the yard area and inside the turbine building to the interface points with the SWS and DTS systems. The RWS-to-DTS interface is upstream of the DTS filters and DTS feed pumps. The primary source of flooding would be from the RWS water that discharges through the break prior to securing the raw water booster pumps. A break in the RWS piping to the DTS or the SWS is bounded by a break in the CWS piping. As discussed in DCD Section 3.4.1.2.2.3, the bounding flooding source inside the turbine building is a break in the CWS piping. Flow from any postulated pipe failures above DCD elevation 100'-0" (NGVD29 elevation 52'-0") would travel down to elevation 100'-0" via floor gratings and stairwells. There is also no safety-related equipment in the turbine building. The CCS and SWS components on elevation 100'-0", which provide RTNSS support for the normal residual heat removal system (RNS) are expected to remain functional following a flooding event in the turbine building since the pump motors and valve operators are above the expected flood level. Therefore, failure of the RWS piping within the turbine building will not adversely impact any safety-related or RTNSS SSCs.

The RWS-to-SWS interface and the effects of RWS failure is as follows:

The RWS to SWS interface is at the SWS make-up control valve V009, as shown in DCD Figure 9.2.1-1. The SWS piping is routed from the control valve V009 to the top of the SWS cooling tower basin. There is an air gap between the SWS cooling tower basin water level and the discharge. The air gap ensures any break upstream of the raw water make-up water path will not result in the draining of the SWS cooling tower basin.

No chemical treatment is anticipated for the LNP RWS freshwater subsystem make-up to SWS. Therefore, there are no chemical releases associated with RWS that could adversely impact control room habitability.

Section 2.4.13 of the LNP COL FSAR presents a conservative analysis of the effect of an accidental release of liquid effluents to the ground water environment through the postulated failure of the liquid waste system effluent holdup tank. A substantial release directly to the Floridan aquifer is unlikely. However, the impact on public and private water use was examined should such a release occur. LNP COL FSAR Table 2.4.13-205 shows bounding activity concentrations that could occur at the nearest private or public well 2 kilometers (km) (1.2 miles (mi)) from the LNP site. With the exception of tritium, the maximum activity concentration for each radionuclide at the closest well is negligible compared to the nuclides' effective concentration limit (ECL). The maximum activity concentration of tritium is less than 0.7 percent of its ECL. Therefore, the accidental release of effluents to groundwater results in effective dose equivalents that are very small fractions of the limits in 10 CFR Part 20, “Standards for protection against radiation,” for water supplies derived from groundwater aquifers.

The RWS has no interconnection with any system that contains potentially radioactive fluids. The RWS operates at a higher system pressure than those systems with which it directly interfaces (at the point of interface) and, therefore, in-leakage is not feasible. Thus, the possibility of releasing radioactivity from the RWS is remote.

Failure of the RWS or its components will not affect the ability of any other safety-related systems to perform their intended safety functions nor will it adversely affect any RTNSS 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, or result in a release of radioactivity to the environment. Because the RWS is not safety-related and its failure does not lead to the failure of any safety-related systems, the requirements of GDC 2, GDC 4, and the guidance of NUREG-0800 Section 9.2.1 regarding safety-related systems, do not apply. Further, the applicant stated that RWS piping and structures are designed and constructed in accordance with nationally recognized codes and standards (such as ASME B31.1 and American Water Works Association (AWWA). Design features have been included (such as the use of buried piping and power supply redundancy) to ensure RWS is reliable and will be available to support normal plant operation and shutdown functions.

As noted in FSAR Section 14.3.2.3.3, this site-specific system (RWS) does not meet the ITAAC selection criteria. ITAAC screening was performed for the RWS, using the screening criteria of FSAR Section 14.3.2.3, which concluded that ITAAC is not applicable as indicated in FSAR Table 14.3-201.

No specific Technical Specifications are required for the RWS and none are applicable.

Technical Specifications for the AP1000 are provided in FSAR Chapter 16, DCD Section 16.1, and were evaluated by the NRC in NUREG-1793, Chapter 16.

There are no availability controls for the RWS and they are not required based on the RTNSS evaluation in NUREG-1793, Chapter 22 and Westinghouse Commercial Atomic Power (WCAP)-15985, "AP1000 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process," Revision 2. Also, FSAR Chapter 16 and DCD Chapter 16 do not identify any availability requirements for the RWS.

The applicant stated that no change to the FSAR is proposed as a result of this response related to GDC 2 or GDC 4. The revised FSAR Section 9.2.11 is provided as part of the response to Question 9.2.1-3 and addresses the information discussed in the response to this question as appropriate, consistent with NRC guidance provided in RG 1.206, Section C.III.

The staff determined that failure of the RWS or its components will not affect the ability of any safety-related systems to perform their intended safety function 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 a postulated pipe break will not reach any safety-related equipment or result in injury to occupants of the control room 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, the staff has determined that appropriate testing of the RWS was addressed in LNP COL FSAR Section 14.2. 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 and the guidance in RG 1.29 have been satisfied; therefore, Question 9.2.1-2 is resolved.

The staff has evaluated the RWS intake structure described in LNP COL FSAR Section 9.2.11.2.2, "Component Description," and concluded that the failure of the intake structure would not impact the ability of safety-related systems to perform their intended functions.

B. Cold Shutdown

The RWS is relied upon for achieving and maintaining cold shutdown conditions, which (in addition to the passive plant policy considerations discussed in SECY-94-084) is necessary for satisfying the 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 does 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 make-up for the two LNP 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 is unable to confirm that the cooldown and policy considerations that apply to passive plant designs are satisfied. The staff requested in RAI Letter #52 Question 9.2.1-3 that the applicant revise LNP COL FSAR Section 9.2.11 adequately define and address RWS design-bases considerations with respect to its cooldown function, and to develop plant-specific ITAAC and initial test program specifications as appropriate.

In a response dated July 22, 2009, the applicant stated the following:

> RWS consists of two subsystems. The freshwater subsystem provides a continuous supply of groundwater for several plant services including make-up to the DTS, PWS, the FPS fire water storage tanks, and SWS cooling tower basins. The saltwater subsystem supplies water from the CFBC for fill and make-up to the CWS mechanical draft cooling tower basins.
>
> This response specifically focuses on the RWS interface with the SWS because, as noted in the response to Question 9.2.1-2, the other functions performed by RWS do not have a direct interface with any other system identified as safety-related, designated for RTNSS, or designated as AP1000 Class D.
>
> RWS provides a water fill/make-up function for the SWS. The SWS has investment protection short-term availability controls as described in 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, SWS is directly providing active core cooling and, as noted in the response to Question 9.2.1-2, was evaluated by Westinghouse 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, as discussed in response to Question 9.2.1-2, the 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 event of a failure of RWS to provide adequate make-up flow to the SWS cooling tower basins during the short time period in which 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, RWS is not a RTNSS system or subject to investment protection short-term availability controls. However, 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 included in the operating procedures, where appropriate.
>
> In DCD Section 5.4.7.1.2.1, the applicant describes that the RNS, in conjunction with its associated support systems, the CCS and 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 SWS. During a normal plant cooldown, the RNS and CCS reduce the temperature of the RCS from approximately 121 °C (350 °F) to approximately 51.6 °C (125 °F) within 96 hours after shutdown. Each unit's RWS is designed to provide ample make-up flow during these conditions using the RWS pumps. The two raw water well pumps provide approximately 3936 liters/minute (1,040 gpm) each from the aquifer to the raw water storage tank, and the four raw water booster pumps provide 1892 liters/minute (500 gpm) each from the raw water storage tank. The SWS design make-up flow is approximately 3142 liters/minute (830 gpm).

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 the 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.

An ample inventory of water is available to provide make-up to the SWS cooling tower basins. As noted in FSAR Section 2.4.12.2.4, as of 2005, Southwest Florida Water Management District (SWFWMD) had permitted approximately 83.133 million liters per day (mld) or 21.956 million gallons per day (mgd) of nondomestic groundwater use in the portion of LNP that falls within the SWFWMD. Approximately only 29.061 mld (7.677 mgd) or permitted capacity was used (total water demand, which includes unpermitted domestic demands, was 35.942 mld (9.495 mgd). As stated in FSAR Section 2.4.1.1, an estimated average of 4.805 mld (1.269 mgd) and a maximum of approximately 22.139 mld (5.848 mgd) of groundwater will be used at the LNP site. Therefore, the groundwater usage at the LNP site will not result in a total groundwater use greater than that already permitted by the SWFWMD and thus, there is sufficient capacity to support cooldown to cold shutdown conditions and maintain the station in Mode 5 for greater than 7 days.

The lack of designation of RWS as RTNSS or Class D indicates there is no performance requirement for the system during a loss of normal ac power or in the event of a single active failure. Nonetheless, RWS is highly reliable based on its design, and a single failure of an active component in RWS would not affect normal plant cooldown. Make-up flow to the CWS is not normally required after the plant is shutdown; therefore, the RWS make-up pumps do not need to operate during a loss of normal ac power to cool the plant down during this event. Each raw water well pump and raw water booster pump can deliver make-up flow to the SWS cooling tower basins to meet demand during all normal modes of operation. Failure of an operating pump, discharge valve, or strainer would not prevent the RWS from providing make-up to the SWS cooling towers. The raw water well and booster pumps, discharge valves, and automatic strainer are powered from the normal ac power system and have a back-up power supply from the diesel generators. In the event of a loss of normal ac power, the components are manually loaded onto the appropriate diesel bus and are manually started by the operator. Only one raw water well pump and one booster pump can be loaded on a diesel generator at a time. The valves associated with flow to the four 50 percent media filters fail in a position to provide continuous RWS filtered flow. The RWS pump discharge valves have handwheels to manually adjust the RWS flow as required. Twenty-four hours after a loss of normal ac power, the make-up requirement is 1014 liters/minute (268 gpm) (with blowdown reestablished), well within the capacity of one 1892 liters/minute (500 gpm) RWS booster pump. The RWS, therefore, continues to maintain the capability to provide make-up water to the SWS cooling tower basins during the loss of normal ac power events even with a single active failure of one standby diesel. The underground RWS piping will be HDPE, which is not susceptible to corrosion. Therefore, periodic inspections of the underground RWS piping are not required.

The saltwater subsystem intake bays at the intake structure will be inspected periodically for silt buildup and cleaned as necessary based on operating experience from other Progress Energy plants. Equipment that remains idle for extended periods of time (pumps, traveling screens, strainers) will be operated periodically in accordance with vendor recommended maintenance practices.

In the 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 RWS is shown to not be a RTNSS system nor subject to investment protection short-term availability controls. Neither 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.

As a follow-up to the applicant's response to Question 9.2.1-3, the staff requested additional clarification in the FSAR as stated in RAI Letter #67, Question 9.2.1-6 regarding: a) saltwater subsystem cross-tie between Units 1 and 2 and GDC 5, "Sharing of Structures, Systems and Components"; b) power supplies (backup) for raw water strainer; c) raw water storage capability; d) booster pump controls or interlocks; and e) system materials. In a letter dated October 22, 2009, the applicant provided the following response:

a) Saltwater subsystem cross-tie discussion: The applicant stated that it is correct that FSAR Figure 10.4-201 describes the CWS and the saltwater subsystem of the RWS and indicates a cross-tie exists between Units 1 and 2. However, as noted in Section 9.2.11.2.1 of the FSAR, the RWS is shown in Figures 9.2-201 (freshwater subsystem) and 10.4-201 (saltwater subsystem). The RWS freshwater subsystem supplies strained and filtered groundwater for makeup to three plant systems and to the service water cooling tower basins. There is no cross-tie between the two units for the RWS freshwater subsystem. The functions of the RWS, other than the SWS makeup, do not have a direct interface with any other system identified with the AP1000, which is safety-related, designated for RTNSS, or designated as AP1000 Class D.

Criterion 5 of 10 CFR Part 50, Appendix A, states that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units. Because only the freshwater subsystem of RWS has a direct interface with any system that is safety-related, RTNSS, or designated as AP1000 Class D, and because the freshwater subsystem of RWS has no cross-tie between units, including a discussion of the RWS system cross-tie in the FSAR is not required.

The staff finds item ‘a’ acceptable since the cross-tie between the saltwater subsystem is not considered important to safety; therefore, GDC 5 does not apply. The freshwater subsystem of RWS has no cross-tie between units. In addition, the freshwater subsystem of RWS has no direct interface with any system that is safety-related, RTNSS, or designated as AP1000 Class D.

b) Strainer power supplies: The applicant stated that the strainer is designed to fail “as-is” under loss of power and does not need a back-up power supply. There is also a bypass line around the strainer with a normally closed manual valve that can be operated in the unlikely event the strainer becomes fouled during loss of normal power. A differential pressure transmitter is installed across the strainer and will alert operators if the strainer becomes fouled.

   As noted in the response to Question 9.2.1-3, the entire FSAR Section 9.2.11 is being revised. Section 9.2.11.2.2.1, “Valves,” states the RWS makeup water pump discharge valves have a backup power feed from the diesel generators. A clarification will be made to the revised FSAR Section 9.2.11.2.2.1, “Automatic Self-Cleaning Strainer,” that the strainer is designed to fail as-is during loss of normal ac power.

The staff finds item ‘b’ acceptable since the strainer can be operated manually on its bypass if the strainer becomes fouled during a loss of normal power, thus the functional loss of the strainer has no negative effects for the RWS.

c) Raw water storage capability: As noted in FSAR Section 9.2.11.2.2.1, “Raw Water Well Pumps,” two 100 percent capacity well pumps for each unit are supplied. Only one of those pumps is designed to operate at a time. Both pumps can be manually loaded onto the standby diesel generator bus although only one can be loaded at a time. Thus, sufficient redundancy is provided in the system design to reasonably expect at least one of the well pumps will be available to supply makeup water in the event of a loss of normal ac power. For this reason, it is not necessary to postulate impacts to the SWS cooling tower basin supply if both well pumps are not available to support cooldown. Minimum dimensions for the raw water storage tank are provided in FSAR Section 9.2.11.2.2.1, “Raw Water Storage Tank,” which is 9.1 meters (30 feet) in diameter x 9.1 meters (30 feet) tall.

The staff finds item ‘c’ acceptable since either of the two freshwater pumps can be manually loaded onto the emergency power supply generator; thus, sufficient redundancy exists for the RWS to support plant cold shutdown. In addition, the staff concludes that the raw water storage tank has a capacity to hold approximately 567,000 liters (150,000 gallons) of raw water to support plant operations to support cold shutdown.

d) Booster pump controls or interlocks: There are no automatic booster pump controls or interlocks associated with the raw water storage tank level. As noted in FSAR Section 9.2.11.6.1, a level control system in the tank provides automatic start and stop control for the raw water well pumps. Normally, one well pump is in operation. The level control system starts the second well pump at very low tank levels and stops the pump when 50 percent level is established in the tank. Because the capacity of the well pumps is approximately double that of the booster pumps, filling of the storage tank by the well pumps occurs more quickly than emptying the tank by the booster pumps.

   In addition, a redundant level transmitter on the raw water storage tank will provide continuous level indication and input to a low level alarm in the main control room (MCR). The low level alarm setpoints for the diverse level instrumentation ensure the operators are informed of an abnormal low level before the minimum net positive suction head (NPSH) requirements for the booster pumps are reached. This will allow plant operators to promptly detect low level in the tank and initiate corrective action as needed. This description will be added to the FSAR.

The staff finds item ‘d’ acceptable since the raw water storage tanks is maintained full by the level control systems that controls the well pumps and low level alarm setpoints for the storage tank ensures the operators are informed of an abnormal low level before the minimum NPSH requirements for the booster pumps are reached.

e) System materials: The FSAR will be revised to reflect the use of HDPE piping in the buried portions of the RWS system. However, the applicant has not discussed the ASME Code for power piping for the RWS in the text of the FSAR. In a telephone call, the applicant agreed to include the RWS power piping code in the FSAR in a revision to this response.

The staff finds item ‘e’ acceptable since buried HDPE will be designed and installed in accordance with industry Codes such as ASME B31.1 and 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 (biofouling) 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.

In summary, the staff finds that the RWS is designed with the provision to protect against single failure since many of the freshwater subsystem RWS components can be supplied with backup power from the onsite diesel generators as necessary or operated locally. During a loss of station power, RWS 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 and other well water subsystem components. Since the RWS is not safety-related and its failure does not lead to the failure of any safety-related systems, the staff concludes that the RWS system design is consistent with the guidance in SECY-94-084; therefore, Question 9.2.1-3 and Question 9.2.1-6 are resolved. All associated FSAR markups provided by the applicant have been incorporated into Revision 2 of the COL FSAR.

C. Regulatory Treatment of Nonsafety-Related System

The RWS supports the SWS cooling function by providing make-up 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 evidently 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 is the case. Also, because the SWS cooling tower basins are very limited in their capacity, it was not clear why the RWS make-up is not required for this situation. Consequently, the staff requested in RAI Letter #52 Question 9.2.1-4 that the applicant revise Section 9.2.11 to explain why the RWS make-up 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 this question dated July 22, 2009, the applicant 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 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 applicant stated that 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. In addition, the applicant stated that neither the 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 recognized throughout the AP1000 DCD and NUREG-1793.

The staff finds the applicant’s response to Question 9.2.1-4 acceptable since: 1) the RWS was previously evaluated in WCAP-15985 in Table 1-1, “Nonsafety-related System Evaluation in AP1000 RTNSS Process,” which was previously approved by the staff; 2) the RWS does not directly provide core cooling; 3) the RWS has no direct RTNSS applicability; and 4) the RWS has adequate stored water within the SWS cooling towers and fire water tank for more than 24 hours to support the SWS RTNSS functions. The 24 hours stored on site water supply provides ample time to restore RWS makeup flow or take the procedural actions necessary to exit the condition of applicability for the SWS and its RTNSS function. Therefore, Question 9.2.1-4 is resolved.

D. System Design Consideration

As stated in LNP COL FSAR Section 9.2.11.4, the liquid waste stream effluent is released offsite through a dilution flow stream. Dilution flow is routed from the RWS to the CWS cooling tower blowdown during shutdown conditions. During normal operation, the CWS circulating water pumps provide dilution flow to the cooling tower blowdown pipe. Contamination of the RWS is not possible since the liquid waste stream effluent enters the blowdown pipe downstream of the RWS interface.

As specified by 10 CFR 20.1406, COL applicants are required to describe how the 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.

Therefore, the staff requested in RAI Letter #52 Question 9.2.1-5 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.

In its response dated July 22, 2009, the applicant stated that the RWS has no interconnection with any system that contains potentially radioactive fluids as shown on FSAR Figures 9.2-201 and 10.4-201. The RWS operates at a higher system pressure than the SWS and CWS, systems that it directly interfaces with (at the point of interface). Therefore in-leakage is not feasible. 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 ground water monitoring program is described in LNP COL FSAR Section 12AA.5.4.14. The staff's evaluation of the groundwater monitoring program is provided in Chapter 12 of this SER. Because there is no interconnection with any system that contains potentially radioactive fluids as indicated in LNP COL FSAR Section 9.2.11.1.1, the staff concludes that the requirements of 10 CFR 20.1406 are satisfied.

The staff finds that the applicant adequately addressed the 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. The staff considers the applicant's resolution of this issue to be acceptable, and Question 9.1.2-5 is resolved.

To address the fire protection interface with the RWS, the applicant states that the freshwater subsystem is filtered by media filters before being delivered to the fire water tanks; therefore, the staff finds this acceptable because it ensures that the FPS is appropriately maintained with respect to the interface with the RWS. The staff's evaluation of the FPS is included in Section 9.5.1.

Based on the above technical evaluation, the NRC staff finds acceptable the information added to the LNP COL FSAR to address LNP SUP 9.2-2 and Interface Items 9.4 and 9.5.

9.2.11.5 *Post Combined License Activities*

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

9.2.11.6 *Conclusion*

The NRC staff has evaluated the RWS as described in LNP COL FSAR Section 9.2.11. The staff's evaluation focused primarily on confirming that: (a) the design of the RWS complies with the requirements of GDC 2 and GDC 4 and conforms with the guidance in RG 1.29; (b) the RWS reliance for the support of SWS for achieving and maintaining cold shutdown conditions and RTNSS considerations is consistent with the guidance in SECY-94-084; (c) the RWS is not considered RTNSS; (d) other system design considerations meet the requirements of 10 CFR 20.1406; and (e) the interaction with the FPS has been properly evaluated.

Based upon the results of this evaluation, the staff concludes that the LNP RWS, as described under LNP SUP 9.2-1 in Section 9.2.11 of the LNP COL FSAR, is acceptable.

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 LNP COL FSAR, Revision 9, 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 LNP COL FSAR Section 9.3, the applicant provided the following:

Departures

- LNP DEP 6.4-2

The applicant provided additional information in Section 9.3.1.1.2 of the LNP COL FSAR about LNP 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 LNP 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 LNP 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 LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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:

> *AP1000 COL Information Item*
>
> - *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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP COL FSAR is acceptable and meets the guidelines given in Section 9.3.1 of NUREG-0800.

- LNP 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 LNP 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 LNP COL FSAR, Revision 9, 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 LNP 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 Post Accident 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 LNP COL FSAR, Revision 9, 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 LNP 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 Post Accident 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 LNP COL FSAR, Revision 9, 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 LNP 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 LNP COL FSAR, Revision 9, 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 LNP 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 LNP COL FSAR, Revision 9, incorporates by reference, Section 9.3.6, "Chemical and Volume Control System," of Revision 19 of the AP1000 DCD. In addition, in the LNP COL FSAR, the applicant provided the following:

Departures

- LNP DEP 7.3-1

The applicant provided additional information in Section 9.3.6 of the LNP COL FSAR about LNP 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 LNP 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 LNP 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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP 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 LNP COL FSAR, Revision 9, 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 LNP COL FSAR Sections 9.4.1, 9.4.1.4, and 9.4.12, the applicant provided the following:

Departures

- LNP DEP 6.4-1

The applicant provided additional information in Section 9.4.1 of the LNP COL FSAR about LNP 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 LNP DEP 6.4-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.2 of this SER.

- LNP DEP 6.4-2

The applicant provided additional information in Section 9.4.1 of the LNP COL FSAR about LNP 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 LNP 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 LNP COL FSAR Section 9.4.12, the applicant provided the following:

- LNP COL 9.4-1b

The applicant provided additional information in LNP 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 LNP 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 LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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:

AP1000 COL Information Items

- *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.

- LNP COL 9.4-1b

The applicant provided additional information in LNP 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 LNP 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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP 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:

- LNP 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.

- LNP 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.

- LNP 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 LNP COL FSAR, Revision 9, 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 LNP 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 LNP COL FSAR, Revision 9, 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 LNP 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 LNP COL FSAR, Revision 9, 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 LNP 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 LNP COL FSAR, Revision 9, 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 LNP COL FSAR Section 9.4.7.4, the applicant provided the following:

AP1000 COL Information Item

- 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 LNP 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 LNP 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 LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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:

<u>AP1000 COL Information Item</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 9.4-1 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.

<u>Correction of Error in the Standard Content Evaluation Text</u>

The NRC staff identified an error in the text reproduced above from Section 9.4.7.4 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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP 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 LNP COL FSAR, Revision 9, 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 LNP 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 LNP COL FSAR, Revision 9, 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 LNP 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 LNP COL FSAR, Revision 9, 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 LNP 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 LNP COL FSAR, Revision 9, 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 LNP 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 LNP COL FSAR, Revision 9, 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 LNP COL FSAR Section 9.5.1, the applicant provided the following:

Departures

- LNP 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 LNP COL FSAR Table 9.5.1-201, providing additional information about LNP 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 LNP DEP 6.3-1 information appearing in other chapters of the LNP COL FSAR, is reviewed in Section 21.1 of the 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 LNP 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 LNP 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.

- LNP COL 9.5-1

The applicant provided additional information in LNP COL 9.5-1 to resolve COL Information Item 9.5-1 regarding applicant-specific aspects for the qualification requirements for the fire protection program.

- LNP COL 9.5-2

The applicant provided additional information in LNP 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 LNP 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.

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 LNP 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 LNP 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, LNP COL 9.5-1, and LNP 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 basis for acceptance of STD SUP 9.5-1 includes the following:

- RG 1.189

9.5.1.4 *Technical Evaluation*

The NRC staff reviewed Section 9.5.1 of the LNP 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 FPS. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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[3] from Section 9.5.1.4 of the VEGP SER:

> *Supplemental Information*
>
> - *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.*
>
> *AP1000 COL Information Items*
>
> - *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 fire protection program, training of firefighting personnel,*

[3] Only the BLN SER text relevant to LNP 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 LNP, it was not reproduced here. Also, the discussion of LNP COL 9.5-2 (corresponds to BLN COL 9.5-2) was moved to the end of this technical evaluation section.

> *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.

Correction of Error in the Standard Content Evaluation Text

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.

Resolution of Standard Content Confirmatory Item 9.5-1

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.

Proposed License Condition 3, Item C.2

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.

AP1000 COL Information Items

- LNP COL 9.5-1

The applicant provided additional information in LNP COL 9.5-1 to resolve COL Information Item 9.5-1 for plant-specific fire protection issues. These plant-specific issues include:

- The responsibilities of the engineer in charge of fire protection and his staff.

- The organization of the fire brigade.

- The engineer in charge of fire protection is responsible for the formulation and implementation of the fire protection program and meets the qualification requirements listed in LNP COL FSAR Section 13.1.2.1.4.9.

The NRC staff compared the plant-specific fire protection issues under LNP COL 9.5-1 with the subject matter addressed by the standard content evaluation of STD COL 9.5-1, as detailed above. The staff concludes that the issues addressed by LNP COL 9.5-1 are included in the subject matter addressed by the staff in its evaluation of STD COL 9.5-1 and, therefore, concludes LNP COL 9.5-1 conforms with the regulatory positions in RG 1.189 regarding the implementation of the LNP Fire Protection Program.

- LNP COL 9.5-2

The applicant provided additional information in LNP 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.

This 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 analysis as required by LNP COL 9.5-2 related to the site-specific fire protection information included under Section 9.5.1.9.2 and Section 9A.3.3 of the LNP 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 Site-Specific RAIs

In addition to the review of the standard content, the staff also reviewed LNP site-specific content and issued letters 6 and 7 that requested site-specific RAIs, RAIs 9.5.1-1 and 9.5.1-2, related to the filtering and chemical treatment of the fire water supply system and qualifications of the engineer in charge of fire protection, respectively.

In its response dated February 19, 2009, to the site-specific RAI related to the filtering and chemical treatment of the fire water supply system to prevent or control bio-fouling or microbiologically-induced corrosion of the fire water system, the applicant revised FSAR Section 9.2.11.3.2 to state that chemical injection points are provided to treat the raw water supply to the FPS fire water storage tanks with sodium hypochlorite. Effectiveness of the treatment is monitored by periodic sample inspections of the wetted portions of the FPS headers. Based on the above, the staff finds the applicant has adequately provided a program for maintaining an adequate level of quality for the fire protection water system in accordance with RG 1.189 and, therefore, is acceptable.

In its response to the site-specific RAI related to the qualifications of the engineer in charge of fire protection, the applicant revised FSAR Section 13.1.2.1.4.9 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. In addition, this FSAR section states that in accordance with RG 1.189, the engineer in charge of fire protection is a graduate of an engineering curriculum of accepted standing and has completed not less than 6 years of engineering experience, three of which were in a responsible position in charge of fire protection engineering work. Based on the above, the staff finds the description of the fire protection engineer qualifications is in accordance with RG 1.189 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 FPS, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP 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:

- LNP 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 LNP 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.

- LNP COL 9.5-1, addressing the plant-specific issues for the fire protection program at LNP, is adequately addressed by the applicant and is resolved.

- LNP COL 9.5-2, addressing the site-specific fire protection analysis information for the LNP yard areas and outlying buildings 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 LNP COL FSAR, Revision 9, 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 LNP COL FSAR Section 9.5.2, the applicant provided the following:

AP1000 COL Information Items

- LNP COL 9.5-9, involving offsite interfaces

The applicant provided additional information in LNP COL 9.5-9 to resolve COL Information Item 9.5-9 (COL Action Item 9.5.2-3).

- LNP COL 9.5-10, involving emergency offsite communications

The applicant provided additional information in LNP 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 LNP COL 9.5-9, addressing interfaces to offsite locations, is based on:

- Appendix E, “Emergency Planning and Preparedness for Production and Utilization Facilities” to 10 CFR Part 50, Section IV.E(9)

The regulatory basis for LNP 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(j), “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 LNP 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 LNP COL application are documented in NUREG-1793 and its supplements.

In addition, in LNP COL FSAR Section 9.5.2, the applicant provided the following:

AP1000 COL Information Items

- LNP COL 9.5-9

The applicant provided additional information in LNP 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 LNP COL 9.5-9 involving offsite interfaces included under Section 9.5.2.5.1 of the LNP COL FSAR. To determine how the applicant addressed NRC Bulletin (BL) 80-15, "Possible Loss of Emergency Notification System (ENS) with Loss of Offsite Power," the staff requested additional clarification on the design of the site's primary and emergency power supplies by issuing RAI 2226 to the applicant. In its response dated March 11, 2011, the applicant committed to revising FSAR Section 9.5.2.2.5 in Revision 3 to provide the following information:

> The Emergency Notification System (ENS) onsite primary power supply is backed up by automatic transfer to a highly reliable secondary power supply, which complies with the requirements of NRC Bulletin 80-15 regarding loss of offsite power to the ENS. The ENS is accomplished by the communications system (EFS). The subsystems of the EFS that accomplish the ENS function are the wireless telephone system, telephone/page system and the private automatic branch system (PABX) system. These communication subsystems are independent of one another; therefore, a failure in one subsystem does not degrade performance of the other subsystems. Per DCD Subsections 9.5.2.2.1, 9.5.2.2.2, and 9.5.2.2.3, the normal 120-V ac power supplies the wireless telephone switch, the telephone/page system, and the PABX system. Upon loss of the normal power, the telephone switch, the telephone/page system, and the PABX system are powered from the non-Class 1E dc and uninterruptible power supply system sized to supply power for 120 minutes.
>
> The non-Class 1E dc and UPS system (EDS) is described in DCD Subsection 8.3.2.1.2 and the on-site standby power system (ZOS) is described in DCD Subsection 8.3.1.1.2.1. The non-Class 1E main ac power system (ECS) is described in DCD Subsection 8.3.1 1.1.

Offsite interfaces and emergency offsite communications are specifically discussed in Section F of the LNP COL Emergency Plan (EP). The emergency offsite communications between the site and NRC are described as follows:

- Emergency Notification System (ENS): Provides initial notifications to the NRC, as well as ongoing information about plant systems, status and parameters. ENS lines are located in the Control Rooms, Technical Support Centers (TSCs), and Emergency Operating Facility (EOF).

- Health Physics Network (HPN): Provides communications regarding radiological and meteorological conditions, assessments, trends, and protective measures. HPN lines are located in the TSCs and EOF.

- Reactor Safety Counterpart Link (RSCL): Allows for internal NRC discussions regarding plant and equipment conditions. RSCL lines are located in the TSCs and EOF.

- Protective Measure Counterpart Link (PMCL): Allows for conduct of internal NRC discussions on radiological releases, meteorological conditions, and protective measures. PMCL lines are located in the TSCs and EOF.

- Emergency Response Data System (ERDS) Channel: Allows transmittal of reactor parametric data from LNP Nuclear Plant (LNP) to the NRC. ERDS data is transmitted to the NRC Operations Center in Rockville, Maryland. ERDS provides a real-time transfer of plant data from LNP information systems. Duke Energy will activate the ERDS within one hour of the declaration of an Alert or higher emergency classification in accordance with LNP implementing procedures.

- Management Counterpart Link (MCL): This system has been established for internal discussions between the NRC Executive Team Director/members and the NRC Site Team Director or Duke Energy management. MCL lines are located in the TSCs and EOF.

- NRC Remote Access: Provides access to the NRC local area network (LAN). Modem access is provided in the TSCs and EOF for NRC access.

Additional onsite/offsite communications methods are described as follows:

- Commercial Telephones: Commercial telephones are located throughout LNP. These phones operate through the Florida Telephone switchboard located in Leesburg, Florida.

- DEF Voicenet System: The Voicenet System interconnects all Duke Energy Florida (DEF) plants, major substations, and main offices. Voicenet serves as the primary connection for ENS and is interconnected with the area public telephone system. This communication service is available throughout the DEF service area. The DEF Voicenet system routes calls independently of the local telephone lines that are used for the ENS function but will use these lines if available to route a call. This also allows the ENS function to be routed geographically independent of the local phone connections, thereby achieving the reliably required in Regulatory Issue Summary (RIS) 2000-11, “NRC Emergency Telecommunications System.” Backup for Voicenet is commercial telephone lines. The Voicenet system is wholly owned and operated by Duke Energy.

- The Florida Emergency Satellite Communications System (ESATCOM): This is an intrastate communications system that is operated by the State of Florida Division of Emergency Management in Tallahassee, Florida. The system connects the State Warning Point-Tallahassee (SWPT), state agencies, all Florida counties; weather service forecast offices, nuclear facilities, and other select locations via a satellite communications link. Voice transmissions from any of the locations are received at all other locations. The satellite dish is located at LNP with connections to the Control

Rooms, TSCs, and EOF. The LNP Control Room ESATCOM will provide back-up communications for notification of an emergency at LNP.

- Private telephone capability to the county and state warning points/Emergency Operation Centers (EOCs).

- Dedicated radio networks to the state and county warning points/EOCs.

- State of Florida Hot Ringdown Telephone System (HRTS): This system serves as the primary means of 24-hour per day communications between the following areas:
    - LNP Control Rooms
    - TSCs
    - SWPT
    - Department of Health, Bureau of Radiation Control (DHBRC)
    - Citrus County EOC
    - Levy County EOC
    - Marion County EOC

The HRTS consists of three separate networks utilizing dedicated telephone circuits to communicate with the SWPT. LNP will be able to dial all stations on the circuit or call a selected station(s). Each network includes LNP; the SWPT; Citrus, Levy and Marion County EOCs; the EOF; and the DHBRC. All stations on the network can call all or a selected number of other stations by utilizing a dial-up code. There are three separate conference-line phone systems established:

- Between the EOF and TSCs for emergency status information.

- Between the Control Rooms, TSCs and EOF for dose assessment information.

- Between the TSCs and Control Rooms for accident assessment information.

Appendix E to 10 CFR Part 50, Section IV.E(9) requires at least one onsite and one offsite communications system; each system shall have a backup power source. In addition, NRC BL 80-15 states that the applicant should provide backup power sources for the ENS in case of loss of offsite power. The emergency communications design for the LNP COL application provides multiple methods for both onsite and offsite communications including landlines dedicated for communications to the NRC, commercial lines and multiple forms of wireless communications such as satellite phones and radio networks. For the LNP COL application, the ENS is powered by the 120V-ac power system. Should a loss of the ac power system occur, the ENS is automatically switched over to the diesel backed, non-Class 1E direct current (dc) and uninterruptible power supply systems.

The staff finds the design of the emergency communications system provides sufficient means for onsite and offsite communications, with adequate backup communications methods. In addition, the staff finds that the design also provides adequate primary and backup power sources, to meet the requirements of Appendix E to 10 CFR Part 50, Section IV.E(9). The use of an uninterruptible power supply and diesel generator to provide backup power to the ENS in case of loss of offsite power adequately addresses NRC BL 80-15. The applicant committed to revising Section 9.5.2.2.5 in Revision 3 of the LNP COL FSAR to add the content quoted above and deleting the content that exists in Revision 2. These actions will be tracked as **Confirmatory Items 9.5-1** and **9.5-2** until such time as the applicant provides the staff Revision 3 of the FSAR with the changes verified.

Resolution of Confirmatory Items 9.5-1 and 9.5-2

Confirmatory Items 9.5-1 and 9.5-2 are applicant commitments to update section 9.5.2.2.5 of the LNP FSAR. The staff verified that LNP COL FSAR Section 9.5.2.2.5 was appropriately updated (or revised). As a result, Confirmatory Items 9.5-1 and 9.5-2 are now closed.

- LNP COL 9.5-10

The applicant provided additional information in LNP COL 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 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 LNP COL 9.5-10 concerning the emergency offsite communication system, including the crisis management radio system included under Section 9.5.2.5.2 of the LNP COL FSAR. The offsite communications interfaces with the site were described in Section 9.5.2.4 of this evaluation. This includes the following methods:

- Local Commercial Telephone System
- DEF Voicenet System
- Florida ESATCOM
- HTRS

The applicant also provides the following alternative communication methods to the dedicated phone lines that comprise the primary onsite and offsite communication methods:

- **Florida Department of Law Enforcement (FDLE) Radio System**: This is the Emergency Plan crisis management radio system. The LNP portion of this radio system is powered by the normal 120-Vac power supply with the non-Class 1E and uninterruptible power supply system providing power on loss of the normal power supply.

- **<u>Portable UHF Radios</u>**: These radios are available to emergency teams for limited communication on the LNP site. During normal day shift operations, key plant staff personnel have ultra high frequency (UHF) radios available for communication with the Control Rooms. These radios are the primary communications link during a fire. The system utilizes UHF repeaters and antennas located in the plant to aid in radio communications. Earphones are provided in high noise areas.

- **<u>Dedicated Radio Networks</u>**: These networks provide communications between state and county warning points and the EOCs.

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 of this evaluation are adequate in providing emergency communications equipment and facilities and thus meet the requirements of 10 CFR 50.47(b)(8). In addition, the staff finds the FDLE radio system adequately serves as the crisis management radio system. The FDLE radio system is a trunked design. The trunked system design for radio communications is commonly used by Federal and state authorities such as fire departments, police dispatch, etc. The trunked system design allows for multiple users (talk-groups), to use a small set of actual radio frequencies without hearing each other's conversations. With a trunked system, there is no 'dedicated' channel as in a conventional radio system so if a particular frequency channel is interrupted, a controlling computer will automatically rotate the affected talk-group to the next available frequency. The design allows two-way continuous communication between plant personnel and offsite authorities at county warning points and other state authorities. Therefore, the staff concludes that COL Action Item 9.5.2-1 has been addressed.

- STD COL 9.5-11

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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP 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:

- LNP 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.

- LNP COL 9.5-10 has been adequately addressed by the applicant in that the LNP 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 LNP COL FSAR, Revision 9, 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 LNP 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 LNP COL FSAR, Revision 9, 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 LNP COL FSAR Section 9.5.4.5.2, the applicant provided the following:

AP1000 COL Information Item

- 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 LNP 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 LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 thought this was a typographical error on the applicant's part because Table 1 of ASTM D975, "Standard Specification for Diesel Fuel Oils," 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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP COL FSAR 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 LNP COL FSAR, Revision 9, 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 LNP 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 LNP COL FSAR, Revision 9, 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 LNP 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 LNP COL FSAR, Revision 9, 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 LNP 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 LNP COL FSAR, Revision 9, 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 LNP 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 Levy Nuclear Plant (LNP) combined license (COL) Final Safety Analysis Report (FSAR), Revision 9, incorporates by reference Section 10.1 of the AP1000 Design Control Document (DCD), Revision 19.

In addition, in LNP 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 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 10.1 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 LNP COL FSAR.

10.1.4 Technical Evaluation

The NRC staff reviewed Section 10.1 of the LNP 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 LNP 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) COL application were equally applicable to the LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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.*

<u>Resolution of Standard Content Open Item 10.1-1</u>

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.*

<u>*Resolution of Standard Content Confirmatory Item 10.1-1*</u>

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 proposed by the applicant 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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The staff concludes that the information presented in the LNP 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 Turbine-Generator

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 LNP COL FSAR, Revision 9, incorporates by reference Section 10.2 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Section 10.2, the applicant provided the following:

Supplemental Information

- STD Supplement (SUP) 10.2-1

The applicant provided supplemental information in LNP 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 LNP 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 LNP 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 LNP 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 LNP 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 LNP 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 LNP 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 LNP 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 COL application were equally applicable to the LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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:

Supplemental Information

- *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 proposed by the applicant 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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP 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 LNP 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 non-safety-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 LNP COL FSAR, Revision 9, incorporates by reference Section 10.3 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Section 10.3, the applicant provided the following:

Supplemental Information

- STD SUP 10.3-1

The applicant provided supplemental information in LNP 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 LNP 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 LNP 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:

- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," 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 of 10 CFR Part 50 and the guidance in RG 1.37, as they relate to quality assurance requirements for the design, fabrication, and construction of safety-related SSCs.

10.3.4 Technical Evaluation

The NRC staff reviewed Section 10.3 of the LNP 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 LNP 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 COL application were equally applicable to the LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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:

> *<u>Supplemental Information</u>*
>
> - *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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP 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 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 <u>Other Features of Steam and Power Conversion System</u>

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 LNP COL FSAR, Revision 9, 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 LNP 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 LNP COL FSAR, Revision 9, 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 LNP COL FSAR Section 10.4.2.2, the applicant provided the following:

<u>*Site-Specific Information Replacing Conceptual Design Information*</u>

- LNP CDI

The applicant provided additional information to replace conceptual design information (CDI) in LNP 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.

- LNP CDI

The applicant provided additional information to replace CDI in LNP 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 LNP 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 LNP COL application are documented in NUREG-1793 and its supplements.

The staff’s review of this application is limited to the following LNP plant-specific design information that replaces the CDI identified in the AP1000 DCD.

<u>Site Specific Information Replacing Conceptual Design Information</u>

- LNP CDI

The LNP plant-specific design information was annotated as “LNP CDI” in LNP COL FSAR Section 10.4.2. 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 LNP CDI in LNP 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 LNP 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 LNP plant-specific design information will have no adverse affects on the capability of the main condenser evacuation system, CWS, or RWS and associated equipment. Also, the staff concludes that adding this LNP 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 LNP 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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP 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:

- LNP CDI, relating to LNP 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.

- LNP CDI, relating to LNP 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 radioactive 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 LNP COL FSAR, Revision 9, 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 LNP 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 LNP COL FSAR, Revision 9, 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 LNP 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 LNP COL FSAR, Revision 9, 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 LNP COL FSAR Section 10.4.5, the applicant provided the following:

AP1000 COL Information Item

- LNP COL 10.4-1

The applicant provided additional information related to the CWS design parameters in LNP 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).

- LNP COL 10.4-3

The applicant provided additional information regarding the chemistry requirements for the source of potable water to resolve the COL information item in Section 10.4.12.3, "Potable Water," of the AP1000 DCD (COL Action Item 10.5-5).

Site-Specific Information Replacing Conceptual Design Information

- LNP CDI

The applicant provided additional information to replace CDI in LNP 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 LNP COL 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 LNP 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 LNP COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the LNP COL FSAR:

AP1000 COL Information Items

- LNP COL 10.4-1

In LNP COL FSAR Section 10.4.5, the applicant provided additional information in LNP 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, "Habitability Systems," of this report. 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 LNP 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 text 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 and addressed its review of the system as follows.

In LNP COL FSAR Section 10.4.5.2.1, the applicant described the LNP site-specific CWS, as specified in LNP COL 10.4-1. The CWS and the cooling towers provide a heat sink for waste heat exhausted from the main steam turbine. The CWS design parameters are provided in LNP COL FSAR Table 10.4-201, "Supplemental Design Parameters for Major Circulating Water System Components." Further, in LNP COL FSAR Section 10.4.5.2.2, the applicant stated that the piping design pressure from the circulating water pump discharge isolation valves, including the condenser and waterboxes, to the discharge to the cooling tower is 75 pounds per square inch gauge. The staff reviewed these site-specific design parameters and compared them to the corresponding data in AP1000 DCD Tier 2 Table 10.4.5-1 and finds them acceptable as the LNP parameters are consistent with those for the certified design.

Also in LNP COL FSAR Section 10.4.5.2.2, the applicant provided information on the chemical treatment program for the CWS. The applicant stated that specific chemicals used within the system are determined by the site water conditions. Additionally, in LNP COL FSAR Section 10.4.5.5, the applicant identified that circulating water chemistry is controlled by cooling tower blowdown via regulating the blowdown valve. The staff finds that the applicant addressed the site-specific chemicals and control and maintenance of CWS chemistry as specified in COL Information Item 10.4-1.

The staff reviewed the information provided in the above LNP COL FSAR sections and finds that the applicant addressed the final configuration of the CWS as specified in COL Information Item 10.4-1. The staff finds that the CWS design parameters of temperature and flow rates in LNP COL FSAR Table 10.4-201 are consistent with the design parameters in AP1000 DCD Table 10.4.5-1. The staff also finds that the piping design pressures of the LNP CWS are consistent with the design pressures of the conceptual (non-site-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.

- LNP COL 10.4-3

The applicant provided additional information to resolve the COL information item in Section 10.4.12.3, "Potable Water," of the AP1000 DCD, which states:

> The Combined License applicant will address the chemistry requirements for the source of potable water. A biocide such as sodium hypochlorite is recommended. For addition, if a municipal site-specific source is not utilized, toxic gases such as chlorine are not recommended. The impact of toxic gases on the main control room compatibility is addressed in Section 6.4.

This COL information item was also captured as COL Action Item 10.5-5 in Appendix F of NUREG-1793:

> The COL applicant is responsible for the site-specific biocide for use in the potable water system.

In LNP COL FSAR Section 10.4.5.2.2, the applicant stated that sodium hypochlorite (NAOCI) would be used as a biocide for the potable water system. The staff's review of NAOCI in terms of toxic gases and control room habitability is addressed in Section 6.4 of this SER. The staff finds that the NAOCI being used by the applicant as a biocide for the potable water system is acceptable because of its use by operating plants as a biocide without any impact to reactor safety.

The staff reviewed the information provided in LNP COL FSAR Section 10.4.5.2.2 and finds that the applicant addressed the requirements specified in LNP COL Information Item 10.4-3.

Site-Specific Information Replacing Conceptual Design Information

- LNP CDI

The applicant provided site-specific design information as part of the FSAR regarding the CWS. The applicant replaced bracketed text throughout Section 10.4.5 of the AP1000 DCD to provide LNP-specific CWS power design generation basis component information, general CWS description, component description, system operation, tests and inspections, and instrumentation applications. The staff reviewed the text added as LNP CDI throughout LNP COL FSAR Section 10.4.5 related to the plant-specific CWS system, and the following provides the staff's evaluation of these CDIs in the application.

In LNP COL FSAR Sections 10.4.5.1, "Design Bases," and 10.4.5.2, "System Description," the applicant provided a description of its plant-specific CWS system configuration. The CWS is a non-safety-related system. The CWS supplies cooling water to remove heat from the main condensers, the turbine building closed cooling water system (TCS) heat exchangers, and the condenser vacuum pump seal water heat exchangers under varying conditions of power plant loading and design weather conditions.

In LNP COL FSAR Section 10.4.5.2.2, “Component Description,” the applicant provided site-specific design information regarding the CWS major components, such as circulating water pumps, cooling tower, cooling tower makeup and blowdown, and associated piping and valves, which address the final configuration of the LNP CWS as specified in LNP COL 10.4-1.

The LNP CWS consists of three 33-1/3 percent capacity circulating water pumps. Each pump discharge line has a motor-operated butterfly valve located between the pump discharge and the main header, which permits isolation of one pump for maintenance and allows two-pump operation.

The LNP cooling tower is a mechanical induced-draft tower. It is designed to cool the water to 89.1°Fahrenheit (F)(31.7° Celsius (c)) with a hot water inlet temperature of 117.8°F (47.6° C) (see LNP COL FSAR Table 10.4-201). The staff finds that the above temperature values are acceptable as they demonstrate an equally effective cooling tower design as listed in AP1000 DCD, Table 10.4.5-1.

In LNP COL FSAR Section 10.4.5.2.3, “System Operation,” the applicant states that the mechanical draft cooling tower is positioned so that its collapse would have no potential to damage SSCs needed for safe shutdown of the plant. However, the staff could not find further details regarding the effects of the cooling tower failure on the nearby safety-related equipment and/or structure of the plant. As described in NUREG-0800, Section 10.4.5, “Circulating Water System,” Acceptance Criteria, 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. The staff did not find sufficient details to provide assurance, in accordance with the requirements of GDC 4, that flooding resulting from cooling tower failure would have no effect on the nearby safety-related SSCs. Therefore, in RAI 10.4.5-1, the staff requested that the applicant provide additional information to ensure that failure of the tower will not affect the safety-related systems or equipment that are located in the proximity of the cooling tower.

In a letter dated July 6, 2009, the applicant provided its response to RAI 10.4.5-1, where the applicant stated that the mechanical draft cooling towers at LNP Units 1 and 2 are less than 300 feet tall to comply with the LNP County Zoning Ordinance. The closest seismic Category I (safety-related) structure to the cooling tower, on each unit, is the auxiliary building, which is approximately 700 feet (213 Meters (m)) away from the cooling tower. Also, the closest seismic Category II structure, in each unit, is the annex building, which is also more than 700 feet (213 meters) away from the cooling tower. The applicant clarified that no safety-related structures or components are located in seismic Category II structures as stated in AP1000 DCD, Sections 3.2.1.1.1 and 3.2.1.1.2. The applicant further stated that in the unlikely event the cooling tower collapsed in the direction of these structures; the cooling tower collapse would not affect either structure.

Additionally, the applicant proposed a revision to LNP COL FSAR Chapter 10, Section 10.4.5.2.2. In the proposed revision, the applicant acknowledged that a collapse of the mechanical draft cooling towers has the potential to rupture the circulating water, blowdown and raw water piping associated with the tower. Referring to Section 3.4.1.1 of the AP1000 DCD, the applicant further stated that failure of the cooling tower or the CWS piping in the yard could result in a potential flood source. However, based on the location of the cooling tower and site grading, the water from the tower or the CWS would be carried by site grading and drainage system away from the safety-related structure. According to the applicant, the consequences of the failure of circulating water piping in the yard and associated with the cooling tower are bounded by the analysis of the CWS piping in the turbine building as described in AP1000 DCD, Sections 3.4.1.1.1, 3.4.1.2.2.3, and 10.4.5.2.3. The applicant concluded, as a result, that the CWS piping rupture bounds the rupture of the RWS piping and the blowdown water line. However, the staff could not find a justification for this “bounding” claimed in the applicant’s response. Therefore, in supplemental RAI 10.4.5-1, the staff requested the applicant provide additional information concerning the justification of the statement regarding the bounding analysis.

In its response to supplemental RAI 10.4.5-1, the applicant provided further details for the AP1000 DCD CWS flow rates and the LNP site-specific CWS flow rates, blowdown flow rates, and RWS flow rates. The applicant identified that the CWS flow used in the AP1000 DCD flood evaluations is 631,100 gallons per minute (gpm), (2,388,973 liters/min) and the LNP CWS flow is 559,365 gpm (2,112,426 L/min), as identified in LNP COL FSAR Table 10.4-201. Part of the LNP CWS flow includes the blowdown flow of 28,260 gpm (106,975 L/min), which is supplied from the CWS pump discharge header. Since the RWS piping rupture flow rate, which provides cooling tower makeup and blowdown water, is 43,290 gpm, the CWS piping rupture due to cooling tower failure bounds the RWS piping rupture.

On the basis of its review, the staff finds that GDC 4 is met because the LNP CWS design includes provisions to accommodate the effects of discharging water from failure of a component or piping in the CWS. For example, in the event of a cooling tower rupture, LNP is designed such that the nearest safety-related structure is 700 (213 meters) feet away from the cooling tower, and water from such flooding would be carried away from safety-related structures by the site grading and drainage system. Additionally, LNP has demonstrated that the CWS piping rupture bounds the rupture of the RWS piping and the blowdown water pipe. Therefore, the staff finds the applicant’s response to RAI 10.4.5-1 and its supplement is acceptable and this RAI is closed. The staff verified that the proposed changes identified in the RAI responses have been incorporated into the FSAR.

CWS cooling tower makeup is provided by the RWS, described in LNP COL FSAR Section 9.2.11, “Raw Water System.” Makeup to and blowdown from the CWS is controlled by its makeup and blowdown control valves and is evaluated in SER Section 9.2.11.

The underground portions of the CWS piping are constructed of prestressed concrete pressure piping. The remainder of the piping is carbon steel, with an internal coating of a corrosion resistant compound. Control valves provide regulation of cooling tower makeup and blowdown. The CWS is designed to withstand the maximum operating discharge pressure of the circulating water pumps. Piping includes the expansion joints. As described earlier in LNP COL 10.4-1, the piping design pressures are in accordance with the AP1000 DCD values, and are therefore, acceptable.

The staff finds that the effects of flooding due to a CWS failure, such as a rupture of an expansion joint, will not result in detrimental effects on safety-related equipment, because the turbine building does not house safety-related equipment 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. Small CWS leaks in the turbine building will drain into the waste water system. Large CWS leaks due to pipe failures will be indicated in the control room by a loss of vacuum in the condenser shell. The staff finds that these provisions of the LNP CWS design meet the requirements of GDC 4, and the acceptance criteria described in NUREG-0800, Section 10.4.5, Section II, as it relates to design provisions to accommodate the effects of discharge water that may result from a failure of a component or piping in the CWS.

In LNP 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 provisions are made during cold weather to direct a portion of the circulating water flow into freeze prevention spray headers on the periphery of the cooling tower, which heats air flowing through the peripheral spray and allows de-icing in the cooling tower fill. The staff finds that these provisions of the LNP CWS design meet the requirements of GDC 4 and the acceptance criteria in NUREG-0800, Section 10.4.5, Section II..

In LNP 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 LNP 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.

The staff finds that CDI information provided in the above LNP COL FSAR sections adequately address the final configuration of the LNP CWS system as specified in the AP1000 DCD.

Based on its review of the information provided by the applicant, the staff concludes that the site-specific design of the LNP CWS meets the requirements of GDC 4, with respect to the effects of discharging water that may result from a failure of component or piping in the CWS.

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 LNP COL FSAR related to this section. The results of the NRC staff’s technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP COL FSAR is acceptable and meets 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:

- LNP 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 chemicals and control and maintenance of the CWS chemistry in order to be consistent with AP1000 DCD.

- LNP COL 10.4-3, relating to sodium hypochlorite (NAOCI) being used as a biocide for the potable water system, is acceptable to the staff because NAOCI is used by operating plants as a biocide without any impact to reactor safety.

- LNP 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 LNP COL FSAR, Revision 9, 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 LNP 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 LNP COL FSAR, Revision 9, 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 LNP COL FSAR Section 10.4.7.2.1, the applicant provided the following:

AP1000 COL Information Item

- LNP COL 10.4-2

The applicant provided additional information in LNP 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 LNP 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 LNP 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 LNP 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 COL application were equally applicable to the LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 LNP COL FSAR:

AP1000 COL Information Item

- LNP COL 10.4-2

In LNP COL FSAR Section 10.4.7.2.1, the applicant provided additional information in LNP 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 oxygen scavenging agent and pH adjuster selection for 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 LNP COL FSAR modified Section 10.4.7.2.1 of the AP1000 DCD, to state:

> The oxygen scavenger agent is hydrazine and the pH control agent is morpholine. During shutdown conditions, carbohydrazide may be used in place of hydrazine.

The NRC staff reviewed the resolution to LNP 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 is the most commonly used oxygen scavenger for PWR secondary systems and is generally recognized as effective for this purpose. Therefore, the staff finds the identified oxygen scavenger agent is 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 morpholine. 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 morpholine is acceptable.

The following portion of this technical evaluation section is reproduced from Section 10.4.7.4 of the VEGP SER:

> <u>*Supplemental Information*</u>
>
> - *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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP 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:

- LNP 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 LNP COL FSAR, Revision 9, 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 LNP COL application are documented in NUREG-1793 and its supplements.

10.4.9 Startup Feedwater System (Related to RG 1.206, Section C.III.1, Chapter 10, C.I.10.4.9, "Auxiliary Feedwater System (PWR)")

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 LNP COL FSAR, Revision 9, 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 LNP 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 LNP COL FSAR, Revision 9, 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 LNP 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 LNP COL FSAR, Revision 9, 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 LNP COL application are documented in NUREG-1793 and its supplements.

10.4.12 Combined License Information

Section 10.4.12 of the LNP COL FSAR, Revision 9, 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 LNP 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 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.

11.0 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 isotope in the reactor coolant, fraction of fission product activity released to the reactor coolant, and concentrations of all nonfission product radioactive isotopes in the reactor coolant. Gaseous and liquid waste sources are considered in the evaluation of effluent releases.

Section 11.1 of the Levy Nuclear Plant (LNP) combined license (COL) Final Safety Analysis Report (FSAR), Revision 9, incorporates by reference, Section 11.1, “Source Terms,” of Revision 19 of the AP1000 Design Control Document (DCD). In addition, in the LNP COL FSAR, the applicant provided the following:

Departures

- LNP DEP 6.4-1

The applicant provided additional information in Section 11.1 of the LNP COL FSAR about LNP 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 LNP DEP 6.4-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.2 of this SER.

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed Section 11.1 of the LNP 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 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 LNP 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).

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 LNP COL FSAR, Revision 9, incorporates by reference Section 11.2 of the AP1000 DCD, Revision 19.

In addition, in LNP 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 LNP 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 to resolve COL Information Item 11.2-2 (COL Action Item 11.2-2). The additional information addresses the methodology for calculating doses and the cost-benefit analysis of population doses in LNP COL FSAR Section 11.2.3.5.

- LNP COL 11.2-1 and LNP COL 13.5-1

The applicant provided additional information in LNP COL 11.2-1 and LNP 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 and RAI 11.02-5.

- LNP COL 11.2-2

The applicant provided additional information in LNP COL 11.2-2 to resolve COL Information Item 11.2-2 (COL Action Item 11.2-2). The additional information addresses the methodology for calculating doses and the cost-benefit analysis of population doses in LNP COL FSAR Section 11.2.3.5.

- LNP COL 2.4-5 and LNP COL 15.7-1

LNP COL FSAR Section 11.2 does not identify LNP COL 2.4-5 and LNP COL 15.7-1 as COL information items applicable to Section 11.2. However, LNP COL 2.4-5 and LNP COL 15.7-1 provide information regarding a postulated liquid waste tank failure, which is evaluated by the NRC staff as part of liquid waste management. Therefore, LNP COL 2.4-5 and LNP COL 15.7-1 are evaluated in Section 11.2.4 of this safety evaluation report (SER). In LNP COL FSAR Section 2.4, the applicant performed the consequence analysis of a postulated liquid waste tank failure in FSAR Section 2.4.13 to address COL Information Items 2.4-5 and 15.7-1.

- LNP COL 11.5-3

The applicant provided additional information in LNP 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 LNP COL FSAR Section 11.2.3.5.

Supplemental Information

- STD SUP 11.2-1

The applicant added in LNP COL FSAR Section 11.2.3.6 supplemental (SUP) information to address the quality assurance (QA) program to be applied to the LWMS.

- LNP SUP 11.2-1

The applicant added in LNP COL FSAR Section 11.2.1.2.4 supplemental information to describe the exterior radwaste discharge piping. In a letter dated May 4, 2011, the applicant proposed to add to a future version of the FSAR supplemental information in LNP SUP 11.2-1 that describes site-specific design feature of the discharge piping. In a letter dated December 7, 2011, the applicant provided a voluntary supplemental response with additional detail to be incorporated in a future revision of the FSAR.

License Condition

- Part 10, License Condition, Radwaste Building Radioactivity Limits

LNP 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 A_2 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.

The regulatory requirements applicable to the LWMS are as follows:

- 10 CFR 20.1301(e)
- 10 CFR 20.1302, "Compliance with dose limits for individual members of the public"
- 10 CFR 20.1406, "Minimization of contamination"
- 10 CFR 50.34a, "Design objectives for equipment to control release of radioactive material in effluents – nuclear power reactors"
- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criteria (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
- 10 CFR 52.80(a)
- Title 40 of the *Code of Federal Regulations* (40 CFR) Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations"

Guidance for accepting the additional 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 form 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.

11.2.4 Technical Evaluation

The NRC staff reviewed Section 11.2 of the LNP 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 LWMS. The results of the NRC staff’s evaluation of the information incorporated by reference in the LNP 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
- LNP COL 11.2-1, Radwaste Building Source Term Inventories
- LNP COL 13.5-1, Radioactive Waste Management Procedures
- LNP COL 11.2-2, Cost-Benefit Analysis of Population Doses
- LNP COL 2.4-5, Accidental Release of Liquid Effluents into Groundwater and Surface Water
- LNP COL 15.7-1, Consequences of Tank Failure
- LNP COL 11.5-3, Individual Dose Limits in 10 CFR Part 50, Appendix I
- STD SUP 11.2-1, Quality Assurance
- LNP SUP 11.2-1, 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 LNP 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. RGs 1.109, 1.112, and 1.113 provide acceptable methods for performing this analysis.

    B. In addition to A, 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 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 that are located outside of containment and impacts of the release of radioactive materials at the nearest potable water supply in an unrestricted area for direct human consumption or indirect consumption through animals, crops, and food processing.

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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 11.2.4 of the VEGP SER:

> *AP1000 COL Information Items*
>
> *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.

- LNP COL 11.2-1 and LNP COL 13.5-1

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 provide 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.

- LNP COL 11.2-2

The applicant provided additional information in LNP 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.

In LNP COL FSAR Section 11.2.3.5.3, the applicant provided a complete cost-benefit analysis for the site according to the guidance in RG 1.110 using the population doses stated in FSAR Section 11.2.3.5.2.

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 processing system at a cost of $11,140 per year. Assuming that this filter will eliminate all radioactive material from the liquid effluent, thereby eliminating all environmental dose consequence, the resulting cost per dose reduction was $9,858 per total body person-rem ($11,140/1.13 person-rem) and $9,207 per thyroid person-rem. These cost-benefit estimates are above the criterion of $1,000 per person-rem reduction, as specified in 10 CFR Part 50, Appendix I, Section II.D, for the inclusion of additional radwaste processing capabilities. Thus, the applicant concluded that the LWMS meets the as low as reasonably achievable (ALARA) requirements and requires no augments.

The NRC staff performed an independent assessment of the population doses, considering the reasonableness of the modeling assumptions as provided by the applicant in LNP COL FSAR Tables 11.2-201 and 11.2-202 and the guidance in RG 1.110. The staff's assessment, with independent calculations, confirmed the applicant's analytical results that the LWMS meets the cost-benefit design criterion of 10 CFR Part 50, Appendix I, Section II.D. Thus, the staff finds the applicant's assessment of the population doses acceptable.

- LNP COL 2.4-5 and LNP COL 15.7-1

The applicant provided additional information in LNP COL 2.4-5 and 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.

LNP COL FSAR Section 2.4.13 addresses accidental release of liquid effluents into ground and surface water. The applicant postulated a release of the contents of the waste liquid system effluent hold-up tank, consistent with the guidance provided in BTP 11-6. 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 indirect consumption through animals (livestock watering), crops (agricultural irrigation), and food processing (water as an ingredient).

All the liquid radwaste system waste tanks were considered in the applicant's evaluation based on their location in a nonseismic building. The applicant determined that the effluent holdup tanks have both the highest potential radioactive isotope inventory and the largest volume, so these tanks were used to perform the analysis. The applicant considered these tanks a conservative selection for the purpose of calculating the effects of the failure of a radioactive liquid-containing tank. There are two 28,000-gallon holdup tanks per unit. For the evaluation, one tank was postulated to fail. The failed tank was assumed to be 80 percent full and contain radionuclide concentrations corresponding to 101 percent of the reactor coolant source term. The concentrations of radionuclides are taken from AP1000 DCD, Table 11.1-2, "Design Basis Reactor Coolant Activity."

The entire contents of the tank were assumed to be released to the Floridan aquifer, the principal source of potable water near the LNP site. This was deemed a conservative assumption based on the hydraulic conductivity of the Floridan aquifer which is about twice as high as that of the surficial aquifer and a presumption that most of the release would be to the surficial aquifer rather that the Floridan aquifer. The release migrates southwest in the direction of decreasing hydraulic head. There are public supply wells in the direction of groundwater flow, at least 5 miles from the LNP site. The nearest resident in the direction of groundwater flow is 2.7 km west-southwest of the LNP site. Groundwater is extracted from the Floridan aquifer for potable use at the LNP site.

The applicant analyzed two cases. The first was a hypothetical nearest well supplied by the Floridan aquifer at 2 km southwest of LNP. This location is in the direction of groundwater flow and is on the LNP site boundary. The second case examined the Lower Withlacoochee River. The applicant identified no users of this surface water, but assumed the pathway is groundwater that moves downgradient from the LNP site and resurfaces within the Lower Withlacoochee River, at a distance of approximately 7 km.

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, effluent concentration limits (ECLs) at the nearest source of potable water. These 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 LNP COL FSAR Section 2.4.13.2.3 and in LNP COL FSAR Tables 2.4.13-204 and 2.4.13-205. Compliance is demonstrated by evaluating the ratios of the calculated aquifer radionuclide concentration to its ECL value for all released radionuclides. Using standard, acceptable groundwater modeling techniques, the applicant demonstrated compliance by showing that the sum of the ECL ratios for both locations was less than unity. The result of this calculation was that the sum of the ratios was less than 10^{-10}, or essentially zero, at the Lower Withlacoochee River location, and 0.007 at the well location, or 0.7 percent of the 50 mrem criterion for inclusion of the pathway in considering the MEI.

The staff's analysis considered whether other surface water pathways, such as ingestion of fish living in water containing radionuclides and ingestion of crops irrigated with water containing radionuclides could significantly increase exposures. The staff performed an independent evaluation of the fish ingestion pathway at the Lower Withlacoochee River location and of the vegetable ingestion pathway for crop irrigation at the well location. The evaluations showed that these additional pathways are not significant. The independent evaluations are presented below.

LNP SER Table 11.2-1 presents the results of a conservative dose assessment for fish consumption from the Lower Withlacoochee River. The radionuclide concentrations assumed for this location are as presented in LNP COL FSAR Table 2.4.13-204. In LNP SER Table 11.2-1, the fifth column is the calculated dose for an individual consuming 21 kilograms (kg) per year fish from the Lower Withlacoochee River assuming the radionuclide concentrations in the river remain at the assumed concentrations for the year. (Assumed fish consumption quantities represent the maximally exposed individual (MEI) values from RG 1.109.)

As SER Table 11.2-1 shows, the conservatively calculated MEI dose for one year of exposure from the fish exposure pathway is less than 10^{-9} mrem, significantly less than the corresponding 50 mrem dose criterion.

The staff also performed a conservative dose assessment for ingestion of vegetables irrigated with groundwater from the hypothetical nearest well in the Floridan aquifer in the direction of groundwater flow. The radionuclide concentrations in water from the hypothetical well are calculated to be higher than those in the river, so the well water concentrations were used for estimating the dose from vegetables irrigated with water. The radionuclide concentrations for this location are those presented in LNP COL FSAR Table 2.4.13-205. Assuming this groundwater concentration for a year following a tank failure and the modeling of RG 1.109 for irrigated crops (60-day growing period and maximum individual vegetable consumption rate of 520 kilograms per year (kg/yr)), the resulting hypothetical dose to an individual would be 0.04 mrem. The staff determined that the calculated MEI dose of 0.04 mrem for one year of exposure from the ingestion of vegetables irrigated with water from the hypothetical nearest well in the Floridan aquifer is well below the threshold requiring inclusion in the comparison with the 50-mrem dose criterion.

In response to RAI 2.4.13-1, the applicant addressed the issue of dose from fish and vegetable ingestion associated with the tank failure accident. Based on a conservative analysis for fish living in water with radionuclide concentrations equal to that in the water of the hypothetical nearest well in the Floridan aquifer, the applicant concluded the dose would be 4.3E-3 mrem per year (mrem/yr). The applicant's estimate of dose from ingestion of vegetables irrigated with well water from the Floridan aquifer was 0.017 mrem/yr. This estimate is lower than the staff's evaluation because of different modeling and assumptions (e.g., the applicant assumed 14 kg/y consumption versus 520 kg/yr assumed by staff). Both the applicant's and the staff's assessments indicate no significant contribution to dose via the fish and irrigated crop pathways for the tank failure analysis.

Based on the above evaluations by the staff and the applicant's analysis in the FSAR and in its response to RAI 2.4.13-1, the staff finds potential doses to members of the public resulting from an accidental release of liquid effluents meet NUREG-0800, Section 2.4.13 Acceptance Criterion No. 5 and the referenced BTP 11-6.

- LNP COL 11.5-3

The applicant provided additional information in LNP COL 11.5-3 to resolve the COL Information Item 11.5-3, 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 LNP 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 LNP COL FSAR Sections 11.3.3.4 and 11.4.)

The NRC staff reviewed the applicant's response to LNP COL 11.5-3 related to compliance with 10 CFR Part 50, Appendix I, Sections II.A and II.D and issued RAI 11.2-1. RAI 11.2-1 requested that the applicant provide the details of the individual and population dose analyses.

In response to RAI 11.2-1, the applicant provided a description of the required model assumptions and input parameters needed to run LADTAP II computer codes to calculate radionuclide concentrations in Crystal Bay (Gulf of Mexico) that are released via the Crystal River Energy Complex (CREC) Discharge Canal.

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 MEI of the public and to the population within 80 km (50 mi) from the postulated liquid effluents discharged.

LNP COL FSAR Table 11.2-201, "Dilution Factors," and Table 11.2-202, "LADTAP II Input for Dose Rates," include liquid pathway parameters used as input to the dose calculation, including cooling tower blowdown flow rate, site-specific dilution factors, transit times to receptors, fish and invertebrate harvest rates, and recreational usage data for the Gulf of Mexico. Discharge is to the Gulf of Mexico, via the CREC. The applicant chose the simple dilution model to calculate dilution of the radioactive effluent. The only dilution assumed was that provided by the effluent mixing with the flow in the Discharge Canal. LNP 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 annual individual total body dose to the teenager of 0.000052 mSv (0.0052 mrem) and a maximum annual individual organ dose to the adult GI-LLI of 0.000714 mSv (0.0714 mrem) from all applicable exposure pathways. 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 0.03 mSv (3 mrem) to the total body and 0.1 mSv (10 mrem) to any organ.

The calculated annual population doses listed in LNP COL FSAR Table 11.2-204 are 0.0113 person-Sv (1.13 person-rem) to the total body, and 0.0121 person-Sv (1.21 person-rem) to the thyroid. The applicant used the population doses in the cost-benefit analysis previously described in this SER.

In response to RAI 11.2-1, the applicant explained the derivation of values used for population, water use, sport fish harvest, commercial fish harvest, and recreational time spent on the Gulf of

Mexico. The staff reviewed the derivation of these values and found them to be reasonable upper-bound estimates that are unlikely to be exceeded. Consequently, the staff used the applicant's values in their independent dose estimation.

The NRC staff performed an independent assessment using the LADTAP II computer code and compared the results to those of the applicant and the Appendix I criteria. The modeling assumptions used by the staff for the MEI and population dose calculations, as shown in SER Table 11.2-2, were consistent with the applicant's. Modeling parameter values, as shown in SER Table 11.2-3, were also consistent with the applicant's. The results of the staff's calculations were consistent with those of the applicant's.

SER Table 11.2-4 compares the resulting dose estimates between the applicant's analysis and the 10 CFR Part 50, Appendix I criteria. Table 11.2-4 shows that all doses are below the Appendix I criteria. The NRC staff determined that the applicant had 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 and RAI 11.2-1 closed.

Liquid Radwaste Discharge Path Recirculation

In the course of an environmental audit site visit, the staff found that periodically detectable levels of tritium from the Crystal River Nuclear Generating Plant Unit 3 (CR-3) discharge have been measured in the CR-3 intake canal water in samples collected as part of the routine radiological environmental monitoring program at CR-3. This indicated a potential recirculation pathway relevant to 10 CFR Part 50, Appendix I guidelines applicable to LNP liquid releases, since the LNP discharge would be via the CR-3 discharge structure/canal. In RAI 11.2-3, the applicant was requested to provide an evaluation of this potential recirculation pathway and to provide additional information, as applicable, on the impact this recirculation path could have on potential doses from liquid effluents. In response to RAI 11.2-3, the applicant stated that the existence of a recirculation path would not have an effect on the calculated doses from LNP liquid effluents or compliance with 10 CFR Part 50, Appendix I. The applicant provided an analysis that showed that any recirculation that would occur was nonuniform in both magnitude and time/duration. As indicated, the receiving water body (Gulf of Mexico) and the periodic recirculation is not defined as a confined system (impoundment) that would lead to buildup in radioactivity levels over time.

The staff performed a simplified, conservative assessment of the periodic recirculation that could occur, considering the CR-3 circulation flow and the LNP discharge flow. Since the flow in the CR-3 circulation loop is calculated as 20 times greater than the discharge flow from LNP, the concentration of radionuclides in the LNP discharge canal would be diluted by a corresponding factor. Under certain tide and wind conditions, some of this discharge, with the diluted concentration of radionuclides, could be drawn back into the CR-3 intake. An earthen dike separates the CR-3 intake canal from the discharge area to minimize any recirculation effect. Recirculation would only occur during flood tidal conditions where the tidal flow would reverse the discharge plume into the area of the intake for CR-3. Since the discharge canal is not a closed loop, recirculation would be for a limited duration, affected by the shifts in local flows caused by diurnal tides and winds. Neglecting the effect of the earthen dike, the near-field concentration entering the intake canal could be as high as that being discharged to the Gulf of Mexico (i.e., the LNP discharge concentration divided by 21). This condition would exist for only as long as the opposing tidal currents prevailed, which would be a maximum of 6 hours before reversal of flow. Depending on the residence time for water in the intake/discharge canal loop, the result of the continuous intermittent reconcentration is that the average concentration, for the purposes of 10 CFR Part 50, Appendix I compliance, would be not more than 1.5 to 1.6 times the LNP discharge concentration, divided by the dilution factor of 21.

The applicant's assessment takes no credit for the earthen dike constructed to minimize the recirculation effect and assumes 100 percent recirculation, which does not appear likely. As presented in the applicant's response, the calculated liquid pathway doses with no recirculation considered were less than 0.8 percent of the applicable dose criterion of 10 CFR Part 50, Appendix I. Thus, a potential 1.5 or 1.6 recirculation factor would not result in doses more than 2 percent of the dose criterion. The NRC staff verified the applicant's statement that the existence of a recirculation path would not have an effect relative to 10 CFR Part 50, Appendix I. Therefore, this issue is resolved and RAI 11.2-3 is closed.

The following portion of this technical evaluation section is reproduced from Section 11.2.4 of the VEGP SER:

Supplemental Information

> *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.*

- LNP SUP 11.2-1

The applicant provided supplemental information addressing the exterior radwaste discharge piping in LNP COL FSAR Section 11.2.1.2.4, "Controlled Release of Radioactivity." In letters dated May 4, 2011, and December 7, 2011, the applicant proposed adding to the FSAR descriptions of the site-specific design features of the discharge piping.

This matter is related to 10 CFR 20.1406 and is addressed in SER Section 12.3.

License Condition

In a letter dated August 23, 2013, 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 A_2 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 LNP COL 11.2-1 and LNP COL 13.5-1.

Demonstrating Compliance with 10 CFR 20.1301(e)

10 CFR 20.1301(e) requires that NRC-licensed facilities comply with the Environmental Protection Agency (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, and radioactive waste storage areas. The EPA standards apply to the entire site or facility, whether it has a single unit or multiple units.

The staff's review of the LNP COL FSAR (Revision 0) revealed that the applicant did not provide sufficient information demonstrating compliance with 10 CFR 20.1301(e). The staff issued RAI 11.2-2 requesting that the applicant provide this information.

The applicant demonstrated compliance with the EPA standard in the LNP COL FSAR by summing the annual individual liquid and gaseous effluent doses for the planned LNP Units 1 and 2, as well as the existing CR-3. The applicant lists the results in LNP COL FSAR Table 11.2-205. SER Table 11.2-5 lists these dose summations and compares them to the dose requirements in 40 CFR Part 190. The expected doses are below the EPA limits. The staff confirmed that the doses listed in Table 11.2-5 are correct and determined that the applicant's effluent releases would be within the 40 CFR Part 190 standard. RAI 11.2-2 is closed.

Demonstrating Compliance with 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 Table 2 of

Appendix B to 10 CFR Part 20. The applicant demonstrated compliance with this requirement by referencing the AP1000 DCD. Section 11.2.3.4 of the DCD 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 would 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 11.2.3.4 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

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 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 LNP COL FSAR Section 12.3. The staff's evaluation and conclusion pertaining to compliance with 10 CFR 20.1406 are included in SER Sections 12.3 and 12.5.

11.2.5 Post Combined License Activities

For the reasons discussed in the technical evaluation section above, the staff proposes to include the following license condition:

- 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 A_2 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 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 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 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 LWMS, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP 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, LNP COL 11.2-1, LNP COL 11.2-2, LNP COL 2.4-5, LNP COL 15.7-1, LNP COL 11.5-3, LNP COL 13.5-1, STD SUP 11.2-1, and LNP SUP 11.2-1) and the proposed license condition 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 has provided sufficient information and that the review and calculations support the conclusions 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. 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, 10 CFR 52.79(a)(3), GDC 60 and 61, and Appendix I to 10 CFR Part 50.

11.3 <u>Gaseous Waste Management System</u>

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 LNP COL FSAR, Revision 9, incorporates by reference Section 11.3 of the AP1000 DCD, Revision 19.

In addition, in LNP 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.

- LNP COL 11.3-1

The applicant provided additional information in LNP 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 LNP COL FSAR Section 11.3.3.4.

- LNP COL 11.5-3

The applicant provided additional information in LNP 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 LNP COL FSAR Section 11.3.3.4.

Supplemental Information

- STD SUP 11.3-1

The applicant added supplemental information in LNP 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 LNP 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 additional information on the GWMS is established in:

- 10 CFR 20.1301(e)
- 10 CFR 20.1302
- 10 CFR 20.1406
- 10 CFR 50.34a
- 10 CFR Part 50, Appendix A, GDC 60
- 10 CFR Part 50, Appendix A, GDC 61
- 10 CFR Part 50, Appendix I, Sections II.B, II.C and II.D
- 10 CFR 52.79(a)(3)
- 10 CFR 52.80(a)

Guidance for meeting these requirements is in:

- 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 NRC staff reviewed Section 11.3 of the LNP 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 GWMS. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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

- LNP COL 11.3-1, Cost-Benefit Analysis of Population Doses

- LNP 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 LNP COL FSAR Section 11.3 met the regulatory requirements in the regulations stated above (SER Section 11.3.3) 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. RGs 1.109 and 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. RGs 1.109 and 1.111 provide acceptable methods for performing this analysis.

D. In addition to 1.A, 1.B, and 1.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 Table 2, Column 1, of Appendix B to 10 CFR Part 20.

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. 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 application.

- 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 offgas 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 11.3.4 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 to COL Information Item 11.3-1 related to the cost-benefit analysis methodology described in VEGP COL 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.*

- LNP COL 11.3-1

The applicant provided additional information in LNP COL 11.3-1 to resolve COL Information Item 11.3-1. COL Information Item 11.3-1 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 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 in 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 NRC staff reviewed the resolution of COL Information Item 11.3-1 related to the cost-benefit analysis included under Sections 11.3.3.4.4 and 11.3.5.1 of the LNP COL FSAR.

The applicant performed a site-specific analysis to determine that the offsite dose due to gaseous effluents is bounded by the AP1000 site parameters included in AP1000 DCD Chapter 1 and Tables 11.3-1, 11.3-2, and 11.3-4. The applicant discussed the site-specific cost-benefit analysis in LNP 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.

As shown in LNP COL FSAR Section 11.3.3.4.4 and SER Table 11.3-1, the LNP population doses are 5.74 person-rem total body and 8.33 person-rem thyroid.

The results of the applicant's analysis showed that the lowest-cost option for gaseous radwaste treatment system augments is a steam generator flash tank vent to main condenser at a total annual cost (TAC) of $6,320. Assuming that this augment will eliminate all radioactive materials from the gaseous effluent, the resulting cost per dose reduction was $1,100 per total body person-rem ($6,320/5.74 person-rem) and $759 per thyroid person-rem ($6,320/8.33 person-rem). While the costs per person-rem reduction exceed the $1,000 per person-rem criterion in Appendix I to Part 50 for the total body dose, the costs per person-rem reduction are below the $1,000 per person-rem criterion for the thyroid dose and, therefore, warranted further evaluation.

The applicant evaluated four potential augments for the thyroid dose as described below. Since the estimated thyroid dose of 8.33 person-rem exceeds the 6.32 person-rem threshold value ($6,320 augment at $1,000 per person-rem), those system augments listed in RG 1.110 with a TAC less than $8,330 were evaluated by the applicant to determine if any would be cost-beneficial.

As noted above, the lowest-cost option evaluated by the applicant for gaseous radwaste treatment system augments is a steam generator flash tank vent to main condenser. The TAC for this augment is $6,320; thus to be cost beneficial at $1,000 per person-rem, this augment must remove at least 6.32 person-rem (thyroid), that is to decrease the thyroid dose from 8.33 to 2.01 person-rem. This augment would be to a system not included in the AP1000 design, installation of a flash tank. Therefore, the TAC for this augment is underestimated in the AP1000 design. The AP1000 design instead uses steam generator blowdown heat exchangers that prevent flashing prior to blowdown flow entering the main condenser, effectively performing the same function as the augment. Therefore, the applicant determined that this augment could not provide enough dose reduction to be cost beneficial.

The second option evaluated was a main condenser vacuum pump charcoal/high-efficiency particulate air (HEPA) filtration system, with a TAC of $7,690. Thus, to be cost-beneficial, this augment would need to decrease the thyroid dose by 7.69 person-rem, from 8.33 to 0.64 person-rem. However, as shown in AP1000 DCD Table 11.3-3, sheet 2 of 3, incorporated by reference by the applicant, no iodine would be released through the condenser air removal system, so this augment does not affect the iodine discharged by the plant, which accounts for 2.63 person-rem of the thyroid population dose. Therefore, the applicant determined that the dose reduction necessary to be cost beneficial could not be achieved by this augment.

The third option evaluated was a 1,000 cubic feet per minute (cfm) charcoal/HEPA filtration system, with a TAC of $7,580. Thus, this augment would need to decrease the thyroid dose by 7.58 person-rem, from 8.33 to 0.75 person-rem. The applicant conservatively assumed that this small capacity augment could be placed in the ventilation system at some point where it would eliminate all iodine and particulate releases. However, this augment would not be effective in reducing noble gas, carbon-14, or airborne tritium releases, which together account for 5.59 person-rem of the 8.33 person-rem thyroid population dose. Therefore, the applicant determined that the dose reduction necessary to be cost beneficial could not be achieved by this augment.

The fourth option evaluated was a 600 ft^3 gas decay tank, with a TAC of $7,460. Thus, this augment would need to decrease the thyroid dose by 7.46 person-rem, from 8.33 to 0.87 person-rem. However, as shown in AP1000 DCD Table 11.3-3, incorporated by reference by the applicant, no iodine is released through the waste gas system. This augment does not affect the iodine discharged by the plant, which accounts for 2.63 person-rem of the thyroid population dose. Therefore, the applicant determined that the dose reduction necessary to be cost beneficial could not be achieved by this augment.

The applicant concluded that none of the radwaste augments are cost-beneficial in reducing the annual dose from gaseous effluents for LNP, as they cannot achieve the dose reduction. The staff reviewed these evaluations and concurred that these augments were not sufficiently effective to be cost beneficial considering the cost criterion of $1,000 per person-rem for an augment in 10 CFR Part 50, Appendix I, Section II.D. Thus, the staff concluded that the GWMS meets ALARA requirements and requires no augments.

- LNP COL 11.5-3

The applicant provided additional information in LNP COL 11.5-3 to resolve COL Information Item 11.5-3. COL Information Item 11.5-3 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 NRC staff reviewed the resolution to COL Information Item 11.5-3 related to compliance with Appendix I to 10 CFR Part 50 included under Section 11.3.3.4 of the LNP COL FSAR and issued RAI 11.3-1 requesting the applicant provide the details of the individual and population dose analyses.

In response to RAI 11.3-1, the applicant evaluated 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 LNP 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 LNP 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 only goat milk (based on no milk cows within 5 miles), while the population consumes only cow milk.

In response to RAI 11.3-1, 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 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.

The applicant maximized the estimated MEI doses by choosing conservative locations and dispersion data for the calculations. Since the application was originally submitted, the applicant accumulated an additional two years of meteorological data and revised the atmospheric dispersion and ground deposition factors.

LNP COL FSAR Tables 11.3-201 through 11.3-205 include gaseous pathway parameters used as input to the dose calculation, including population data and site-specific agricultural usage information. The applicant provided detailed justifications for these parameter values in the response to RAI 11.3-1. LNP COL FSAR Tables 11.3-206, 11.3-207, and 11.3-208 list the gaseous pathway doses to the MEI and surrounding population.

The applicant calculated the gaseous pathway doses to the MEI. The results (LNP COL FSAR Tables 11.3-206 and 11.3-207) show, using conservative locations, a gamma annual air dose of 0.0167 milliGray (mGy) (1.67 mrad), a beta annual air dose of 0.0935 mGy (9.35 mrad); a total annual body dose of 0.0306 mSv (3.06 mrem) and an annual skin dose of 0.0839 mSv (8.39 mrem).

The calculated annual population doses listed in LNP COL FSAR Table 11.3-208 are 0.0574 person-Sv (5.74 person-rem) to the total body, and 0.0833 person-Sv (8.33 person-rem) to the thyroid. The applicant uses the population doses in the cost-benefit analysis described in the LNP COL FSAR and evaluated in this SER.

The NRC staff performed an independent assessment using the GASPAR II computer code and compared its results to the applicant's and the Appendix I criteria. The modeling assumptions used and parameter values used were consistent with the applicant's.

In response to RAI 11.3-1, the applicant explained the derivation of values used for agricultural and usage parameters including the total production of vegetables, milk, and meat in the 50-mile area around the site. The staff evaluated and verified the derivation of these values and found them to be reasonable upper bound estimates that are unlikely to be exceeded. Consequently, the staff used the applicant's agricultural and usage values listed in LNP COL FSAR Table 11.3-201 for its 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 FSAR by independently running the GASPAR II computer code with the applicant's parameter values. SER Table 11.3-2 compares the dose estimates resulting from the applicant's analyses to the 10 CFR Part 50, Appendix I criteria. All doses are below the 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 verified the population doses in the LNP 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. SER Table 11.3-3 summarizes the results of the applicant's and staff's analysis of population doses. The NRC 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. RAI 11.3-1 is closed.

The following portion of this technical evaluation section is reproduced from Section 11.3.4 of the VEGP SER:

Supplemental Information

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.

Postulated Radioactive Release Due to a Waste Gas Leak or Failure

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. As documented in the staff's SER for the AP1000 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****.*

Demonstrating Compliance with 10 CFR 20.1301(e)

The staff discusses compliance with 10 CFR 20.1301(e) in Section 11.2.4 of this SER.

Demonstrating Compliance with 10 CFR 20.1302

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.

Resolution of Standard Content Confirmatory Item 11.3-1

Confirmatory Item 11.3-1 is a commitment by the staff to confirm the site-specific characteristics for the LNP site are enveloped by the DCD site parameters. The staff reviewed and compared the LNP site-specific and DCD parameters and confirmed that the site-specific parameters are enveloped by the DCD parameters. As a result, Confirmatory Item 11.3-1 is now closed.

11.3.5 Post Combined License Activities

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

11.3.6 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 GWMS, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff evaluated the additional COL information (STD COL 11.3-1, LNP COL 11.3-1, LNP COL 11.5-3, STD SUP 11.3-1, and STD SUP 11.3-2) in the application against the relevant NRC regulations, acceptance criteria defined in NUREG-0800, Section 11.3, and other NRC regulatory guides. The applicant has satisfactorily addressed all RAIs related to Section 11.3.

STD SUP 11.3-2, related to a postulated radioactive release resulting from a leakage or failure of a waste gas storage tank or offgas charcoal delay bed, is acceptable because it demonstrates compliance with 10 CFR 50.34a.

In other areas of the evaluation of the GWMS, the staff verified that the applicant has 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 60 and 61 of Appendix A to 10 CFR Part 50 and the requirements of 10 CFR 50.34a. The staff finds that the applicant meets the requirements in GDC 60 and 61 by demonstrating conformance to 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, 10 CFR 52.79(a)(3), GDC 60 and 61, and Appendix I to 10 CFR Part 50.

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 LNP COL FSAR, Revision 9, incorporates by reference Section 11.4 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Section 11.4, the applicant provided the following:

AP1000 COL Information Items

- STD COL 11.4-1

The applicant provided additional information in STD COL 11.4-1 to address COL Information Item 11.4-1 (COL Action Item 11.4-1). The additional information provides a process control program (PCP) for both wet and dry solid wastes.

- LNP COL 11.4-1

The applicant provided additional information in LNP COL FSAR Section 11.4.2.4.3 to address alternatives for temporary storage of Class B and C LLRW. In addition, the applicant provided additional information related to conformance with RG 1.143, Revision 2.

Supplemental Information

- STD SUP 11.4-1

The applicant provided supplemental information in LNP COL FSAR Section 11.4.5 to address how the solid radwaste system complies with the guidance in RG 1.143, Revision 2.
STD SUP 11.4-1 also addresses the processes to be followed to ship waste that complies with 10 CFR 61.55 and 10 CFR 61.56 in LNP COL FSAR Section 11.4.6.1.

- LNP SUP 11.4-1

The applicant added supplemental information in LNP COL FSAR Sections 11.4.2.4.3 and 11.4.6.3 describing alternatives for management of Class B and C low-level radioactive waste (LLRW) and long term onsite storage facilities for LLRW, respectively.

License Conditions

- Part 10, License Condition 3, Operational Program Implementation

LNP COL FSAR Section 13.4, Table 13.4-201, "Operational Programs Required by NRC Regulations," identifies Item 9, the PCP, as a program required by regulations that must be implemented by a milestone (prior to initial fuel load) to be identified as a license condition.

- 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:

- 10 CFR Part 20, "Standards for protection against radiation"
- 10 CFR Part 50, "Domestic licensing of production and utilization facilities"
- 10 CFR Part 50, Appendix A, GDC 60
- 10 CFR 52.79(a)(3)
- 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, "Licensing requirements for land disposal of radioactive waste," or equivalent State regulations
- Table 1 and Regulatory Positions C.3.2 and C.3.3 of RG 1.143, Revision 2

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 NRC staff reviewed Section 11.4 of the LNP 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 SWMS. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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
- LNP COL 11.4-1, Alternatives for B and C Wastes
- STD SUP 11.4-1, Quality Assurance
- LNP SUP 11.4-1, Long Term Onsite Storage Facility

In addition to the above items, the staff reviewed the entire section against NUREG-0800, Section 11.4, to determine if the information in LNP COL FSAR Section 11.4 met the regulatory requirements in the regulations stated above (SER Section 11.4.3) 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, Revision 2, as they relate to the definition of the boundary of the SWMS beginning at the interface from plant systems, including multi-unit 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, "Technical specifications on effluents from nuclear power reactors;" 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 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 LNP COL application, there is a difference in how the LNP 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 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 07-10, "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, Revision 3. Specifically, the NRC staff indicated that for COL applications NEI 07-10, 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 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:

<u>Resolution of Standard Content Confirmatory Item 11.4-1</u>

To address Confirmatory Item 11.4-1 in the BLN SER with open items, the applicant updated VEGP COL 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.

<u>Resolution of Standard Content Open Item 11.4-1</u>

To address Open Item 11.4-1 in the BLN SER with open items, the applicant updated VEGP COL 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 contained in VEGP COL 11.4-1 and VEGP SUP 11.4-1 and is evaluated below.

<u>Resolution of Standard Content Open Item 11.4-1</u>

The resolution of Open Item 11.4-1 in the BLN SER with open items is addressed below in the evaluation of LNP COL 11.4-1.

- LNP COL 11.4-1

The applicant's response to RAIs 11.4-1 and 11.4-2 addressed Open Item 11.4-1. The applicant expects to have no need for permanent on-site storage of LLRW. In addition, should the need ever arise to increase the temporary storage capacity of LLRW beyond the capacity of the AP1000 design, the applicant stated they would design and build an onsite temporary storage facility in accordance with the design and operational objectives and guidance in Appendix 11.4-A of NUREG-0800, Section 11.4. The applicant indicated they could perform this work after performing an analysis under 10 CFR 50.59 and, if necessary, request a license amendment.

The applicant indicated in their response that they did not, at the time of application, have offsite disposal capacity for Class B and C LLRW, but that they do have access to offsite disposal of Class A waste. The applicant provided a voluntary supplemental response with additional detail and update of the status of waste disposal access. The staff evaluated the applicant's long-term ability to store Class B and C waste onsite without having to add a temporary storage facility.

The applicant described potential to extend the storage capacity of the AP1000 design for Class B and C waste by prudently managing waste throughput. The applicant indicated that the AP1000 design has more than one year of storage capacity in the Auxiliary Building for Class B and C wet wastes. In addition, the staff's independent analysis of the capacity of the AP1000 Radwaste Building demonstrated that the volume of Class B and C waste comprises less than 2 percent of all LLRW, and determined this is reasonably consistent with the applicant's supplemental response to RAI 11.04-2 modifying FSAR Section 11.4.2.4.3, which conservatively estimates that 5 percent of all solid LLRW generated during operation are Class B and C wastes. The applicant indicated that by frequently disposing of Class A waste, the AP1000 design can accommodate between 10 and 20 years' generation of Class B and C waste in the Radwaste Building. Based on this analysis, the staff concludes that the applicant will not need an additional onsite storage facility for Class B and C waste until a significant portion of the operating life of the plant has transpired.

Should the need for additional onsite storage capacity arise during the lifetime of operation, the licensee described their capability to follow the regulatory process in 10 CFR 50.59 or apply for a license amendment to add more capacity. In its responses to RAIs 11.4-1 and 11.4-2, the applicant committed to follow the guidance in Appendix 11.4-A of NUREG-0800, Section 11.4

for the design and operation of an additional temporary storage facility. The responses provided an additional assessment of the potential capacity needs and contingency arrangements should additional onsite storage become necessary after commencement of operation and proposed further revisions to FSAR Sections 11.4.6 and 11.4.7. In a supplemental response letter dated April 14, 2011, the applicant provided additional detail and revisions to FSAR Section 11.4. Based on the staff's analysis of the long-term storage capacity of the AP1000 and the applicant's commitment to follow the proper design and operational guidance in Appendix 11.4-A of NUREG-0800 should it need to add additional storage capacity, the staff considers Open Item 11.4-1 resolved. The staff verified that the LNP COL FSAR was appropriately revised to reflect the responses to RAIs 11.4-1 and 11.4-2. The staff is tracking the FSAR revisions proposed in the April 14, 2011, voluntary letter as **LNP Confirmatory Item 11.4-1.**

Resolution of LNP Confirmatory Item 11.4-1

LNP Confirmatory Item 11.4-1 is a commitment by the applicant to revise LNP COL FSAR Sections 11.4.2.4.3 and 11.4.6, including adding Section 11.4.6.3, to provide additional information on alternatives for management of Class B and C LLRW and long term onsite storage facilities for LLRW, respectively, as indicated in the April 14, 2011, letter. The staff confirmed that the LNP COL FSAR has been appropriately revised. As a result, LNP Confirmatory Item 11.4-1 is now closed.

In addition to RAIs 11.4-1 and 11.4-2, the staff issued RAI 11.02-5, asking the applicant to provide additional information in response to COL Information Item 11.4-1 in order to ensure that the inventory of radioactive materials contained in all unpackaged waste held in the Radwaste Building is controlled in accordance with the RG 1.143, Revision 2, dose acceptance criteria. As a result, the applicant provided additional information in FSAR Section 11.4.6 indicating that when disposable filtration and adsorption media are removed from radwaste processing systems in the Radwaste Building, that the mobile radwaste processing system not be placed back into service until the media that have been removed are packaged and ready for shipment. In addition, the applicant provided additional information in FSAR Sections 11.2.1.2.5.2 and 13.5.2.2.5 related to controlling the quantity of unpackaged waste in the Radwaste Building as part of the response to RAI 11.02-5. This information and associated operational commitments ensure that all unpackaged waste held in the Radwaste Building are controlled in accordance with the provisions of RG 1.143, Revision 2. The information resolving COL Information Item 11.4-1 was included in FSAR Revision 6. Therefore, COL Information Item 11.4-1 is resolved. A more detailed discussion related to the resolution of RAI 11.02-5 is included in SER Section 11.2.4, above.

The following portion of this technical evaluation section is reproduced from Section 11.4.4 of the VEGP SER:

> *Supplemental Information*
>
> *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.

- LNP SUP 11.4-1

In a December 4, 2009, response to RAI 11.04-2, the applicant explained that, should it need additional onsite storage of LLRW, it could construct an additional onsite storage facility, and that Greater Than Class C LLRW would be addressed similarly to spent fuel.

The applicant made a subsequent voluntary response to RAI 11.04-2 by letter dated April 14, 2011, and proposed a new Section 11.4.6.3 containing information about expanding onsite LLRW storage capacity in the event that disposal facilities or offsite storage facilities are not available.

The staff reviewed the applicant's plans for increasing onsite storage and determined that the applicant would be able to comply with the applicable requirements of 10 CFR Part 20 and 10 CFR Part 50, concerning occupational and public exposures, ALARA programs, and radiological monitoring for onsite and offsite exposures and releases.

Based on the independent analysis and safety review, the NRC staff concludes that the applicant has provided sufficient information to demonstrate that it can safely handle and store LLRW that might accumulate due to unavailability of permanent disposal capacity. The staff considers RAI 11.4-1, RAI 11.4-2, and Open Item 11.4-1 resolved, and the staff assigned tracking of the FSAR revisions proposed in the April 14, 2011, voluntary letter as **LNP Confirmatory Item 11.4-1**.

Resolution of LNP Confirmatory Item 11.4-1

LNP Confirmatory Item 11.4-1 is a commitment by the applicant to revise LNP COL FSAR Sections 11.4.2.4.3 and 11.4.6, including adding Section 11.4.6.3, to provide additional information on alternatives for management of Class B and C LLRW and long term onsite storage facilities for LLRW, respectively, as indicated in the April 14, 2011, letter. The staff confirmed that the LNP COL FSAR has been appropriately revised. As a result, LNP Confirmatory Item 11.4-1 is now closed.

The following portion of this technical evaluation section is reproduced from Section 11.4.4 of the VEGP SER:

License Conditions

- *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, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria [ITAAC]," and is acceptable.

Compliance with 10 CFR Part 50 Appendix I Design Criteria

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 proposes to include the following two license conditions:

- 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 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 SWMS, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff evaluated the COL information (STD COL 11.4-1, LNP COL 11.4-1, STD SUP 11.4-1, and LNP SUP 11.4-1) in the application against the relevant NRC regulations, the acceptance criteria in NUREG-0800, Section 11.4, and other NRC regulatory guides. The applicant has satisfactorily addressed the RAIs related to LNP COL FSAR Section 11.4, Standard Content Confirmatory Item 11.4-1, Open Item 11.4-1, and LNP Confirmatory Item 11.4-1.

Based on the evaluation above, the staff determined that the applicant's means for handling radioactive solid waste during normal operations, including anticipated operational occurrences, are consistent with GDC 60. In accordance with 10 CFR 52.79(a)(3), the staff also determined 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 Appendix A of 10 CFR Part 50; 10 CFR 50.34a; 10 CFR 52.79(a)(3);10 CFR 20.1301(e); 10 CFR 20.1406; Appendix I to 10 CFR Part 50; 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 system includes subsystems used to collect process and effluent samples during normal operation and anticipated operational occurrences, and under post-accident conditions.

11.5.2 Summary of Application

Section 11.5 of the LNP COL FSAR, Revision 9, incorporates by reference Section 11.5 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Section 11.5, the applicant provided the following:

Departure

- LNP DEP 6.4-1

The applicant provided additional information in Section 11.5 of the LNP COL FSAR about LNP 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 LNP 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.

- LNP COL 11.5-2

The applicant provided additional information in LNP COL 11.5-2 to add language to LNP COL FSAR Section 11.5.3 addressing extension of the applicant's program for QA of radioactive effluent and environmental monitoring for their existing licensed facilities to apply to LNP Units 1 and 2.

- LNP COL 11.5-3

The applicant provided additional information in LNP 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

LNP 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 that are 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 the NRC's 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 additional information on radiation monitoring is established in:

- 10 CFR Part 50, Appendix A, GDC 64, "Monitoring radioactivity releases"
- 10 CFR Part 20
- 10 CFR Part 50
- 10 CFR Part 52, "Licenses, certifications, and approvals for nuclear power plants"
- 10 CFR Part 61
- 10 CFR Part 71
- 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 are identified in NUREG-0800, Section 11.5.

11.5.4 Technical Evaluation

The NRC staff reviewed Section 11.5 of the LNP 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 radiation monitoring system. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The staff's review of this application included the following information in the LNP COL FSAR:

- STD COL 11.5-1, ODCM
- STD COL 11.5-2, Programmatic Aspects of the Effluent Monitoring and Sampling Program
- LNP COL 11.5-2, which adds language to LNP COL FSAR Section 11.5.3 addressing extension of the applicant's existing program for quality assurance of radioactive effluent and environmental monitoring to apply to LNP Units 1 and 2.
- LNP 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 LNP 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 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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, "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, 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, 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). 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.*

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.*

- LNP COL 11.5-2

In LNP COL 11.5-2, in addition to accepting NEI 07-09A, the applicant extended its existing, NRC-accepted program for QA, including RG 4.15, Revision 1, for effluent and environmental monitoring, to LNP Units 1 and 2. By using the current program, the applicant will avoid confusion and the potential for error because the program for the existing and planned units will share the same equipment and personnel. The staff finds this acceptable.

- LNP COL 11.5-3

The applicant provided additional information in LNP 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 LNP COL FSAR Sections 11.2.3.5 and 11.3.3.4 for liquid and gaseous effluents, respectively.

The NRC staff reviewed the resolution of LNP COL 11.5-3 related to compliance with 10 CFR Part 50, Appendix I, as discussed in SER Sections 11.2.4 and 11.3.4, 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:

Section 11.5.4.2, Representative Sampling

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.

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:

License Conditions

- *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 proposes to include the following two license conditions:

- 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 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 radiation monitoring system, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP 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, LNP COL 11.5-2, and LNP COL 11.5-3) in the application against the relevant NRC regulations, the 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 and Confirmatory Item 11.5-1.

LNP 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 Appendix A of 10 CFR Part 50; applicable requirements of 10 CFR Parts 20, 50, and 52; the guidance in ANSI/HPS N13.1, ANSI N42.18, RGs 1.21 and 4.15; and applicable acceptance criteria in NUREG-0800, Section 11.5.

Table 11.2-1. Dose from Consumption of Fish from Lower Withlacoochee River from Postulated Tank Release

Radionuclide	Surface Water Concentration in the Lower Withlacoochee River [C][1] (pCi/liter)	Total Body Dose Conversion Factor [DCF][2] (mrem/pCi)	Bio-accumulation Factor [BF][3] (pCi/kg per pCi/liter)	Dose from consumption of fish from Withlacoochee River for 1 year [DF][4] (mrem)
H-3	1.8E-5	1.05E-7	9.0E-1	3.6E-11
I-129	8.5E-8	9.21E-6	1.5E+1	2.5E-10
Mn-54	5.7E-87	8.73E-7	4.0E+2	~ 0
Fe-55	1.3E-29	4.43E-7	1.0E+2	~ 0
Co-60	6.0E-17	4.72E-6	5.0E+1	~ 0
Sr-90	9.6E-48	1.86E-3	3.0E+1	~ 0
Total				2.9E-10

1) Surface water concentrations from LNP COL FSAR Table 2.4.13–204.
2) Ingestion dose conversion factors for adults from RG 1.109, Table E-11, except for DCF for I-129, which is from NUREG-0172, Table 4.
3) Bio-accumulation factors for freshwater fish from RG 1.109, Table A-1.
4) DF = C x DCF x BF x 21 kg/year. The 21 kg/year of fish consumption is the amount consumed by an adult MEI (from RG 1.109, Table A-1).

Table 11.2-2. Comparison of Important Modeling Assumptions Used to Demonstrate Compliance with 10 CFR Part 50, Appendix I Criteria

Pathways and Parameters	Application	NRC Staff's Analysis
Drinking water pathway for MEI and population	No	No
Fish ingestion pathway for MEI and population	Yes	Yes
Recreational use of river for MEI and population	Yes	Yes
Irrigation pathway for the MEI (including irrigated vegetable ingestion and ingestion of milk and meat from livestock grazing on irrigated land)	No	No
Surface Water Dilution Model	Mixing in CR-3 Discharge Canal	Mixing in CR-3 Discharge Canal

Table 11.2-3. Modeling Parameter Values Used to Demonstrate Compliance with 10 CFR Part 50, Appendix I Criteria *

Parameter	Value	Basis
Annual radionuclide release (Ci/yr)	Multiple values	DCD Table 11.2-7
Effluent discharge rate (cfs)	63	FSAR Table 11.2-201
Dilution factors	21	FSAR Table 11.2-201
Transit time (hr)	0	FSAR Table 11.2-202
Reconcentration model	None	FSAR Table 11.2-202
Sport fishing harvest (kg/yr)	210,246	FSAR Table 11.2-202
Commercial fishing harvest (kg/yr)	734,960	FSAR Table 11.2-202
Sport Invertebrate harvest (kg/yr)	142,438	FSAR Table 11.2-202
Commercial Invertebrate harvest (kg/yr)	1,424,384	FSAR Table 11.2-202
Swimming/Boating/Shoreline usage (person-hours per year)	32,071,440 for Boating 32,541,940 for Others	FSAR Table 11.2-202

* Staff used LADTAP II default values for parameters not listed in the table

Table 11.2-4. Comparison of Maximum Individual Doses (mrem/yr) Used to Demonstrate Compliance with 10 CFR Part 50, Appendix I Criteria

Organ/Body	Application*	10 CFR Part 50, Appendix I, Section II.A
GI-LLI	7.14E-02	10
Total Body	5.20E-03	3
Thyroid	1.27E-02	10

* Taken from LNP COL FSAR Table 11.2-203

Table 11.2-5. Comparison of Maximum Individual Doses to 40 CFR Part 190 (mrem/yr)

Organ/Body	Application*	40 CFR Part 190
Total Body	5.5	25
Thyroid	12.9	75
Other Organ (Child - Bone)	19.5	25

* Taken from LNP COL FSAR Table 11.2-205

Table 11.3-1. Population Doses Breakdown by Source

Source	Total Body (person-rem)	Thyroid (person-rem)	Percent of Total Thyroid Dose
Noble Gases	1.02E+00	1.02E+00	12 Percent
Iodine	5.08E-03	2.63E+00	32 Percent
Particulates	1.33E-01	9.83E-02	1 Percent
C-14	3.48E+00	3.48E+00	42 Percent
H-3	1.09E+00	1.09E+00	13 Percent
Total	5.74E+00	8.33E+00	100 Percent

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) • Beta Dose (mrad) • Total Body (mrem) • Skin (mrem)	 1.67* 9.35* 3.06** 8.39**	 10 20 5 15
Radioiodines and Particulates • Maximum Organ (mrem)	 9.71***	 15

* Taken from LNP COL FSAR Table 11.3-207
** Taken from LNP COL FSAR Table 11.3-206
*** Dose for the child bone (conservatively includes plume exposure pathway)

Table 11.3-3. Comparison of Population Doses (person-rem/yr)

Organ/Body	Application*	NRC Staff's Analysis
Total Body	5.74	5.75
Thyroid	8.33	8.08

* Taken from LNP COL FSAR Table 11.3-208

12.0 RADIATION PROTECTION

This chapter provides information on radiation protection methods and estimated occupational radiation exposures (OREs) of operating and construction personnel during 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). 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-Reasonably Achievable (Related to RG 1.206, Section C.III.1, Chapter 12, C.I.12.1, "Ensuring 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 Levy Nuclear Plant (LNP) combined license (COL) Final Safety Analysis Report (FSAR).

12.1.2 Summary of Application

Section 12.1 of the LNP COL FSAR, Revision 9, incorporates by reference Section 12.1 of the AP1000 Design Control Document (DCD), Revision 19.

In addition, in LNP COL FSAR Section 12.1, the applicant provided the following:

AP1000 COL Information Item

- 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), which addresses ALARA and operational policies and compliance with RGs. The applicant provided additional information to incorporate 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 LNP COL FSAR Section 12.1 and NEI 07-03A, "Generic FSAR Template Guidance for Radiation Protection Program Description," in Appendix 12AA.

The applicant also added information in their FSAR that was different from NEI 07-08A, Section 12.1.2, to state that ALARA procedures are consistent with 10 CFR 20.1101 and the quality assurance criteria described in Part III of the Quality Assurance Program Description. This change is consistent with the applicable requirements.

Supplemental Information

- STD SUP 12.1-1

The applicant provided supplemental (SUP) information by addressing equipment layout 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."

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the ALARA program are given in Section 12.1 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)."

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
- 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 ALARA," Revision 3
- RG 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures ALARA," Revision 1-R
- NUREG-1736, "Consolidated Guidance: 10 CFR Part 20 – Standards for Protection Against Radiation"

12.1.4 Technical Evaluation

The NRC staff reviewed Section 12.1 of the LNP 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 ensuring that the ORE to personnel will be kept ALARA. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 12.1.4 of the VEGP SER:

> *The following portion of this technical evaluation section is reproduced from Section 12.1.4 of the BLN SER.*

[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.

<u>AP1000 COL Information Item</u>

- *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 Section 12.1.4 of the VEGP SER:

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 RPP) are discussed in Section 12.5.5 of this SER.

12.1.6 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 ALARA, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP COL FSAR is acceptable based on the relevant acceptance criteria provided in Section 12.1 of NUREG-0800. 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 LNP COL FSAR and meets the applicable regulatory requirements and guidance specified in Sections 12.1.3 and 12.1.4 of this SER.

- 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 <u>Radiation Sources</u>

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

Section 12.2 of the LNP COL FSAR, Revision 9, incorporates by reference Section 12.2 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Section 12.2, the applicant provided the following:

Departures

- LNP DEP 6.4-1

The applicant provided additional information in Section 12.2 of the LNP COL FSAR about LNP 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 LNP 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), which 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.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the radiation sources are given in Section 12.2 of NUREG-0800.

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 Criterion (GDC) 61, "Fuel Storage and Handling and Radioactivity Control"

12.2.4 Technical Evaluation

The NRC staff reviewed Section 12.2 of the LNP 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 radiation sources. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 12.2.4 of the VEGP SER:

> *The following portion of this technical evaluation section is reproduced from Section 12.2.4 of the BLN SER:*
>
> *AP1000 COL Information Item*
>
> - *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 Section 12.2.4 of the VEGP SER:

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 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 radiation sources, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The staff concludes that the relevant information presented in the LNP COL FSAR is acceptable based on the relevant acceptance criteria provided in Section 12.2 of NUREG-0800. The staff based its conclusion on the following:

- LNP 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 12.2-1, which addresses miscellaneous sources, is acceptable because the applicant has incorporated the approved reference NEI 07-03A into the LNP COL FSAR and meets the requirements of 10 CFR 20.1801, 10 CFR 20.1802, and GDC 61.

12.3 Radiation Protection Design Features

Section 12.3, "Radiation Protection Design Features" and the following Section 12.4, "Dose Assessment," are treated as separate sections in the SER (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 both RG 1.206 and NUREG-0800, with the material discussed under the section "Dose Assessment" included in a section at the end of Section 12.3.

12.3.1 Introduction

This section addresses the issues related to radiation protection equipment and design features used to ensure that OREs are ALARA. It takes into account design dose rates, AOOs, and accident conditions. These issues include the facility design features, shielding, ventilation, area radiation and airborne radioactivity monitoring instrumentation, and dose assessment.

12.3.2 Summary of Application

Section 12.3 of the LNP COL FSAR, Revision 9, incorporates by reference Section 12.3 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Section 12.3, the applicant provided the following:

Departures

- LNP DEP 6.4-1

The applicant provided additional information in Section 12.3 of the LNP COL FSAR about LNP 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 LNP 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 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), which 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), which 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, which 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, which addresses the program to ensure documentation of operational events deemed to be of interest for decommissioning.

- LNP SUP 11.2-1

The applicant provided supplemental information in LNP SUP 11.2-1 describing the liquid radwaste system discharge piping exiting the Radwaste Building and the wastewater system blowdown line piping running to the plant outfall at the Crystal River Energy Complex discharge canal.

12.3.3 Regulatory Basis

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

In addition, the acceptance criteria associated with the relevant requirements of Commission regulations for radiation protection design features are given in Section 12.3 of NUREG-0800.

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
- 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”
- 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"

12.3.4 Technical Evaluation

The NRC staff reviewed Section 12.3 of the LNP 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 radiation protection design features. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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 12.3-1, Administrative Controls for Radiological Protection
- STD COL 12.3-2, Criteria and Methods for Radiological Protection
- STD COL 12.3-3, Groundwater Monitoring Program
- LNP SUP 11.2-1, Supplemental Information on Exterior Radwaste Discharge Piping
- STD COL 12.3-4, Record of Operational Events of Interest for Decommissioning

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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 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:*
>
> *<u>AP1000 COL Information Items</u>*
>
> - *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 Section 12.3.4 of the VEGP SER:

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 [sic] *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 Section 12.3.4 of the VEGP SER:

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.

Evaluation of Site-Specific Information in Standard Content Evaluation (LNP SUP 11.2-1)

The applicant provided supplemental information in LNP SUP 11.2-1 that describes both the exterior radwaste discharge piping and the cooling tower blowdown piping. The exterior radwaste discharge piping runs from the auxiliary building to the radwaste building and then from the radwaste building to the cooling tower blowdown piping. LNP SUP 11.2-1 describes that portion of the cooling tower blowdown piping from the point where the radwaste discharge piping connects to the blowdown piping (where the radwaste is diluted with the cooling tower blowdown) to the discharge point at the Crystal River Energy Complex discharge canal.

The last paragraph of the standard content evaluation of STD COL 12.3-3, reproduced from Section 12.3.4 of the BLN SER above, provides the staff's evaluation of the exterior radwaste discharge piping for BLN. In an April 27, 2009, letter to NRC, the LNP applicant endorsed the December 16, 2008, standard content portions of the BLN supplemental response to RAI 12.3-12.4-1. In May 4, 2011, and December 7, 2011, letters to NRC, the LNP applicant provided additional plant-specific information regarding the radwaste discharge piping and cooling tower blowdown piping and proposed to modify LNP SUP 11.2-1 in FSAR Section 11.2.1.2.4 to address this additional information. In Section 11.2.4 of this SER, the staff states that LNP SUP 11.2-1 is evaluated in SER Section 12.3.

In LNP SUP 11.2-1, the applicant stated that the exterior radwaste discharge piping is enclosed within a guard pipe and is monitored for leakage. As discussed above, this design is in compliance with 10 CFR 20.1406 and the staff finds it to be acceptable. The cooling tower blowdown piping, which runs approximately 13 miles to the discharge point at the Crystal River Energy Complex discharge canal, will be a buried, 54-inch diameter single-walled pipe constructed of High Density Polyethylene. There will be two cooling tower blowdown pipes, one for each Levy unit. A manual vent valve will be installed just upstream of the elevation drop on each blowdown pipe where the blowdown pipes travel beneath the Cross Florida Barge Canal. These vent valves are included to remove air either coming out of solution or air introduced by the vacuum breakers (located on each blowdown pipe at the high point of the system upstream of the point where the radwaste discharge pipe connects with the blowdown pipe) in the event that the air is not swept out of the blowdown line during system startup. The vent valve in each blowdown line is located where air would be most likely to collect in the line. These vent valves would be installed in manholes and therefore would not protrude from the ground. During normal operation, the vent lines are capped and the vent valves are locked closed to prevent any spillage. As required during pump startup, personnel will be present at the vent valves to allow air to escape from the blowdown lines and then to close the valves when the vent lines fill with water. The applicant stated that any spillage from the vent valves shall be contained and properly disposed of. The applicant also stated that leak detection of the blowdown pipe will be accomplished by ground water monitoring, as part of the groundwater monitoring program, and by performing periodic walkdowns of the vent valves, in accordance with NEI 08-08A. In order to maintain these vent valves in good operating condition and thereby reduce the potential for undetected leakage from the vent valves to the environment, in accordance with the requirements of 10 CFR 20.1406, the applicant will include these vent valves in the site’s routine maintenance program. 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. NEI 08-08A states that the COL applicant will establish an on-site ground water monitoring program to ensure timely detection of inadvertent radiological releases to the ground water. On the basis that LNP Supplement 11.2-1 states that the applicant will cap and keep the vent valves closed when not being used, will maintain these valves in good operating condition to minimize the potential for leakage, and will implement a ground water monitoring program for the cooling tower blowdown pipe, the staff finds that the information provided in LNP Supplement 11.2-1 complies with the requirements of 10 CFR 20.1406 and is therefore acceptable. Until the applicant includes the modified LNP SUP 11.2-1 in a future version of the FSAR, this will be tracked as **LNP Confirmatory Item 12.3-1.**

<u>Resolution of LNP Confirmatory Item 12.3-1</u>

LNP Confirmatory Item 12.3-1 involves an applicant commitment to revise section 11.2.1.2.4 of the FSAR to include additional plant-specific information regarding the radwaste discharge piping and cooling tower blowdown piping. The staff verified that the FSAR had been appropriately revised. As a result, LNP Confirmatory Item 12.3-1 is closed.

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 Section 12.3.4 of the VEGP SER:

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.

12.3.5 Post Combined License Activities

The post-COL activities related to the RPP are discussed in SER Section 12.5.5.

12.3.6 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 radiation protection design features and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP COL FSAR is acceptable based on the relevant acceptance criteria provided in Section 12.3 of NUREG-0800. The staff based its conclusion on the following:

- LNP 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 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 LNP COL FSAR in order to demonstrate conformance with the applicable regulatory requirements and guidance specified in Sections 12.3.3 and 12.3.4 of this SER.

- 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 SER.

- STD COL 12.3-3 and LNP SUP 11.2-1, which address the groundwater monitoring program beyond the normal radioactive effluent monitoring program, are 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 SER.

- 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 LNP COL FSAR in order to demonstrate conformance with the applicable regulatory requirements and guidance specified in Sections 12.3.3 and 12.3.4 of this SER.

12.4 Dose Assessment

12.4.1 Introduction

This section addresses the issues related to estimating the annual personal 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

Section 12.4 of the LNP COL FSAR, Revision 9, incorporates by reference Section 12.4 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Section 12.4, the applicant provided the following:

Supplemental Information

- LNP SUP 12.4-1

The applicant provided supplemental information to address dose to construction workers by adding new sections after 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 the NUREG-1793 and its supplements.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for the dose assessment are given in Section 12.4 of NUREG-0800.

The applicable regulatory requirements for LNP 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 NRC staff reviewed Section 12.4 of the LNP 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 dose assessment. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting.

The staff reviewed the information in the LNP COL FSAR:

Supplemental Information

- LNP SUP 12.4-1

The applicant provided supplemental information regarding dose to construction workers in LNP COL FSAR Section 12.4.1.9 (Sections 12.4.1.9.1 through 12.4.1.9.5), "Dose to Construction Workers." Section 12.4.1.9.1 describes the site layout as depicted in Figure 2.1.1-203 of the LNP COL FSAR. The sources of radiation exposure to the construction workers are described in Section 12.4.1.9.2. However, since there is not another operating facility on the plant site, there is no source of radiation exposure until LNP Unit 1 is operating. Section 12.4.1.9.3 includes the assumptions used to calculate annual exposure estimates, and Section 12.4.1.9.4 identifies the regulatory requirements that are applicable to the construction workers. Section 12.4.1.9.5 identifies the collective annual exposure estimate for all workers and compares an individual's exposure estimate with the applicable limits.

In LNP COL FSAR Section 12.4.1.9.2, “Radiation Sources,” the sources of radiation that could be encountered by construction workers are identified. There is no radiation source until LNP Unit 1 is operational, so LNP Unit 1 construction workers would not receive any radiation exposure from an adjacent nuclear facility. LNP Unit 2 construction workers would potentially receive radiation exposure from direct radiation, gaseous effluents, and liquid effluents from the operation of LNP Unit 1. The applicant stated that there is no direct radiation from the containment or other buildings, as described in AP1000 DCD Section 12.4.2. Gaseous effluents have been identified as contributing to occupational exposure, but liquid effluents are not expected to contribute.

The staff agrees that there is no significant source of radiation exposure for LNP Unit 1 construction workers from an adjacent nuclear facility. The staff issued RAI 12.3-12.4-1 to obtain further information to support the applicant’s conclusions that the radiation doses to LNP Unit 2 construction workers from direct radiation and liquid effluents would be negligible. In RAI 12.3-12.4-1, the staff questioned the applicant’s statements that the direct radiation contribution from the containment and other plant buildings is negligible, and that no exposure to workers would occur from performing the tie-in of LNP Unit 2 liquid effluent piping. In a letter dated September 3, 2009, the applicant responded to the RAI by stating that according to the DCD, the radiation levels outside the containment building would be less than 0.25 millirem (mrem)/hour (hr). The dose at the fence around the LNP Unit 1 protected area, which is further removed, would be negligible; and, as a result, the direct dose contribution to the construction workers would be negligible. The applicant also stated that no exposure to LNP Unit 2 construction workers performing the tie-in of LNP Unit 2 effluent piping would occur because these activities would be completed prior to LNP Unit 1 start of operation. The staff agrees that the potential direct radiation dose from LNP Unit 1 at the LNP Unit 2 construction site, including potential dose from the tie-in of LNP unit 2 effluent piping, would be negligible because of the separation of the two facilities (0.25 miles). The staff finds that these are reasonable bases to substantiate that construction worker doses would be negligible from these sources.

In LNP COL FSAR Section 12.4.1.9.3, “Construction Worker Dose Estimates,” the applicant identifies the methodology used for the construction worker dose estimate as a result of LNP Unit 1 gaseous effluents. Although LNP Unit 2 would be situated 402 meters (m) (1320 feet [ft.], 0.25 mile) directly north of LNP Unit 1, the applicant chose to conservatively utilize the highest χ/Q for any sector at that distance for the dose estimates. In this case, that is a χ/Q of 1.52E-04 seconds per cubic meter (s/m^3) in the worst meteorological sector (WSW) direction. This section also identified that doses were adjusted by a factor of 0.24 to account for annual construction worker residence time of 2080 hours of work per 8760 hours in a calendar year. The staff has reviewed the methodologies and assumptions used to estimate the worker doses. The use of the worst case (WSW) χ/Q results in a conservative assessment of the estimated worker dose (in the N sector) and is an acceptable approach.

In addition, in RAI 12.3-12.4-1, the staff also requested further information about the assumptions and bases used for selecting the exposure distances (402 m) and exposure time (2080 hours per year) to calculate the construction worker exposures, considering the potential overtime that is typically used on such construction projects. In the letter dated September 3, 2009, the applicant responded to the RAI by stating that 402 m is the distance from LNP Unit 1 to the center of the LNP Unit 2 nuclear island, which is where the majority of the construction activities would take place. The applicant also stated that construction worker exposure time of 2080 hours was selected to represent 40 hours per week for 52 weeks. Given the magnitude of the calculated dose estimates, overtime of up to 84 hours per week would still result in exposures well below applicable limits for members of the public. On the basis of the applicant's response, the staff concludes that the distance of 402 m used to calculate the χ/Q values is an accurate representation of the distance from the LNP Unit 1 where the majority of the construction workers for LNP Unit 2 will be located. The staff also agrees that even though some construction workers may work more than the estimated 2080 hours per year, the resulting increased doses to these workers would still be well below applicable dose limits for members of the public.

In LNP COL FSAR Section 12.4.1.9.4, "Compliance with Dose Regulations," the applicant states that the annual construction worker dose meets the requirements for members of the public as stated in 10 CFR 20.1301; 10 CFR Part 50, Appendix I; and Title 40 of the *Code of Federal Regulations* (40 CFR) Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations," during construction of LNP Units 1 and 2; and therefore, the workers would not be classified as radiation workers nor would they require monitoring. The staff agrees with this statement.

LNP COL FSAR Section 12.4.1.9.5, "Collective Doses to LNP Unit 2 Workers," identifies the total exposure that is expected to be received by all construction workers on site, given the estimated worker dose contribution from gaseous effluents and assumed workers and hours spent at the construction site. The collective construction worker exposure of 0.119 person-Sievert (person-Sv)(11.9 person-rem) is shown to be less than 1.5 percent of the average annual collective exposure from background sources (both natural and man-made). Section 12.4.1.9.5 also refers to LNP COL FSAR Table 12.4-201, "Comparison of LNP Construction Worker Estimated Radiation Doses to 10 CFR 20.1301 Public Dose Criteria," which describes the construction worker exposure estimate results and compares them to the 10 CFR 20.1301 requirements. This table identifies that the expected construction worker exposure is 4.4 mrem/year (yr) and 2.1E-3 mrem in any one hour, compared to 10 CFR 20.1301 limits of 100 mrem/yr and 2 mrem in any one hour. The staff has performed an independent analysis of the construction worker dose estimates and determined that the dose values estimated by the applicant are consistent with the staff's estimates.

In summary, the NRC staff determined that the information provided in LNP SUP 12.4-1, regarding dose to construction workers, in the new Section 12.4.1.9, is acceptable. In accordance with the discussion in the above paragraphs, RAI 12.3-12.4-1 is closed.

The following portion of this technical evaluation section is reproduced from Section 12.4.4 of the VEGP SER:

- *STD SUP 12.4-1*

The following portion of this technical evaluation section is reproduced from Section 12.4.4 of the BLN SER:

- *BLN SUP 12.4-1*

In RAI 12.3-12.4-6, the applicant was requested to describe the program that will ensure the construction workers will be monitored and that exposures will be minimized and maintained ALARA in accordance with 10 CFR 20.1101(b). This is identified as ***Open Item 12.4-1****.*

<u>Resolution of Open Item 12.4-1</u>

In a letter dated July 16, 2009, the applicant proposed to add supplemental information to Section 12.4.1.9.5 of the VEGP COL FSAR 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 LNP.

12.4.5 Post Combined License Activities

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

12.4.6 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 dose assessment, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP COL FSAR is acceptable based on the relevant acceptance criteria provided in Section 12.3 of NUREG-0800. The staff based its conclusion on the following:

- LNP SUP 12.4-1, which provides supplemental information to address dose to construction workers, is acceptable because the staff concludes that the doses to workers will be in accordance with the 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 incorporates this information into the LNP COL FSAR in order to meet 10 CFR 20.1301 and 10 CFR 20.1302.

12.5 Health Physics Facilities Design (Related to RG 1.206, Section C.III.1, Chapter 12, C.I.12.5, "Operational Radiation Protection Program")

12.5.1 Introduction

This section addresses the objectives and design of the health physics (HP) facilities. The HP facilities are designed with the objectives of:

- Providing 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.

- Providing 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

Section 12.5 of the LNP COL FSAR, Revision 9, incorporates by reference Section 12.5 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Section 12.5, the applicant provided the following:

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 Condition 3, Items C.1, D.2, G.4, and K.1

The actual milestones for the RPP are listed in Table 13.4-201.

- 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 the Commission regulations for the HP facilities design are given in Section 12.5 of NUREG-0800.

The applicable regulatory requirements and guidance for STD COL 12.5-1 and LNP COL 12.5-1 are as follows:

- 10 CFR Part 20
- RG 8.2
- RG 8.4, "Direct Reading and Indirect Reading Pocket Dosimeters"
- RG 8.6, "Standard Test Procedures for Gieger-Muller Counters"
- RG 8.8, Revision 3
- RG 8.9, Revision 1
- RG 8.10, Revision 1-R
- RG 8.28, "Audible Alarm Dosimeters"
- NUREG-1736

The applicable regulatory requirement for License Condition 3, Items C.1, D.2, G.4, and K.1 is as follows:

- 10 CFR 20.1101

12.5.4 Technical Evaluation

The NRC staff reviewed Section 12.5 of the LNP 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 HP facilities design. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 12.5.4 of the VEGP SER:

> *The following portion of this technical evaluation section is reproduced from Section 12.5.4 of the BLN SER:*
>
> <u>*AP1000 COL Information Item*</u>
>
> - *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)~~ 20.1101(b)* [corrected]*, 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~~ N13.7-1981 [corrected] *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 Section 12.5.4 of the VEGP SER:

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.*

<u>Resolution of VEGP Confirmatory Item 12.5-2</u>

Appendix 1AA of the LNP COL FSAR correctly references American National Standards Institute (ANSI) N13.27-1981 under the conformance discussion of RG 8.28. Therefore, VEGP Confirmatory Item 12.5-2 is resolved for the LNP COL application.

The following portion of this technical evaluation section is reproduced from Section 12.5.4 of the VEGP SER:

> <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 Section 12.5.4 of the VEGP SER:

Resolution of Standard Content Confirmatory Item 12.5-1

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.

- *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 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 radiation protection design features, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP COL FSAR is acceptable based on the relevant acceptance criteria provided in Section 12.5 of NUREG-0800. The staff based its conclusion on the following:

- STD COL 12.5-1, which addresses the RPP description, is acceptable because the applicant has demonstrated compliance with the applicable regulatory requirements and guidance specified in Sections 12.5.3 and 12.5.4 of this SER.

13.0 CONDUCT OF OPERATIONS

13.1 Organizational Structure of Applicant

13.1.1 Introduction

The 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. Its 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 also describes the structure, functions, and responsibilities of the onsite organization established to operate and maintain the plant.

13.1.2 Summary of Application

Section 13.1 of the Levy Nuclear Plant (LNP) combined license (COL) Final Safety Analysis Report (FSAR), Revision 9, incorporates by reference Section 13.1 of the AP1000 Design Control Document (DCD), Revision 19.

In addition, in LNP COL FSAR Section 13.1, the applicant provided the following:

AP1000 COL Information Items

- LNP COL 13.1-1

The applicant[1] provided additional information in LNP 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. LNP COL 13.1-1 describes organizational positions of the nuclear power station and owner/applicant corporations and associated functions and responsibilities.

- LNP COL 9.5-1

The applicant provided additional information in LNP COL 9.5-1, describing the fire protection program in Section 9.5.1.8. For this LNP COL item, the applicant added a new Section 13.1.1.2.10, “Fire Protection.” LNP COL 9.5-1 is also addressed in Section 13.1.2.1.4.9, “Supervisor - Fire Protection.” Table 1.8-202, “COL Item Tabulation,” provides LNP COL 9.5-1 cross-references.

- LNP COL 18.6-1

The applicant provided additional information in LNP COL 18.6-1, describing the qualifications of the nuclear plant technical support personnel. LNP COL 18.6-1 is addressed under Section 13.1.1.4, “Qualifications of Technical Support Personnel”; Section 13.1.3.1, “Minimum Qualification Requirements”; Section 13.1.3.2, “Qualification Documentation”; and Table 13.1-201, “Generic Position/Site-Specific Position Cross-Reference.” Table 1.8-202, “COL Item Tabulation,” provides LNP COL 18.6-1 cross-references.

- LNP COL 18.10-1

The applicant provided additional information in LNP COL 18.10-1 to address the responsibilities of the manager in charge of nuclear training. LNP COL 18.10-1 is addressed in Section 13.1.1.3.2.4, “Manager – Training LNP.” Table 1.8-202, “COL Item Tabulation,” provides LNP 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 [FSER] Related to Certification of the AP1000 Standard Design,” and its supplements.

---

[1] The applicant, Duke Energy Florida, LLC, was formerly identified as Duke Energy Florida, Inc., and Progress Energy Florida, Inc. In a letter dated April 15, 2013, Progress Energy Florida notified the NRC that its name was changing to Duke Energy Florida, Inc., effective April 29, 2013. The name change and a 2012 corporate merger between Duke Energy and Progress Energy are described in Chapter 1 of the SER. Because a portion of the review described in this chapter was completed prior to the name change, the NRC staff did not change references to “Progress Energy Florida” or “PEF” to “Duke Energy Florida” or “DEF” in this chapter.

In addition, the acceptance criteria associated with the relevant requirements of the Commission regulations for LNP COL 13.1-1, LNP COL 9.5-1, LNP COL 18.6-1, and LNP COL 18.10-1 are given in Sections 13.1.1, "Management and Technical Support Organization," and 13.1.2-13.1.3, "Operating Organization," of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants."

The applicable regulatory guidance for the organizational structure of the applicant is as follows:

- American National Standards Institute/American Nuclear Society (ANSI/ANS) -3.1-1993, 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 50.48, "Fire Protection"
- 10 CFR 50.71, "Maintenance of records, making of reports"
- 10 CFR 50.50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"
- 10 CFR 52.47, "Contents of applications; technical information"
- 10 CFR 50.54, "Conditions of licenses"
- 10 CFR 52.79, "Contents of applications; technical information in final safety analysis report"
- RG 1.33, Revision 2, "Quality Assurance Program Requirements (Operation)"
- 10 CFR 55, "Operator's Licenses."
- Regulatory Guide 1.8, "Qualification and Training of Personnel for Nuclear Power Plants."
- Regulatory Guide 1.28, "Quality Assurance Program Criteria (Design and Construction)."
- Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)."
- Regulatory Guide 1.68, "Initial Test Programs for Water-cooled Nuclear Power Plants."
- Regulatory Guide 1.114, "Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit."
- Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."
- Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."
- Regulatory Guide 1.175, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing."
- Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications."
- Regulatory Guide 1.178, "An Approach for Plant-Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping."
- Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants."

- Regulatory Guide 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, "Clarification of TMI Action Plan Requirements."
- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition."

13.1.4 Technical Evaluation

The Nuclear Regulatory Commission (NRC) staff reviewed Section 13.1 of the LNP 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.[2] The NRC staff's 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 NRC staff's evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the LNP COL FSAR:

AP1000 COL Information Items

- LNP COL 13.1-1

The NRC staff reviewed LNP COL 13.1-1 related to the organizational structure of the COL applicant included under Section 13.1 of the LNP COL FSAR. Section 13.1 of the LNP COL FSAR describes the organizational positions of a nuclear power plant and owner/applicant corporations and associated functions and responsibilities.

The applicant provided the following additional LNP 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 Appendix F of NUREG-1793, which states:

> The COL applicant will describe its organizational structure.

---

[2] 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 applicant provided additional information as part of the LNP 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 LNP 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 LNP COL FSAR Section 13.1 beyond the structure given in RG 1.206, “Combined License Applications for Nuclear Power Plants (LWR [light-water reactor] Edition).” The new section titles are:

13.1.1, “Management and Technical Support Organization”
13.1.2, “Operating Organization”
13.1.3, “Qualifications of Nuclear Plant Personnel”
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”
Figure 13.1-202, “Shift Operations Organization”
Figure 13.1-203, “Corporate and Engineering Organization”
Figure 13AA-201, “Construction Management Organization”
Figure 13AA-202, “Hiring Schedule for Plant Staff”

In addition, the applicant added a new appendix to Chapter 13 titled “Appendix 13AA Construction-Related Organization.” This appendix describes the applicant’s construction organization. Once plant operation commences, this appendix will become historical information.

The NRC staff has reviewed LNP 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) and 10 CFR 50.80, “Transfer of licenses,” as applicable. This conclusion is based on the following:

The applicant has 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 has described its plans for managing the project and utilizing 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 has 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 Table 1.9-202, “Conformance with SRP Acceptance Criteria,” of the LNP COL FSAR, 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 are to be included in the FSAR. The staff finds this exception to the guidance 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,” states that the applicant’s operating organization is characterized as follows:

1. The applicant is technically qualified, as specified in 10 CFR 50.40(b) and 10 CFR 50.80, as applicable.

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) – (m).

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.

7. The applicant has complied with TMI Action Plan items I.A.1.1 and I.A.1.3.

The NRC 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) and 10 CFR 50.80, as applicable. 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.

- LNP COL 9.5-1

The applicant added text to LNP 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 LNP COL FSAR Section 9.5, and that the Vice President Nuclear Operations is responsible for the fire protection program. The applicant added text to LNP COL FSAR Section 13.1.2.1.4.9, “Supervisor - Fire Protection,” describing the responsibilities of the supervisor in charge of the fire protection program.

The NRC staff reviewed LNP COL 9.5-1 relative to the text added to Sections 13.1.1.2.10 and 13.1.2.1.4.9 of the LNP COL application. Based on the management descriptions provided in Sections 13.1.1.2.10 and 13.1.2.1.4.9, the staff finds the applicant’s fire protection organization meets the guidance of NUREG-0800. The technical review for LNP COL 9.5-1 as it relates to the programmatic requirements is addressed in Section 9.5 of this safety evaluation report (SER).

- LNP COL 18.6-1

The NRC staff reviewed LNP COL 18.6-1, which describes the qualifications of the nuclear plant technical support personnel. The technical review for LNP COL 18.6-1 is addressed in Section 18.6 of this SER.

The applicant added text to Section 13.1.1.4, “Qualification of Technical Support Personnel,” stating that 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 also stated that the qualification and experience requirements of headquarters staff are established in corporate nuclear policy and procedure manuals. This section is cross-referenced to LNP COL FSAR, Section 18.6.

The applicant added text to LNP COL FSAR Section 13.1.3, “Qualification Requirements,” stating, in Section 13.1.3.1, the qualifications of managers, supervisors, operators, and technicians of the operating organization will meet the qualification requirements in education and experience for those described in ANSI/ANS-3.1-1993 and RG 1.8. In addition, 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. This section is cross-referenced to LNP COL FSAR, Section 18.6.

The applicant added Table 13.1-201, “Generic Position/Site-Specific Position Cross Reference” and Table 13.1-202, “Minimum On-Duty Operations Shift Organization for Two-Unit Plant.” Table 13.1-201 describes the plant management, technical support, and plant operating organizations and provides a cross-reference to identify the corresponding generic position titles. 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 in the Technical Specifications, administrative procedures, Table 13.1-201, and Table 13.1-202, and are illustrated in Figure 13.1-202.

The NRC staff reviewed the text added to LNP COL FSAR Sections 13.1.1.4, 13.1.3.1, and 13.1.3.2 relative to LNP COL 18.6-1 and concludes that the qualification requirements are acceptable and meet the requirements of 10 CFR 50.40(b) and 10 CFR 50.80, as applicable. This conclusion is based on the following:

The applicant has 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 has described its plans for managing the project and utilizing the NSSS vendor and AE. These plans give adequate 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.

- LNP COL 18.10-1

The NRC staff reviewed LNP COL 18.10-1 included under Section 13.1.1.3.2.4, “Manager – Training LNP.” This section describes the responsibilities of the site training manager relative to the site training programs required for the safe and proper operation and maintenance of the plant. This item is cross-referenced to LNP COL FSAR Section 18.10. The NRC staff concludes that the qualification requirements are acceptable and meet the requirements of 10 CFR 50.40(b) and 10 CFR 50.80, as applicable, 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.

Additional technical review for LNP COL 18.10-1 is in Section 18.10 of this SER.

13.1.5 Post Combined License Activities

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

13.1.6 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 organizational structure of the applicant, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff’s technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The applicant has 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 NSSS vendor and AE. These plans have been reviewed and give 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 50 Appendix B, 10 CFR 52.47, 10 CFR 52.79, and 10 CFR 50.80, as applicable; that is, the applicant is technically qualified to engage in design and construction activities.

The applicant has 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 NRC 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 has 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 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 NRC staff finds that the operating organization proposed by the applicant will conform to these characteristics and will comply with the requirements of 10 CFR 50.40(b) and 10 CFR 50.80, as applicable. That is, the applicant is technically qualified to operate a nuclear power plant; and 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.

In addition, the staff concludes that the information presented in the LNP 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:

- LNP COL 13.1-1, related to the organizational structure of the COL applicant, is acceptable because it meets the requirements of 10 CFR 50.40(b) and 10 CFR 50.80, as applicable.

- LNP COL 9.5-1, related to the fire protection organization meets the guidance of Section 13.1 of NUREG-0800 and is acceptable.

- LNP COL 18.6-1, related to the qualifications of nuclear plant technical support personnel, is acceptable because it meets the requirements of 10 CFR 50.40(b) and 10 CFR 50.80, as applicable.

- LNP COL 18.10-1, related 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) and 10 CFR 50.80, as applicable.

13.2 Training

13.2.1 Introduction

This section addresses the description and schedule of the training program for reactor operators (ROs) and senior reactor operators (SROs), i.e., licensed operators. It 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 LNP COL FSAR includes the description and schedule of the training program for non-licensed plant staff.

13.2.2 Summary of Application

Section 13.2 of the LNP COL FSAR, Revision 9, incorporates by reference Section 13.2 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Section 13.2, the applicant provides 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 (V&V) 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 LNP 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), (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 SER.

- 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 LNP 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 the Commission regulations for the description and schedule of the training program for licensed operators are given in Sections 13.2.1 and 13.2.2 and Chapter 18 of NUREG-0800.

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, “Training and qualification of nuclear power plant personnel”
- 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:

- 10 CFR 50.34(b), “Final safety analysis report”
- 10 CFR 50.54(i)
- 10 CFR 55.59, “Requalification”

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 NRC staff reviewed Section 13.2 of the LNP 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. The NRC 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 NRC staff’s evaluation of the information incorporated by reference in the LNP 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 (VEGP), Units 3 and 4) were equally applicable to the LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 completed its review and found the evaluation performed for the standard content to be directly applicable to the LNP 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 13.2.4 of the VEGP SER:

> *<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.

<u>License Conditions</u>

- *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 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 description and schedule of the training program for licensed operators, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the information presented in the LNP 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 NEI 06-13A, Revision 1, which provides an acceptable template for describing licensed operators and non-licensed plant staff training programs. The staff determined that this is acceptable, as it applies an NRC-endorsed approach.

- STD COL 18.10-1, relating to training, references Section 13.2 of the LNP COL FSAR, in which the applicant has 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 addresses the plans, design features, facilities, functions, and equipment necessary for radiological emergency planning (EP) that must be considered in a COL application. The LNP COL application includes the onsite, 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 the plans can be implemented. The emergency plans are an expression of the overall concept of operation, and describe the essential elements of advanced planning that have been considered and the provisions that have been made to cope with radiological emergency situations.

13.3.2 Summary of Application

Section 13.3 of the LNP COL Final Safety Analysis Report (FSAR), Revision 9, incorporates by reference Section 13.3 of the AP1000 DCD, Revision 19, without any EP related departures. In addition, in LNP COL FSAR Section 13.3, the applicant provided the following:

AP1000 COL Information Items

- 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:

> COL applicants referencing the AP1000 certified design will address emergency planning including post-72 hour actions and its communication interface.

- STD COL 13.3-2

The applicant provided additional information in STD COL 13.3-2 to address COL Information Item 13.3-2 (COL Action Item 13.3.3.3.5-1) of the AP1000 DCD, which states:

> COL applicants referencing the AP1000 certified design will address the activation of the emergency operations facility consistent with current operating practice and NUREG-0654/FEMA-REP-1 ["Criteria for Preparation and Evaluation of Radiological Emergency Plans and Preparedness in Support of Nuclear Power Plants," Revision 1].

Supplemental Information

- STD SUP 13.3-1

The applicant provided additional information in STD SUP 13.3-1 that provides milestones for EP implementation.

Part 5, “Emergency Plan,” Revision 6 of the LNP COL application includes the following:

Onsite Emergency Plans

Part 5, “Emergency Plan,” of the LNP COL application includes the Emergency Plan (the LNP Emergency Plan). The LNP Emergency Plan consists of a basic plan and seven appendices. The seven appendices provide additional information regarding various aspects of the LNP

Emergency Plan (e.g., List of Emergency Plan Supporting Procedures, Evacuation Time Estimate (ETE) Study Summary, and Certification Letters).

Offsite Emergency Plans

Part 5 of the COL application includes current State and local emergency plans. In addition, Part 5 includes the detailed ETE Report.

ITAAC

Part 10, "Proposed License Conditions (Including ITAAC)," Revision 7, of the LNP COL application provides information regarding EP - Inspections, Tests, Analyses, and Acceptance Criteria (EP ITAAC). The EP ITAAC are evaluated in Section 13.3C.19 of this safety evaluation report (SER).

License Conditions

- 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 LNP COL application.

- 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 EP.

- Part 10, License Condition 11

The applicant proposed the following license conditions:

A. Duke Energy Florida (DEF) shall submit a fully developed set of site-specific emergency action levels (EALs) for LNP Units 1 [Unit 2] to the NRC in accordance with Nuclear Energy Institute (NEI) 07-01, revision 0, with no deviations. These 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 at least 180 days prior to initial fuel load.

B. Deleted.

C. Prior to the full-participation exercise to be conducted in accordance with the requirements of Appendix E to 10 CFR Part 50, DEF will have available for NRC inspection the Letters Of Agreement established with the following entities:

    a. Florida Division of Emergency Management
    b. Citrus County, Florida Emergency Management Agency
    c. Levy County, Florida Emergency Management Agency
    d. Marion County, Florida Emergency Management Agency

e. Citrus Memorial Hospital
f. Seven Rivers Regional Medical Center
g. Citrus County, Department of Public Safety Fire Rescue Division
h. Nature Coast Emergency medical Services Fire Department

These Letters of Agreement shall specify the emergency measures to be provided in support of the LNP emergency organization to include response to a hostile action event at the site; the mutually acceptable criteria and availability of adequate resources for their implementation; and arrangements for exchange of information.

D. Prior to the full-participation exercise to be conducted in accordance with the requirements of Appendix E to 10 CFR Part 50, DEF will demonstrate the integrated capability and functionality of the Emergency Operations Facility for simultaneous-dual activation of the Facility by the LNP and Crystal River Unit 3 Emergency Response Organizations for a simulated emergency condition. Integrated communication and data capability and functionality will include the LNP and Crystal River Technical Support Centers, NRC site-teams, NRC Incident Response Centers, and other Federal, State, and local coordination centers as appropriate.

E. DEF shall distribute the initial LNP public information publications, consistent with the LNP Emergency Plan, within 180 days prior to fuel load at LNP. DEF must coordinate the development, initial and annual redistribution, and maintenance of this information with CR3 as long as the NRC requires CR3 to distribute public information publications.

F. At least two (2) years prior to scheduled initial fuel load, DEF 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.

13.3.3 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.

The applicable regulatory requirements and guidance for EP are as follows:

- 10 CFR 52.79(a)(21), “Contents of applications; technical information in final safety analysis report,” and 10 CFR 52.79(a)(22)(i) require that the FSAR include emergency plans that comply with the requirements of 10 CFR 50.47, “Emergency plans,” and Appendix E to 10 CFR Part 50, and certifications from State and local governmental agencies with EP 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, under 10 CFR 50.47(a)(2), the NRC will base its finding on a review of the FEMA findings and determinations as to whether State and local 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 considered the applicable requirements in 10 CFR 52.77, "Contents of applications; general information"; 10 CFR 52.80, "Contents of applications: additional technical information"; 10 CFR 50.33(g), "Content of the application: general information"; and 10 CFR 100.21, "Non-seismic siting criteria."

- NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants" identifies NUREG-0654/FEMA-REP-1, Revision 1 and other related guidance that the staff considered 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-6863, "Development of Evacuation Time Estimate Studies for Nuclear Power Plants," January 2005, and Interim Staff Guidance NSIR/DPR-ISG-01, "Emergency Planning for Nuclear Power Plants," Revision 0, November 2011.

- In addition, Appendix A to 44 CFR 353, "Memorandum of Understanding (MOU) Between Federal Emergency Management Agency and Nuclear Regulatory Commission Relating to Radiological Emergency Planning and Preparedness," September 14, 1993, states that FEMA is responsible for making findings and determinations as to whether offsite emergency plans are adequate and can be implemented. FEMA radiological emergency preparedness (REP) guidance documents provide guidance on various topics for use by State and local organizations responsible for radiological emergency preparedness and response. NUREG-0654/FEMA REP-1 provides guidance to provide a basis for State and local governments to develop radiological emergency plans.

13.3.4 Technical Evaluation

The NRC staff reviewed Section 13.3 of the LNP COL FSAR and checked the referenced DCD to ensure the combination of the DCD and COL application represents the complete scope of information relating to this review topic[3]. The NRC staff's review confirmed that the information in the application and incorporated by reference addresses the required information relating to EP. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the LNP COL FSAR:

AP1000 COL Information Items

- STD COL 13.3-1
- STD COL 13.3-2

[3] 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 NRC staff's evaluation related to STD COL 13.3-1 and 13.3-2 is addressed in Attachment 13.3A of this SER.

Supplemental Information

- STD SUP 13.3-1

The NRC staff's review of STD SUP 13.3-1 is addressed in Attachment 13.3A of this SER.

The NRC staff's review of the information provided in the application that is not part of the LNP Emergency Plan is addressed in Attachment 13.3B, "Additional Required Emergency Planning Information," of this SER. The NRC staff's review of the LNP Emergency Plan is addressed in Attachment 13.3C, "Onsite Emergency Plan," of this SER.

The NRC staff reviewed the application against the generic EP ITAAC provided in Table 14.3.10-1, "Emergency Planning Generic Inspections, Tests, Analyses, & Acceptance Criteria (EP ITAAC)," pursuant to Section 14.3.10 of NUREG-0800.

By letter dated September 26, 2013, from DEF to NRC, DEF requested exemptions for Crystal River 3 (CR3) from specific EP standards of 10 CFR 50.47 and specific requirements of Appendix E to 10 CFR Part 50. The staff evaluated the requested exemption in "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Request for Exemptions from Portions of 10 CFR 50.47 and 10 CFR Part 50, Appendix E, Duke Energy Florida, Inc. et al., Crystal River Unit 3 Nuclear Generating Plant, Docket No. 50-302," and by letter dated March 30, 2015, from NRC to DEF, the NRC approved the exemption. The NRC staff reviewed the LNP Emergency Plan with respect to these exemptions which relate to the use of shared facilities (e.g., the Emergency Operations Facility (EOF)), the conduct of exercises, communications capabilities between the LNP and Crystal River Technical Support Centers (TSCs), and the distribution of public information.

FEMA has reviewed the emergency plans for the State of Florida and the local government plans for Levy, Citrus, and Marion counties pursuant to 44 CFR 350, and provided its Interim Findings Report (IFR) for Reasonable Assurance, dated December 17, 2009, to the NRC in a letter dated February 17, 2010. FEMA has concluded that based on its review of the currently available offsite plans and procedures for the 10-mile plume exposure pathway emergency planning zone (EPZ), as well as the 50-mile ingestion exposure pathway EPZ, the offsite plans are adequate and there is reasonable assurance that the plans can be implemented with no corrections needed. In a letter dated August 20, 2012, NRC provided FEMA with an updated State of Florida Radiological Emergency Preparedness Plan (REPP) revised November 2011. By letter dated October 18, 2012, FEMA provided its response to NRC stating that FEMA has reviewed the updated State of Florida REPP and the February 17, 2010, reasonable assurance finding for off-site emergency planning is still valid. FEMA again re-evaluated the IFR after the NRC granted exemptions to DEF for the CR3 site. By letter dated September 28, 2015, FEMA determined there is no need to revise the findings of the December 17, 2009, IFR for LNP. Specifically, the IFR determined that the offsite plans are adequate and there is reasonable assurance that the plans can be implemented with no corrections needed. The NRC staff has reviewed the FEMA report and based its overall reasonable assurance finding on the FEMA findings and determinations regarding offsite EP.

Based on the staff's evaluation of the applicant's emergency plan found in Attachment 13.3C, the staff finds that the applicant's onsite emergency plan meets the standards in 10 CFR 50.47(b) and the requirements in Appendix E to 10 CFR Part 50, including the requirements of the Final Rule on Enhancements to Emergency Preparedness Regulation effective November 23, 2011.

Based on the IFR and the staff's evaluations detailed in Attachments 13.3A, 13.3B, and 13.3C of this SER, the staff finds that there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. Therefore, the staff finds that the LNP Emergency Plan meets the requirements in 10 CFR 50.33(g), 10 CFR 50.34(b)(6)(v), 10 CFR 50.34(f)(2); 10 CFR 50.47; Appendix E to 10 CFR Part 50; 10 CFR 52.77; 10 CFR 52.79(a)(21); 10 CFR 52.79(a)(22)(i); 10 CFR 52.80; and 10 CFR 52.81, including the requirements of the Final Rule on Enhancements to Emergency Preparedness Regulation effective November 23, 2011.

License Conditions

- Part 10, License Condition 1

The applicant provided a license condition in Part 10 of the LNP COL application, which will incorporate into the COL the ITAAC identified in the tables in Appendix B of Part 10. Appendix B includes the EP ITAAC. The proposed text in License Condition 1 is evaluated in Chapter 1 of this SER. The NRC staff's evaluation of the EP ITAAC identified in Table 3.8-1 of Appendix B to Part 10 of the LNP COL application is documented in Section 13.3C.19 of the SER. Table 13.3-1 of this SER provides the EP ITAAC identified in Table 3.8-1 of Appendix B to Part 10 of the LNP COL application. Therefore, the staff will include the ITAAC in SER Table 13.3-1 in the license.

- Part 10, License Condition 6

The applicant proposed a license condition to provide a schedule which supports the NRC's inspection of operational programs including EP. Specifically, the applicant proposed, in part, 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. This schedule shall also address:
>
> a. the emergency planning implementing procedures to the NRC consistent with 10 CFR Part 50, Appendix E, Section V.

The staff reviewed the above proposed license condition against the recommendations in SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria [ITAAC]" as endorsed by the related Staff Requirements Memorandum (SRM) dated February 22, 2006. The staff concludes that this proposed license condition conforms to the guidance in SECY-05-0197 and is, therefore, acceptable. For additional details on the staff's evaluation of proposed License Condition 6, see Section 13.4.4 of this SER.

- Part 10, License Condition 11

The applicant proposed several license conditions related to the site-specific EALs, finalized LOAs, and the shared EOFs' exercise demonstrating simultaneous activation of the LNP and CR3 EROs. In addition, the applicant proposed License Condition 11(F) for performance of a staffing analysis in response to the Final Rule on Enhancements to Emergency Preparedness Regulation effective November 23, 2011. Specifically, the applicant proposed the following:

A. Duke Energy Florida (DEF) shall submit a fully developed set of site-specific emergency action levels (EALs) for LNP Units 1 [Unit 2] to the NRC in accordance with Nuclear Energy Institute (NEI) 07-01, revision 0, with no deviations. These 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 at least 180 days prior to initial fuel load.

B. Deleted.

C. Prior to the full-participation exercise to be conducted in accordance with the requirements of Appendix E to 10 CFR Part 50, DEF will have available for NRC inspection the Letters Of Agreement established with the following entities:

    a. Florida Division of Emergency Management
    b. Citrus County, Florida Emergency Management Agency
    c. Levy County, Florida Emergency Management Agency
    d. Marion County, Florida Emergency Management Agency
    e. Citrus Memorial Hospital
    f. Seven Rivers Regional Medical Center
    g. Citrus County, Department of Public Safety Fire Rescue Division
    h. Nature Coast Emergency medical Services Fire Department

These Letters of Agreement shall specify the emergency measures to be provided in support of the LNP emergency organization to include response to a hostile action event at the site; the mutually acceptable criteria and availability of adequate resources for their implementation; and arrangements for exchange of information.

D. Prior to the full-participation exercise to be conducted in accordance with the requirements of Appendix E to 10 CFR Part 50, DEF will demonstrate the integrated capability and functionality of the Emergency Operations Facility for simultaneous-dual activation of the Facility by the LNP and Crystal River Unit 3 Emergency Response Organizations for a simulated emergency condition. Integrated communication and data capability and functionality will include the LNP and Crystal River Technical Support Centers, NRC site-teams, NRC Incident Response Centers, and other Federal, State, and local coordination centers as appropriate.

In response to RAI 13.3-48 and subsequent correspondence dated January 10, 2014, the applicant proposed the following addition to License Condition 11:

E. DEF shall distribute the initial LNP public information publications, consistent with the LNP Emergency Plan, within 180 days prior to fuel load at LNP. DEF must coordinate the development, initial and annual redistribution, and maintenance of this information with CR3 as long as the NRC requires CR3 to distribute public information publications.

In response to the Final Rule on Enhancements to Emergency Preparedness Regulation effective November 23, 2011, the applicant proposed the following addition to License Condition 11:

F. At least two (2) years prior to scheduled initial fuel load, DEF 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.

The NRC staff's evaluation of the EALs is documented in Section 13.3C.4 of this SER.

Pursuant to the requirements of 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50, the staff confirmed the language in License Conditions 11(A) and (C) incorporate the requirement for State and local review and agreement of the LNP initial EALs, and for development of finalized LOAs. License Condition 11(B), dealing with LOAs, was subsequently incorporated into License Condition 11 (C) to eliminate redundancy.

The staff finds the modifications to License Conditions 11(A) and 11(C), and the deletion of License Condition 11(B) in Part 10 to the COL application, to be acceptable.

The staff's evaluation of written agreements is documented in Section 13.3C.1.7 of this SER.

The staff's evaluation of the EOF function is documented in Section 13.3C.8.19 of this SER. As described in Section 13.3C.8.19 of this SER, the staff deleted License Condition 11(D), since it was no longer needed as a result of the decommissioning of CR3.

The staff revised License Condition 11(E) to reflect the decommissioning of CR3 as shown below.

> DEF shall distribute the initial LNP public information publications, consistent with the LNP Emergency Plan, within 180 days prior to fuel load at LNP.

The staff's evaluation of public education and information is documented in Section 13.3C.7 of this SER. With the staff's revision to License Condition 11(E) to reflect the decommissioning of CR3, the staff finds License Condition 11(E) to be acceptable since it meets the requirements of 10 CFR 50.47(b)(7) and Appendix E to 10 CFR Part 50.

The staff's evaluation of on-shift and augmented emergency response staff is documented in Section 13.3C.2.7 of this SER. The staff finds proposed License Condition 11(F) to be acceptable with the exception of the reference to the scheduled date for initial fuel load. License Condition (13-7) below is modified to be consistent with the completion of EP ITAAC 2.0.

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) – The licensee shall develop a schedule that supports planning for and conduct of NRC inspections of the operational programs listed in LNP 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 LNP 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 (13-4) – 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 LNP 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.

- 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, DEF shall have available for NRC inspection the LOAs established with the following entities:

    a. State of Florida Division of Emergency Management
    b. Citrus County, Florida Emergency Management Agency
    c. Levy County, Florida Emergency Management Agency
    d. Marion County, Florida Emergency Management Agency
    e. Citrus Memorial Hospital
    f. Seven Rivers Regional Medical Center

g. Citrus County, Department of Public Safety Fire Rescue Division
h. Nature Coast Emergency Medical Services Fire Department

These Letters of Agreement shall specify the emergency measures to be provided in support of the LNP emergency organization to include response to a hostile action event at the site; the mutually acceptable criteria and availability of adequate resources for their implementation; and arrangements for the exchange of information.

- License Condition (13-6) – DEF shall distribute the initial LNP public information publications, consistent with the LNP Emergency Plan, within 180 days prior to fuel load at LNP.

- License Condition (13-7) – 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.

13.3.6 Conclusion

The staff reviewed the application, checked the referenced DCD, and reviewed the safety evaluation for decommissioning CR3. The staff's review confirmed that the applicant addressed the required information relating to EP, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The ITAAC that are applicable to EP for LNP are included in SER Table 13.3-1 and are addressed in Section 13.3C.19. Pursuant to 10 CFR 52.80(a), the LNP COL application includes 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, and the NRC's rules and regulations.

FEMA has reviewed the emergency plans for the State of Florida and the local government plans for Levy, Citrus, and Marion counties pursuant to 44 CFR 350, and provided its IFR for Reasonable Assurance, dated December 17, 2009, to the NRC in a letter dated February 17, 2010. FEMA has concluded that based on its review of the currently available offsite plans and procedures for the 10-mile plume exposure pathway EPZ, as well as the 50-mile ingestion exposure pathway EPZ, the offsite plans are adequate and there is reasonable assurance that the plans can be implemented with no corrections needed. In a letter dated August 20, 2012, NRC provided FEMA with an updated State of Florida REPP revised November 2011. By letter dated October 18, 2012, FEMA provided its response to NRC stating that FEMA has reviewed the updated State of Florida REPP and the February 17, 2010, reasonable assurance finding for off-site emergency planning remains valid. FEMA again re-evaluated the IFR after the NRC granted exemptions to DEF for the CR3 site. By letter dated September 28, 2015, FEMA determined there is no need to revise the findings of the December 17, 2009, IFR for LNP. Specifically, the IFR determined that the offsite plans are adequate and there is reasonable assurance that the plans can be implemented with no corrections needed. The staff has reviewed the FEMA report and based its overall reasonable assurance finding on the FEMA findings and determinations regarding offsite EP.

Based on the staff's evaluation of the applicant's emergency plan for proposed Units 1 and 2 found in Attachment 13.3C, the staff finds that the applicant's onsite emergency plan meets the standards in 10 CFR 50.47(b) and the requirements in Appendix E to 10 CFR Part 50, including the requirements of the Final Rule on Enhancements to Emergency Preparedness Regulation effective November 23, 2011.

Based on the IFR and the staff's evaluations detailed in Attachments 13.3A, 13.3B, and 13.3C of this SER, the staff finds that there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. Therefore, the staff finds that the LNP Emergency Plan meets the requirements in 10 CFR 50.33(g); 10 CFR 50.34(b)(6)(v), 10 CFR 50.34(f)(2); 10 CFR 50.47, Appendix E to 10 CFR Part 50; 10 CFR 52.77; 10 CFR 52.79(a)(21); 10 CFR 52.79(a)(22)(i); 10 CFR 52.80, and 10 CFR 52.81, including the requirements of the Final Rule on Enhancements to Emergency Preparedness Regulation effective November 23, 2011.

Table 13.3-1 Emergency Plan ITAAC			
Planning Standard	**EP Program Elements**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
1.0 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 established, and each principle response organization has staff to respond and to augment its initial response on a continuous basis.	1.1 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]** [**References in brackets throughout this table correspond to with NUREG-0654/FEMA-REP-1 Evaluation Criteria]	1.1 An inspection of the emergency plan implementing procedures will be performed.	1.1 Emergency plan implementing procedures provide for 24-hour per day emergency response staffing and manning of communications links, including continuous operations for a protracted period.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
2.0 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.	2.1 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]	2.1 An inspection of the emergency plan implementing procedures will be performed.	2.1 Emergency plan implementing procedures provide minimum and augmented on-shift staffing levels, consistent with Table B-1 of the Levy Nuclear Plant Units 1 & 2 Combined License (COL) Application Emergency Plan.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
3.0 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.	3.1 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.2]	3.1 An inspection of the Control Rooms, Technical Support Centers (TSCs), and Emergency Operations Facility (EOF) will be performed to verify that they have displays for retrieving facility system and effluent parameters are specified in the Emergency Classification and EAL scheme and the displays are functional.	3.1 The specified parameters are retrievable in the Control Rooms, TSC and EOF, and the ranges of the displays encompass the values specified in the Emergency Classification and EAL scheme.
4.0 Notification Methods and Procedures			
10 CFR 50.47(b)(5) – Procedures have been established for notification, by the licensee, of State and local	4.1 The means exist to notify responsible State and local organizations within 15 minutes after the licensee declares an emergency. [E.2]	4.1 A test will be performed to demonstrate the capabilities for providing initial notification to the offsite authorities after a simulated emergency classification.	4.1 The State of Florida and the counties of Levy, Citrus, and Marion receive notification within 15 minutes after the declaration of an emergency from the control room and the EOF.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
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 Planning Zone have been established.	4.2 The means exist to notify emergency response personnel. [E.1] 4.3 The means exist to notify and provide instructions to the populace within the plume exposure EPZ. [E.3]	4.2 A test of the primary and back-up Emergency Response Organization (ERO) notification systems will be performed. 4.3 The full test of notification capabilities will be conducted.	4.2 The primary and back-up ERO notification system tests result in: • Emergency response personnel receiving the notification message; • Mobilization communication is validated by personnel response to the notification system or by telephone; • Response to electronic notification and plant page system is accomplished during normal working hours, and off hours. 4.3 Notification and clear instructions to the public are successfully accomplished in accordance with the emergency plan requirements.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
5.0 Emergency Communications			
10 CFR 50.47(b)(6) – Provisions exist for prompt communications among principal response organizations to emergency personnel and to the public.	5.1 The means exist for communications among the Control Rooms, TSCs, EOF, principal State and local emergency operations centers (EOCs), and radiological field assessment teams. [F3, F.5]	5.1 A test will be performed of the capabilities. The test for the contact with the principal EOCs and the radiological field assessment teams will be from the Control Room and the EOF. The TSC communication with the Control Room and the EOF will be performed.	5.1 Communications (both primary and secondary methods/systems) are established between the Control Rooms, TSC and the EOF with Florida Division of Emergency Management (DEM) warning point and EOC; Levy County Warning Point and EOC; Citrus County Warning Point and EOC; and Marion County Warning Point and EOC. Communications are established between the Control Rooms, TSC and the EOF with the LNP radiological monitoring teams.
	5.2 The means exist for communications from the Control Rooms, TSCs, and EOF to the NRC headquarters and regional office EOCs (including establishment of the Emergency Response Data System (ERDS) [or its successor system] between the onsite computer system and the NRC Operations Center.) [F.2.6]	5.2 A test is performed of the capabilities to communicate using ENS from each operating Control Room, TSC and EOF to the NRC headquarters and regional office EOCs. The Health Physics Network (HPN) is tested to ensure communications between the TSC and EOF with the NRC Operations Center. ERDS is established [or its successor system] between the onsite computer systems and the NRC Operations Center.	5.2 Communications are established between the Control Rooms, TSC and EOF to the NRC headquarters and regional office EOCs utilizing the ENS. The TSC and EOF demonstrate communications with the NRC Operations Center using HPN. The access port for ERDS [or its successor system] is provided and successfully completes a transfer of data from the plant computer system to the NRC Operations Center.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
6.0 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.	6.1 The licensee has provided space which may be used for a limited number of the news media. [H.1.5]	6.1 A test of the facility/area provides adequate equipment to support Emergency News Center (ENC) operation, including communications with the site and with the Emergency Operation Centers in the state and emergency planning zone (EPZ) counties.	6.1 The ENC includes equipment to support ENC operations, including communications with the EOF and State and EPZ County EOCs.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
7.0 Emergency Facilities and Equipment			
10 CFR 50.47(b)(8) – Adequate emergency facilities and equipment to support the emergency response are provided and maintained.	7.1 The licensee has established a TSC and onsite Operations Support Center (OSC). [The TSC and OSC may be combined at a single location.] [H.1.2, H.1.3, Annexes 1 and 2]	7.1.1 An inspection of the as-built TSCs and OSCs will be performed, including a test of the capabilities. These facilities will meet the criteria of NUREG-0696.	7.1.1 Each TSC has at least 1875 ft^2 of floor space (75 ft^2 per person for a minimum of 25 persons). 7.1.2 The TSC is close to the control room, and the walking distance from the TSC to the control room does not exceed two minutes. 7.1.3 Communications equipment is installed, and voice transmission and reception are accomplished between the Control Rooms, TSC, OSCs, and EOF. 7.1.4 The TSC ventilation systems include a high efficiency particulate air (HEPA), and charcoal filter and radiation monitors are installed. 7.1.5 The TSC receives, stores, processes, and displays plant and environmental information, and enables the initiation of emergency measures and the conduct of emergency assessment. These capabilities are demonstrated during testing and acceptance activities. 7.1.6 There is an OSC located inside the Unit's Protected Area. It is separate from the Control Room and TSC within the Protected Area.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			7.1.7 Communications equipment is installed, and voice transmission and reception are accomplished between the OSC and OSC Teams, the TSC, and Control Rooms.
	7.2 The licensee has established an EOF. [H.1.4]	7.2 An inspection of the as-built EOF will be performed, including a test of the capabilities. The EOF will meet the criteria of NUREG-0696 and 0737.	7.2.1 Communications equipment is installed and voice transmission and reception are accomplished between the Control Rooms, TSC, EOF, radiological monitoring teams (RMTs), NRC, State and county agencies, and ENS. 7.2.2 Radiological data, meteorological data, and plant system data is acquired, displayed and evaluated pertinent to offsite protective measures in the EOF. 7.2.3 The EOF is structurally built in accordance with the Uniform Building Code. 7.2.4 The EOF is environmentally controlled to provide room air temperature, humidity, and cleanliness appropriate for personnel and equipment. 7.2.5 The EOF is provided with industrial security when it is activated to exclude unauthorized personnel and when it is idle to maintain its readiness.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
	7.3 The means exist to initiate emergency measures, consistent with Appendix 1 of NUREG-0654/FEMA-REP-1, Rev. 1. [H.5] 7.4 The means exist to acquire data from, or for emergency access to, offsite monitoring and analysis equipment. [H.6] 7.5 The means exist to provide offsite radiological monitoring equipment in the vicinity of the nuclear facility. [H.7] 7.6 The means exist to provide meteorological information, consistent with Appendix 2 of NUREG-0654/FEMA-REP-1, Rev. 1. [H.8]	7.3 – 7.6 A test will be performed of the capabilities.	7.3 The means exist to initiate emergency measures, consistent with Appendix 1 of NUREG-0654/FEMA-REP-1, Rev. 1. EALs will be classified within 15 minutes or less of initiating condition. 7.4 The means exist to acquire data from, or for emergency access to, offsite monitoring and analysis equipment. EALs using offsite dose monitoring and analysis equipment will be made within 15 minutes of initiating conditions. 7.5 The means exist to provide offsite radiological monitoring equipment in the vicinity of LNP for environmental monitoring including radiological monitoring team dosimetry. 7.6 The means exist to provide meteorological information, consistent with Appendix 2 of NUREG-0654/FEMA-REP-1, Rev. 1. LNP meteorological equipment will be able to assess and monitor actual or potential offsite consequences of a radiological condition related to atmospheric measurements.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
8.0 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.	8.1 The means exist to provide initial and continuing radiological assessment throughout the course of an accident. [I, I.3]	8.1 A test will be performed to demonstrate that the means exist to provide initial and continuing radiological assessment throughout the course of an accident through the plant computer or communications with the Control Room.	8.1 Using selected monitoring parameters, simulated degraded plant conditions are assessed, and protective actions are initiated in accordance with the following 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 drill. B. Radiological Assessment and Control 1. Demonstrate the ability to obtain onsite radiological surveys and samples. 2. Demonstrate the ability to continuously monitor and control radiation exposure to emergency workers.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			3. Demonstrate the ability to activate: a. One radiological monitoring team (2 personnel) within 30 minutes of event declaration and, b. A second radiological monitoring team (2 personnel) within 60 minutes of event declaration. 4. Demonstrate the ability to satisfactorily collect and disseminate field team data. 5. Demonstrate the ability to develop dose projections. 6. Demonstrate the ability to make the decision whether to issue radioprotective drugs (KI) to emergency workers. 7. Demonstrate the ability to develop appropriate protective action recommendations (PARs) and notify appropriate authorities within 15 minutes of development.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
	8.2 The means exist to determine the source term of releases of radioactive material within plant systems, and the magnitude of the release of radioactive materials based on plant system parameters and effluent monitors. [I.3]	8.2 A test will be performed to demonstrate that the means exist to determine the source term of releases of radioactive material within plant systems, and the magnitude of the release of radioactive materials based on plant system parameters and effluent monitors.	8.2 Emergency plan implementing procedures provide sufficient direction to calculate the source terms and the magnitude of the release of postulated accident scenario releases.
	8.3 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]	8.3 A test will be performed to demonstrate that the impact of a radiological release to the environment is able to be assessed by utilizing the relationship between effluent monitor readings, and onsite and offsite exposures and contamination for various meteorological conditions.	8.3 Response personnel can 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 under drill conditions.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
	8.4 The means exist to acquire and evaluate meteorological information. [I.6]	8.4 A test will be performed to acquire and evaluate meteorological data/information.	8.4 The following parameters are displayed in the Control Room, TSC and EOF: • Wind speed (at 10m and 60m) • Wind direction (at 10m and 60m) • Delta-temperature (between 10m and 60m) • Ambient temperature (at 10m and 60m) • Dew point temperature (at 10m) • Precipitation (at 2m) This data is in the format needed for the appropriate emergency plan implementing procedures.
	8.5 The means exist to determine the release rate and projected doses if the instrumentation used for assessment is off-scale or inoperable. [I.4]	8.5 A test will be performed of the capabilities to determine the release rate and projected doses if the instrumentation used for assessment is off-scale or inoperable.	8.5 A drill or exercise is conducted that demonstrates the capability to determine the release rate and projected doses with the instrumentation used for assessment off-scale or inoperable.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
	8.6 The means exist for field monitoring within the plume exposure EPZ. [I.7]	8.6 A test will be performed of the capabilities for field monitoring within the plume exposure EPZ.	8.6 A drill or exercise is conducted that demonstrates the ability of the radiological monitoring teams to be dispatched and locate and monitor a radiological release within the plume exposure EPZ.
	8.7 The means exist to make rapid assessments of actual or potential magnitude and locations of radiological hazards through liquid or gaseous release pathways, including activation, notification means, field team composition, transportation, communication, monitoring equipment, and estimated deployment times. [I]	8.7 A test will be performed of the capabilities to make rapid assessments of actual or potential magnitude and locations of any radiological hazards through liquid or gaseous release pathways, including activation, notification means, field team composition, transportation, communication, monitoring equipment, and estimated deployment times.	8.7 A drill or exercise is conducted that demonstrates the capability to activate the radiological monitoring team(s). The team(s) demonstrates the capability to make rapid assessment of actual or potential magnitude and locations of any radiological hazards through simulated liquid or gaseous release pathways. A qualified radiological monitoring team is capable of being notified, activated, briefed and dispatched from the EOF during a radiological release scenario. The team demonstrates conformance with procedural guidance for team composition, use of monitoring equipment, communication from the field, and locating specific sampling locations.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
	8.8 The capability exists to detect and measure radioiodine concentrations in air in the plume exposure EPZ, as low as 10^{-7} µCi/cc (microcuries per cubic centimeter) under field conditions. [I.7.1]	8.8 A test will be performed of the capabilities to detect and measure radioiodine concentrations in air in the plume exposure EPZ, as low as 10^{-7} µCi/cc (microcuries per cubic centimeter) under field conditions.	8.8 A drill or exercise is conducted that demonstrates the capability of a radiological monitoring team to be dispatched during a radiological release scenario and use sampling and detection equipment for air concentrations in the plume exposure EPZ, as low as 10^{-7} µCi/cc.
	8.9 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.4]	8.9 A test will be performed of the capabilities to estimate integrated dose from the projected and actual dose rates, and for comparing these estimates with the EPA PAGs.	8.9 A drill or exercise is conducted that demonstrates the ability to estimate integrated dose from the dose assessment program and the radiological monitoring team reading during a radioactive release scenario for the following radioisotopes: Kr-88, Ru-106, I-131, I-132, I-133, I-134, I-135, Te-132, Xe-133, Xe-135, Cs-134, Cs-137, Ce-144. Results are compared with the PAGs.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
9.0 Protective Response			
10 CFR 50.47(b)(10) – A range of protective actions has been developed for the plume exposure 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 EPZ appropriate to the locale have been developed.	9.1 The means exist to warn and advise onsite individuals of an emergency, including those in areas controlled by the operator, including:[J.1.1] 1. employees not having emergency assignments; 2. visitors; 3. contractor and construction personnel; and 4. Other persons who may be in the public access areas, on or passing through the site, or within the owner controlled area.	9.1 A test will be performed of the capabilities.	9.1 The following objectives to warn and advise onsite individuals using the plant public address system are successfully satisfied during a drill or exercise: A. Demonstrate the ability to perform assembly and accountability for all onsite individuals, including those identified below, within 30 minutes of an emergency requiring protected area evacuation and accountability: 1. non-essential employees; 2. visitors; 3. contractor and construction personnel. B. Demonstrate the ability to warn and advise other personnel within the owner controlled area in a timely manner (about 15 minutes). C. Demonstrate the ability to perform site dismissal.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
	9.2 The means exist to radiologically monitor people evacuated from the site. [K.4]	9.2 A test will be performed of the capabilities.	9.2 A drill or exercise is conducted that demonstrates the capability to radiologically monitor people evacuated from the site. Equipment is available, and personnel have been assigned and trained to procedures that are approved and in place to accomplish this activity.
	9.3 The means exist to notify and protect all segments of the transient and resident populations. [J.2.1]	9.3 A test will be performed of the capabilities.	9.3 A drill or exercise is conducted to demonstrate the capability of the Public Alert and Notification System to successfully initiate a broadcast message to notify and protect all segments of the transient and resident populations.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
10.0 Radiological Exposure Control			
10 CFR 50.47(b)(11) – Means for controlling radiological exposures, in an emergency, are established for emergency workers. The means for controlling radiological exposures shall include exposure guidelines consistent with EPA Emergency Worker and Lifesaving Activity PAGs.	10.1 The means exist to provide onsite radiation protection. [K.2] 10.2 The means exist to provide 24-hour-per-day capability to determine the doses received by emergency personnel and maintain dose records. [K.3] 10.3 The means exist to decontaminate relocated onsite and emergency personnel, including waste disposal. [K5.b, K.7] 10.4 The means exist to provide onsite and contamination control measures. [K.6]	10.1 An analysis of site procedures will be performed. 10.2 An analysis of emergency plan implementing procedures will be performed. 10.3 An analysis of emergency plan implementing procedures will be performed. 10.4 An analysis of site procedures will be performed.	10.1 Site Procedures provide the means for onsite radiation protection. 10.2 Emergency plan implementing procedures provide the means for 24-hour-per-day capability to determine the doses received by emergency personnel and maintain dose records. 10.3 Emergency plan implementing procedures provide a means to decontaminate relocated onsite and emergency personnel, including waste disposal. 10.4 Site procedures provide the means for onsite contamination control measures.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
11.0 Medical and Public Health Support			
10 CFR 50.47(b)(12) – Arrangements are made for medical services for contaminated, injured individuals.	11.1 Arrangements have been implemented for local and backup hospital and medical services having the capability for evaluation of radiation exposure and uptake. [L.1]	11.1 An analysis of emergency plan implementing procedures will be performed.	11.1 Arrangements have been implemented for local and backup hospital and medical services having the capability for evaluation of radiation exposure and uptake per Letter(s) of Agreement and emergency plan implementing procedures.
	11.2 The means exist for onsite first aid capability. [L.2]	11.2 An analysis of station procedures and emergency plan implementing procedures will be performed.	11.2 The means exist for onsite first aid capability to include a designated first aid station, supplies and site medical response team per station procedures and Emergency plan implementing procedures.
	11.3 Arrangements have been implemented for transporting victims of radiological accidents, including contaminated injured individuals, from the site to offsite medical support facilities. [L.4]	11.3 An analysis of emergency plan implementing procedures will be performed.	11.3 Arrangements have been implemented for transporting victims of radiological accidents, including contaminated injured individuals, from the site to offsite medical support facilities per Letter(s) of Agreement and emergency plan implementing procedures.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
12.0 Exercises and Drills			
10 CFR 50.47(b)(14) – Periodic exercises are (will be) conducted to evaluate major portions of emergency response capabilities, periodic 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.	12.1 Licensee conducts a full participation exercise to evaluate major portions of emergency response capabilities, which includes participation by each State and local agency within the plume exposure EPZ, and each State within the ingestion control EPZ. [N.1]	12.1 A full participation exercise (test) will be conducted within the specified time periods of Appendix E to 10 CFR Part 50.	12.1.1 The exercise is completed within the specified time periods of Appendix E to 10 CFR Part 50, onsite exercise objectives listed below have been met, and there are no uncorrected onsite exercise deficiencies. *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 in accordance with emergency plan implementing procedures. <u>Standard Criteria:</u> a. The appropriate EAL condition associated with a parameter or symptom was recognized. b. The correct emergency classification is declared within 15 minutes of the time that the EAL condition was present.

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, in accordance with emergency plan implementing procedures. <u>Standard Criteria:</u> a. Initiate a plant page announcement using the appropriate message scenario for ERO notification. b. Activate the computer based automated callout system at declaration of an Alert classification or higher. 2. Demonstrate the ability to notify responsible State and local government agencies within 15 minutes and the NRC within 60 minutes after declaring an emergency, in accordance with emergency plan implementing procedures.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			<u>Standard Criteria:</u> a. Transmit information to state and local agencies within 15 minutes of event classification. b. Transmit follow-up information to state and local agencies within 60 minutes of last transmittal. c. Transmit information within 60 minutes of event classification for an initial notification to the NRC. 3. Demonstrate the ability to warn or advise onsite individuals of emergency conditions in a timely manner (about 15 minutes), in accordance with emergency plan implementing procedures. <u>Standard Criteria:</u> a. Initiate notification of onsite individuals of event declaration (via plant page, telephone, etc.)

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			4. Demonstrate the capability of the Public Alert and Notification System to operate properly for public notification when required, in accordance with emergency plan implementing procedures. Standard Criteria: a. Greater than 94% of Alert and Notification System (ANS) sirens are capable of performing their function as indicated by the feedback system. The clarifying notes listed in NEI 99-02, Regulatory Assessment Performance Indicator Guideline, will be used for this test. *C. Emergency Response* 1. Demonstrate the capability to direct and control emergency operations, in accordance with emergency plan implementing procedures.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			<u>Standard Criteria:</u> a. Facility command and control is demonstrated by the Nuclear Shift Manager - Operations in the Control Room (simulator) upon event declaration, and by the Emergency Coordinator - TSC in the Technical Support Center (TSC) and the EOF Director in the Emergency Operations Facility (EOF) within 60 minutes of ERO notification. 2. Demonstrate the ability to transfer overall command and control from the Nuclear Shift Manager - Operations in the Control Room (simulator) to the Emergency Coordinator - TSC in the TSC and EOF Director in the EOF, in accordance with emergency plan implementing procedures.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			<u>Standard Criteria:</u> a. Evaluation of briefings that were conducted prior to turnover includes current plant conditions, radiological release information, response efforts and priorities, and the formal relief of delegable and non- delegable responsibilities. 3. Demonstrate the ability to maintain continuous staffing of the emergency response facilities for a protracted period, in accordance with emergency plan implementing procedures. <u>Standard Criteria:</u> a. Complete shift relief schedule adequate to support 24-hour staffing. 4. Demonstrate the ability to perform assembly and accountability for all onsite individuals within 30 minutes of an emergency requiring a Protected Area evacuation and accountability, in accordance with emergency plan implementing procedures.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			<u>Standard Criteria:</u> a. All Protected Area personnel are assembled in their designated assembly area and accountability is completed within 30 minutes of an emergency requiring Protected Area evacuation and accountability. *D. Emergency Response Facilities* 1. Demonstrate activation of the Operations Support Center (OSC), Technical Support Center (TSC), Emergency Operations Facility (EOF), and Emergency News Center (ENC), in accordance with emergency plan implementing procedures. <u>Standard Criteria:</u> a. The TSC and OSC, are activated within approximately one (1) hour of an Alert or higher emergency declaration with at least minimum staffing. b. The EOF is activated within approximately one (1) hour of a Site Area Emergency or higher emergency declaration with at least minimum staffing. c. The ENC minimum staffing positions are available within approximately two (2) hours of a Site Area Emergency or higher emergency declaration.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			2. Demonstrate the adequacy of equipment, security provisions, and habitability precautions for the TSC, OSC, EOF, and ENC, as appropriate, in accordance with emergency plan implementing procedures. <u>Standard Criteria</u> a. The adequacy of the emergency equipment in the emergency response facilities, including availability and consistency with emergency plan implementing procedures, supported the accomplishment of all of the evaluated performance objectives. b. The Security Coordinator implements and performs all appropriate steps from the emergency plan implementing procedures for the ingress, egress, and control of onsite and offsite personnel responding to the site during the scenario. c. The Radiation Controls Coordinator and staff correctly implement and perform all appropriate steps from the emergency plan implementing procedures when a simulated onsite/offsite release has occurred during the scenario. d. Demonstrate the capability of TSC and EOF equipment and data displays to clearly identify and reflect the affected unit.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			3. Demonstrate communications from the emergency response facilities and the adequacy of communications for all emergency support resources, in accordance with emergency plan implementing procedures. <u>Standard Criteria:</u> a. Emergency response communications are available and operational. b. Communications systems are adequate to support CR, TSC, OSC, EOF, and ENC activation. c. Demonstrate emergency response personnel are able to operate all specified communication systems. d. Clear primary and backup communications links are established and maintained for the duration of the exercise. *E. Radiological Assessment and Control* 1. Demonstrate the ability to obtain onsite radiological surveys and samples.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			Standard Criteria: a. Radiation Protection (RP) personnel demonstrate the ability to obtain appropriate instruments (range and type) and take surveys for scenario conditions that allow EPA PAGs to be exceeded. b. Airborne samples are properly taken, reported and assessed and utilized when the conditions indicate the need for the information. 2. Demonstrate the capability to establish emergency exposure guidelines consistent with EPA-400 and the ability to continuously monitor and control radiation exposure to emergency workers. Standard Criteria: a. Demonstrate the ability to determine doses received by emergency personnel and volunteers 24 hours/day and provisions for distribution of both self-reading and permanent record devices.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			b. Demonstrate that exposures are controlled to 10 CFR Part 20 limits until the Emergency Coordinator authorizes the use of emergency EPA limits. c. Exposure records are available, either from the ALARA computer or a hard copy dose report, and are updated and reviewed throughout the scenario. 3. Demonstrate the methods, equipment, and expertise available to make rapid assessments of the actual or potential magnitude and locations of radiological hazards from both gaseous and liquid pathways. <u>Standard Criteria:</u> a. One radiological monitoring team (2 personnel) is ready to be deployed no later than 30 minutes from the declaration of an Alert or higher emergency. b. A second radiological monitoring team (2 personnel) is ready to be deployed no later than 60 minutes from the declaration of an Alert or higher emergency.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			4. Demonstrate the ability to satisfactorily collect and disseminate radiological monitoring team data. <u>Standard Criteria:</u> a. Offsite radiological environmental data collected is provided as dose rate and counts per minute (cpm) from the plume, both open and closed window, and air sample (gross and net cpm) for particulate and iodine, if applicable. b. Offsite radiological environmental data is communicated from the radiological monitoring team to the Radiation Control Coordinator. 5. Demonstrate the ability to estimate integrated dose from projected and actual dose rates and to compare these estimates with EPA Protective Action Guidelines (PAGs).

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			Standard Criteria: a. The Dose Projection Team Leader and Dose Projection Team perform dose projections in accordance with emergency plan implementing procedures, and report them to the Radiation Controls Manager. 6. Demonstrate the availability and use of potassium iodide (KI) for onsite emergency response personnel. Standard Criteria: a. KI is considered as a potential dose reducing option for situations where airborne radioactive iodine is present. b. KI was administered for activities where personnel dose to the thyroid was calculated, or estimated, to be > 25 Rem committed dose equivalent (CDE).

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			7. Demonstrate the ability to recommend protective actions to appropriate offsite authorities, in accordance with emergency plan implementing procedures. <u>Standard Criteria:</u> a. Total effective dose equivalent (TEDE) and committed dose equivalent (CDE) to the thyroid dose projections from the dose assessment model are compared to the PAGs. b. PARs are developed within 15 minutes of the time information of the condition warranting a PAR was available to the ERO. c. PARs are transmitted within 15 minutes of development. Changes to recommendations are communicated to offsite authorities within 15 minutes of a new PAR.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			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. <u>Standard Criteria:</u> a. Information provided to the media/public is prepared at a level that the public can understand. Visuals and handouts are provided as needed to clarify the information. b. Information is coordinated with Federal, State and local agencies to maintain factual consistency. c. Media briefings are provided within approximately 60 minutes of significant events (i.e., declaration of a Site Area Emergency or initiation of a radiological release.)

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			2. Demonstrate the capability to establish and effectively operate rumor control in a coordinated fashion, in accordance with emergency plan implementing procedures. <u>Standard Criteria:</u> a. Calls are answered in a timely manner with the correct information. b. Calls are returned or forwarded, as appropriate, to demonstrate responsiveness. c. Rumors are identified and addressed, and recurring rumors are addressed in subsequent press briefings and news releases.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			G. Recovery and Reentry 1. Demonstrate the ability to enter recovery and reentry conditions, in accordance with emergency plan implementing procedures. <u>Standard Criteria:</u> a. The appropriate EAL condition and emergency classification is downgraded to a lower classification or terminated. b. Proper notifications are made to onsite and offsite emergency response agencies, including State and local agencies. *H. Evaluation* 1. Demonstrate the ability to conduct a post- exercise critique, to determine areas requiring improvement and corrective action, in accordance with emergency plan implementing procedures.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			<u>Standard Criteria:</u> a. An exercise time line is developed, followed by an evaluation of the objectives against the expectations of the timeline. b. Significant problems in achieving the objectives are discussed to ensure understanding of why objectives were not fully achieved. c. Areas requiring improvement are entered in the Levy Corrective Action Program. 12.1.2 Onsite emergency response personnel are mobilized in sufficient numbers to fill emergency response positions and successfully perform assigned responsibilities (see Note 1).

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
			12.1.3 The exercise was completed within the specified time periods of Appendix E to 10 CFR Part 50, offsite exercise objectives were met, and there were no uncorrected offsite exercise deficiencies, or a license condition requires offsite deficiencies to be corrected prior to operation above 5% of rated power as described in 10 CFR 50.54(gg). (Note 1: The assigned responsibilities for onsite Emergency Response Organization members are identified in Sections B.1 through B.7 of the Levy COL Application Emergency Plan and Emergency Plan Implementing Procedures.)

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
13.0 Radiological Emergency Response Training			
10 CFR 50.47(b)(15) – Radiological emergency response training is provided to those who may be called on to assist in an emergency.	13.1 Site-specific emergency response training has been provided for those who may be called upon to provide assistance in the event of an emergency. [O.1]	13.1 An inspection of the emergency response organization training program will be performed.	13.1 Site-specific emergency response training has been provided for the: • LNP emergency response organization, and • Offsite medical, local law enforcement and firefighter personnel that may be called upon to provide assistance in the event of an emergency as documented on training records.

Planning Standard	EP Program Elements	Inspections, Tests, Analyses	Acceptance Criteria
14.0 Responsibility for the Planning Effort: Development, Periodic Review, and Distribution of Emergency Plans			
10 CFR 50.47(b)(16) – Responsibilities for plan development and review and for distribution of emergency plans are established, and planners are properly trained.	14.1 The emergency response plans have been forwarded to all organizations and appropriate individuals with responsibility for implementation of the plans. [P.5]	14.1 An inspection of the distribution list will be performed.	14.1 The LNP emergency response plan was forwarded to Florida Emergency Management, Citrus County Emergency Management, Levy County Emergency Management and Marion County Emergency Management.
15.0 Implementing Procedures			
10 CFR Part 50, App. E.V – No less than 180 days prior to the scheduled issuance of an operating license for a nuclear power reactor or a license to possess nuclear material, the applicant's detailed implementing procedures for its emergency plan shall be submitted to the Commission.	15.1 The licensee has submitted detailed implementing procedures for its emergency plan no less than 180 days prior to fuel load.	15.1 An inspection of the submittal letter will be performed.	15.1 The date of the submittal letter from the licensee demonstrates that the detailed implementing procedures for the onsite emergency plan were submitted no less than 180 days prior to fuel load.

ATTACHMENT 13.3A - COL INFORMATION ITEMS, SUPPLEMENTAL INFORMATION ITEMS AND DEPARTURES

Introduction

This section addresses the COL information and supplemental information items associated with EP.

Section 13.3 of the COL application does not include any EP related departures from the AP1000 certified design for the LNP site that must be addressed by the COL applicant.

13.3A.1 Regulatory Basis

The applicable regulatory requirements for STD COL 13.3-1 and STD COL 13.3-2 associated with EP are established in 10 CFR 50.33(g), 10 CFR 52.79(a)(17), 10 CFR 52.79(a)(21), 10 CFR 50.34(f)(2)(xxv), 10 CFR 50.47(b)(6) and (8), and the guidance is provided in NUREG-0654/FEMA-REP-1, Revision 1 and Supplement 1 to NUREG-0737, "Clarification of TMI Action Plan Requirements."

With respect to STD SUP 13.3-1, the guidance related to implementation milestones for the EP program is provided in the Sample FSAR Table 13.4-X, "Operational Programs Required by NRC Regulation and Program Implementation," in NUREG-0800.

13.3A.2 COL Information Items

Technical Information in the Application:

- STD COL 13.3-1

Section 13.3, "Emergency Planning," of the LNP COL FSAR states:

> The emergency planning information is submitted to the Nuclear Regulatory Commission as a separate licensing document and is incorporated by reference (see Table 1.6-201).
>
> 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 direct current (dc) equipment rooms are established in operating procedures.

In the request for additional information (RAI) 13.3-26(A), the staff requested the applicant explain why STD COL 13.3-1 did not address communication interfaces as described in

NUREG-1793, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design.” In response, the applicant stated that the LNP Emergency Plan addresses communication interfaces primarily in Sections E and F, which provide discussion of emergency notification methods and various communication systems, including their locations, reliability, and periodic testing.

- STD COL 13.3-2

Section 13.3 of the LNP COL FSAR, STD COL 13.3-2 states:

> The emergency plan describes the plans for coping with emergency situations, including communication interfaces and staffing of the emergency operations facility.

In RAI 13.3-26(B), the staff requested the applicant explain why COL Action Item 13.3.3.3.5-1 in Appendix F of NUREG-1793 addresses activation of the EOF, while the corresponding action item in the LNP COL FSAR, STD COL 13.3-2, addresses staffing and communication interfaces of the EOF, and does not address activation of the facility. The applicant’s response stated that 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 emergency response facility (ERF), establishing the required communications interfaces, and declaring the facility to be operational. The applicant provided references to the LNP Emergency Plan that address these activities and stated that this information will be included in the emergency plan implementing procedures (EPIP).

Technical Evaluation

- 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 communications interface. The applicant addressed STD COL 13.3-1 by listing the LNP Emergency Plan for Units 1 and 2 in FSAR Table 1.6-201, “Additional Material Referenced,” with a reference to FSAR Section 13.3, “Emergency Plan,” including submittal of its Emergency Plan in Part 5, “Emergency Plan,” of the COL application. The staff finds the applicant’s submittal of the onsite emergency plan for LNP in Part 5 of the COL application acceptable because it meets the requirements of Appendix E to 10 CFR Part 50 and 10 CFR 52.79(a)(21). As described above, the applicant provided additional information in response to RAI 13.3-26(A) that adequately addresses communications interfaces, including interfaces among the control rooms (CRs), TSCs, EOFs, other ERFs (e.g., State and local emergency operation centers [EOCs], and the NRC) to support the LNP site in an emergency. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1, Revision 1. Additionally, the staff’s evaluation of communications interfaces is addressed in Sections 13.3C.5, “Notification Methods and Procedures,” and 13.3C.6, “Emergency Communications,” of this SER.

In regard to post 72-hour actions associated with the AP1000 DCD, the applicant referenced operating procedures and various related DCD sections. The staff identified additional AP1000 DCD Tier 2 sections that address post-72 hour support actions, which include DCD Sections 6.4, “Habitability Systems,” 9.4, “Air-Conditioning, Heating, Cooling, and Ventilation System,” and 9.5, “Other Auxiliary Systems” (e.g., plant lighting systems described in Section 9.5.3). As described in AP1000 DCD Section 1.9.5.4, post-72 hour support actions relate to an extended loss of the nonsafety-related systems for both onsite and offsite alternating current (ac) power sources for more than 72 hours. For purposes of the staff’s review of EP information in the COL application and in the context of COL Information Item 13.3-1, the reference to post-72 hour support actions is limited to the reliability of the electrical power supply for the TSC ventilation system and associated communications equipment. The evaluation of the reliability of the electrical power supply for the TSC is addressed in the AP1000 DCD sections referenced above. The habitability and functionality of the TSC is addressed above in Section 13.3C.8, “Emergency Facilities and Equipment,” of this SER.

The staff finds that the applicant has addressed EP, including communication interfaces (see STD COL 13.3-2, below), in support of LNP Units 1 and 2 in the Emergency Plan. In addition, the applicant has addressed post-72 hour actions through reference to the AP1000 DCD sections (identified above) that specifically address an extended loss of the nonsafety-related systems for both onsite and offsite 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. Operating procedures to address post-72 hour support actions are being tracked by STD COL 13.5-1 in Section 13.5, “Plant Procedures,” of this SER. In consideration of the applicant’s response to RAI 13.3-26(A), the staff finds that the COL applicant has adequately addressed STD COL 13.3-1.

- STD COL 13.3-2

STD COL Information Item 13.3-2 requires that COL applicants referencing the AP1000 certified design will address activation of the EOF consistent with current operating practice and NUREG-0654/FEMA-REP-1. In 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 communications interfaces and staffing of the EOF.

In response to RAI 13.3-26(B), the applicant provided reference to various sections of the LNP Emergency Plan that outline the overall roles and responsibilities of the Emergency Coordinator (EC) and EOF Director when the EOF is declared operational. In addition, these references describe the location and size of the EOF, functions to be performed by the facility, and capabilities specific to communications and data display. The applicant proposed to include this information in an EPIP. However, this information is inconsistent with the guidance provided in Supplement 1 to NUREG-0737 for activation of the EOF.

In RAI 13.3-21(B), discussed in Section 13.3C.8 of this SER, the staff requested the applicant provide a discussion in the LNP Emergency Plan regarding the timely activation of ERFs. The applicant's response, in part, stated that the applicant will staff the EOF, under the discretion of the EC, at the declaration of a Notification of Unusual Event or Alert emergency classification. Staffing of the EOF will be required at the declaration of a Site Area or General Emergency classifications. The applicant provided a discussion regarding response time goals for minimum staffing of the EOF. Specifically, the applicant stated that a goal of 60 minutes has been established for minimum staffing of the EOF, and it is the goal of the organization to be capable of declaring the EOF operational within 15 minutes.

The information provided in response to RAI 13.3-21(B) provides sufficient detail regarding EOF activation, consistent with operating practice. The staff finds the applicant's response to RAI 13.3-21(B) to be acceptable because it conforms to the guidance in Supplement 1 to NUREG-0737. The staff confirmed that the information provided in response to RAI 13.3-21(B) is incorporated into the LNP Emergency Plan.

The staff finds that the applicant's onsite emergency plan in Part 5 of the COL application adequately addresses activation of the EOF and communication interfaces between the ERFs and the CR as described in the staff's evaluation of STD COL Information Item 13.3-1 above. Therefore, the staff finds the information in the LNP Emergency Plan associated with STD COL 13.3-2 and the response to RAI 13.3-21(B) acceptable because it meets the guidance in NUREG-0737, Revision 1, and applicable requirements of 10 CFR 50.47(b)(6) and (8).

13.3A.3 Supplemental Information Items

Technical Information in the Application:

- STD SUP 13.3-1

Section 13.3 of the LNP FSAR, STD SUP 13.3-1 states:

> Table 13.4-201 provides milestones for emergency planning implementation.

Technical Evaluation

- STD SUP 13.3-1

The applicant provided acceptable milestones for EP program implementation in Table 13.4-201, "Operational Programs Required by NRC Regulations," of the LNP COL FSAR consistent with the requirements in Appendix E to 10 CFR Part 50 and the acceptance criteria in NUREG-0800. The staff's evaluation of EP milestones to support issuance of 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," licenses is in Section 1.5 of this SER.

13.3A.4 Post Combined License Activities

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

13.3A.5 Conclusion

The staff reviewed the LNP COL application, referenced AP1000 DCD, and the applicant's response to RAIs. The staff's review confirmed that the applicant addressed the required information relating to EP, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The staff has compared the COL information and supplemental information items in the LNP COL application to the applicable NRC requirements, acceptance criteria defined in Section 13.3 of NUREG-0800, and other NRC regulatory guidance. The staff concludes that the applicant is in compliance with the applicable regulatory requirements in 10 CFR 50.33(g), 10 CFR 52.79(a)(17), 10 CFR 52.79(a)(21), 10 CFR 50.34(f)(2)(xxv), 10 CFR 50.47(b)(6) and (8), and the guidance provided in NUREG-0654/FEMA-REP-1, Revision 1, Supplement 1 to NUREG-0737, and NUREG-0800.

ATTACHMENT 13.3B – ADDITIONAL REQUIRED EMERGENCY PLANNING INFORMATION

Introduction

This section of the SER includes the staff's evaluation of EP information that is required to be provided in the COL application, but does not address the applicant's plans for responding to a radiological emergency, which are evaluated in Attachment 13.3C in this SER.

13.3B.1 Regulatory Basis

The applicable regulatory requirements for EP information are as follows:

- 10 CFR Part 50, Appendix E, Section I, "Introduction," describes the EPZ.

- 10 CFR Part 50, Appendix E, Section E.III, "The Final Safety Analysis Report," requires that the FSAR include plans for coping with emergencies.

- 10 CFR 52.79(a)(21) and 10 CFR 50.34(b)(6)(v), also require that the FSAR include an onsite emergency plan that meets the requirements in 10 CFR 50.47 and 10 CFR Part 50, Appendix E.

- 10 CFR 50.33 and 10 CFR 52.77, require in part, the submittal of State and local emergency plans.

- 10 CFR 50.33(g) requires, in part, a description of the plume exposure pathway and the ingestion pathway EPZs. In addition, 10 CFR 50.47(c)(2) states generally, the plume exposure pathway EPZ for nuclear power plants shall consist of an area about 10 miles (16 kilometers [km]) in radius and the ingestion exposure pathway EPZ shall consist of an area about 50 miles (80 km) in radius. The exact size and configuration of the EPZs surrounding a particular nuclear power reactor shall 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. The plans for the ingestion pathway shall focus on such actions as are appropriate to protect the food ingestion pathway.

- 10 CFR 50.34(b)(6)(v) requires plans for coping with emergencies, which shall include the items specified in Appendix E. 10 CFR 50.34(h)(1)(i) and 10 CFR 52.79(a)(41) require that the COL application include an evaluation of the facility against NUREG-0800. Section 13.3 of NUREG-0800 provides guidance for the review of onsite emergency plans for nuclear power plants. 10 CFR 50.34(h)(2) and (3) require that the evaluation identify and describe all differences from the NUREG-0800 acceptance criteria in Section 13.3 and evaluate how the proposed alternatives to the NUREG-0800 criteria provide an acceptable method of complying with the Commission's regulations. Where differences exist, the evaluation should discuss how the proposed alternative provides an acceptable method of complying with the Commission's regulations or portions thereof that underlie the corresponding NUREG-0800 acceptance criteria.

- 10 CFR 52.73, "Relationship to other subparts," states that the application for a COL may reference a standard design.

- 10 CFR 52.79(a)(22)(i) requires certifications from State and local governmental agencies with EP responsibilities 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.

- 10 CFR 52.81 states that COL applications will be reviewed according to the standards in 10 CFR Part 50 and 10 CFR Part 100, "Reactor site criteria." Therefore, the requirements of 10 CFR Part 100, Subpart B, "Evaluation Factors for Stationary Power Reactor Site Applications on or after January 10, 1997," are applicable.
  10 CFR 100.1(c), "Reactor site criteria, purpose," requires the identification of physical characteristics unique to the proposed site that could pose a significant impediment to the development of emergency plans. In addition, 10 CFR 100.21(g) also requires that applications for site approval identify physical characteristics unique to the proposed site.

- 10 CFR 100.1(c) states siting factors and criteria are important in assuring that radiological doses from normal operation and postulated accidents will be acceptably low, that natural phenomena and potential man-made hazards will be appropriately accounted for in the design of the plant, that site characteristics are such that adequate security measures to protect the plant can be developed, and that physical characteristics unique to the proposed site that could pose a significant impediment to the development of emergency plans are identified.

- 10 CFR 100.21(g) states physical characteristics unique to the proposed site that could pose a significant impediment to the development of emergency plans must be identified.

13.3B.2 FSAR and Onsite Emergency Plan

Technical Information in the Application: {Appendix E, Section III} (10 CFR 52.79(a)(21)) (10 CFR 50.34(b)(6)(v))
Section 13.3 of the LNP COL FSAR states, in STD COL 13.3-1, that EP information is submitted to the NRC as a separate licensing document and is incorporated by reference (see Table 1.6-201). The document is Part 5, "Emergency Plan," (LNP Emergency Plan) of the COL application. Section 1.0, "Introduction," of the LNP Emergency Plan states that the emergency plan is developed in compliance with the requirements of 10 CFR Part 52. The requirements in 10 CFR Part 52 invoke the EP requirements in 10 CFR Part 50. Consistent with the requirements of both 10 CFR Part 50 and 10 CFR Part 52, the emergency plan is based on the requirements of 10 CFR 50.47 and Appendix E to 10 CFR Part 50. In addition, the applicant states that the emergency plan is consistent with the guidance provided in NUREG-0654/FEMA-REP-1, Revision 1.

The LNP Emergency Plan consists of a basic plan and seven appendices. The seven appendices provide additional information regarding various aspects of the LNP Emergency Plan (e.g., List of Emergency Plan Supporting Procedures, ETE Study Summary, and Certification Letters).

Technical Evaluation: {Appendix E, Section III} (10 CFR 52.79(a)(21)) (10 CFR 50.34(b)(6)(v))
The staff finds that the LNP COL FSAR includes an emergency plan for coping with emergencies at the LNP site, which meets the applicable requirements in Section III of Appendix E to 10 CFR Part 50, 10 CFR 52.79(a)(21), and 10 CFR 50.34(b)(6)(v).

13.3B.3 Submittal of State and Local Emergency Plans

Technical Information in the application: (10 CFR 50.33(g))
Section 1.3.1, "Plume Exposure Pathway EPZ," of the LNP Emergency Plan states that the State of Florida and respective counties within the 10-mile EPZ have prepared plans for a

response to an emergency at LNP. The plans that describe the State and local EP documents are included as supplemental information. The list of State and local EP documents includes:

- State of Florida Radiological Emergency Management Plan
- Citrus County Sheriff's Office Radiological Emergency Preparedness Plan
- Levy County Emergency Management Radiological Emergency Preparedness Plan
- Marion County Emergency Management Radiological Emergency Preparedness Plan

Technical Evaluation: (10 CFR 50.33(g))
The applicant submitted offsite emergency plans for the State of Florida and Levy, Citrus, and Marion counties, which are wholly or partially within the plume exposure pathway EPZ. This is acceptable because it meets the requirements in 10 CFR 50.33(g).

13.3B.4 Description of Emergency Planning Zones

Technical Information in the Application: {Appendix E, Section I} (10 CFR 50.33(g)) (10 CFR 50.47(c)(2))
Section 1.3, "Emergency Planning Zones," of the LNP Emergency Plan defines the plume exposure and ingestion exposure pathway EPZs as follows:

The plume exposure pathway EPZ consists of an area within an approximate 10-mile radius of the LNP. Figure Intro-3, "Plume Exposure Pathway EPZ (10-Mile)," provides an illustration of the 10-mile plume exposure pathway EPZ for the LNP site. Section 1.3.1 further describes the plume exposure pathway EPZ as the area in which principal exposure sources from the plume exposure pathway consist of external exposure to gamma and beta radiation from the plume and deposited materials, and exposure of internal organs to gamma and beta radiation from inhaled radioactive gases or particulates.

Section 1.3.2, "Ingestion Exposure Pathway EPZ," states that the ingestion exposure pathway EPZ consists of an area within an approximate 50-mile radius of the LNP. Figure Intro-4, "Ingestion Exposure Pathway EPZ (50-Mile)," provides an illustration of the ingestion exposure pathway EPZ, which includes the Florida counties of Alachua, Citrus, Dixie, Gilchrist, Hernando, Lake, Levy, Marion, Pasco, Putnam, and Sumter. The ingestion exposure pathway EPZ is described as the area in which the exposure sources are from contaminated water or food, such as milk or fresh vegetables.

In RAI 13.3-27, the staff asked the applicant to discuss in the LNP Emergency Plan whether the exact sizes and configurations of the EPZs surrounding the LNP site were determined in relation to the local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries. The applicant's response stated that the plume exposure pathway and ingestion exposure pathway EPZs for the LNP site were determined in accordance with criteria described in NUREG-0654/FEMA-REP-1, Part 1, Section D.1.a, D.1.b, and Section D.2. The applicant stated that the exact size and configuration of the EPZs were discussed and coordinated with representatives from the State of Florida Division of Emergency Management and Levy, Citrus, and Marion County emergency management directors from the 10-mile EPZ risk counties. In addition, the applicant stated that demographical data, topographical information, land characteristics, access routes and jurisdictional boundaries were all taken into consideration in the determination of the 10-mile and 50-mile EPZ boundaries.

Technical Evaluation: {Appendix E, Section I} (10 CFR 50.33(g)) (10 CFR 50.47(c)(2))

The staff finds the applicant's response to RAI 13.3-27 to be acceptable because it conforms to the guidance in NUREG-0396/EPA520/1-78-016, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," and NUREG-0800. The staff confirmed that information provided by the applicant was incorporated into the LNP Emergency Plan.

The onsite emergency plan describes the plume exposure pathway EPZ as consisting of an area about 10 miles in radius and the ingestion exposure pathway EPZ consisting of an area about 50 miles in radius. The exact size and configuration of the EPZs were determined in relation to the local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries. The description of the EPZs provided by the applicant conforms to the guidance in NUREG-0396/EPA520/1-78-016, NUREG-0654/FEMA-REP-1, Revision 1, and meet the acceptance criteria in NUREG-0800.

Based on the information in the LNP Emergency Plan and the applicant's response to RAI 13.3-27, the staff finds that the EPZ sizes are acceptable and meet the requirements in 10 CFR 50.33(g), 10 CFR 50.47(c)(2), and Section 1 of Appendix E to 10 CFR Part 50.

13.3B.5 Certifications from State and Local Governments

Technical Information in the Application: (10 CFR 52.79(a)(22)(i))
Appendix 3, "Certification Letters," of the LNP Emergency Plan includes certification letters between Progress Energy Florida (PEF, currently Duke Energy Florida (DEF)), and State and local governmental agencies with EP responsibilities. These agencies include:

- Citrus County Emergency Management
- Levy County Emergency Management
- Marion County Emergency Management
- State of Florida Division of Emergency Management

Technical Evaluation: (10 CFR 52.79(a)(22)(i))
The applicant provided certification letters from the State and local governmental agencies with EP responsibilities, which 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. This is acceptable because it meets the requirements of 10 CFR 52.79(a)(22)(i).

13.3B.6 Evaluation Against the Standard Review Plan

Technical Information in the Application: (10 CFR 52.79(a)(41)) (10 CFR 50.34(h)(1)(i)) (10 CFR 50.34(h)(2) and (3))
LNP COL FSAR Table 1.9-202, "Conformance with SRP Acceptance Criteria," in STD SUP 1.9-1 indicates conformance with the acceptance criteria in NUREG-0800, which is acceptable for Section 13.3 with no differences identified.

Technical Evaluation: (10 CFR 52.79(a)(41)) (10 CFR 50.34(h)(1)(i)) (10 CFR 50.34(h)(2) and (3))
The applicant provided the results of its evaluation of the facility against the acceptance criteria in NUREG-0800. The staff finds the applicant addressed the applicable requirements as referenced above for Section 13.3 with no differences identified.

13.3B.7 Reference to a Standard Design

Technical Information in the Application: (10 CFR 52.73)
Section 13.3 of the LNP COL FSAR states that the AP1000 DCD is incorporated by reference with supplements and no departures.

Technical Evaluation: (10 CFR 52.73)
There are no EP-related departures from the AP1000 DCD. The staff finds that the AP1000 DCD was incorporated by reference in the LNP COL FSAR and the evaluation of the supplements is addressed in Attachment 13.3A of this SER. This is acceptable because it meets the requirements of 10 CFR 52.73.

13.3B.8 Impediments to the Development of Emergency Plans

Technical Information in the Application: (10 CFR 52.81) (10 CFR 100.1(c)) (10 CFR 100.21(g))
Appendix 6, "Evacuation Time Estimate Study Summary," of the LNP Emergency Plan states that the ETE Report, "Levy Nuclear Plant, Development of Evacuation Time Estimates," dated August 2009, describes the analyses undertaken and the results obtained by a study to develop ETEs for the proposed LNP. Section 1.3, "Preliminary Activities," of the ETE Report states, in part, that the entire highway system within the EPZ, and for some distance outside of the EPZ, was driven while characteristics of each section of the highway were recorded. These characteristics include unusual characteristics such as narrow bridges, sharp curves, poor pavement, flood warning signs, and inadequate delineations. This information was referenced while preparing the input stream for the traffic simulation modeling software system.

In RAI 13.3-3(G), the staff asked the applicant to explain the significance of the unusual characteristics of the highway system identified within the EPZ, and for some distance outside of the EPZ, and how they impact the proposed LNP site. In addition, the staff requested the applicant address whether any unusual characteristics unique to the proposed LNP site could pose a significant impediment to the development of the LNP Emergency Plan. The applicant's response references its response to RAI 13.3-11(A) through RAI 13.3-11(C) as including a detailed discussion of the road survey performed. In addition, the applicant's response to RAI 13.3-11(B)(1) states that the number of bridges, sharp curves, narrow shoulders, and other capacity-reducing features on the evacuation network were observed and considered in estimating capacity. These features are identified in Appendix K to the ETE Report.

In supplemental RAI 13.3-33, the staff asked the applicant to clarify in the ETE analysis whether any physical characteristics unique to the proposed LNP site exist that could pose a significant impediment to the development of the LNP Emergency Plan. The applicant's response, in part, stated that the April 2008 and August 2009 ETE Reports were discussed by KLD Associates, Progress Energy, and Emergency Management personnel from the State of Florida and local counties of Citrus, Levy, and Marion, and that there were no physical characteristics unique to the proposed LNP site identified that could pose a significant impediment to protecting the public.

Technical Evaluation: (10 CFR 52.81) (10 CFR 100.1(c)) (10 CFR 100.21(g))
The staff finds the applicant's response to supplemental RAI 13.3-33, in consideration of its responses to RAI 13.3-3(G) and RAI 13.3-11(B)(1), acceptable because it confirms that there are no physical characteristics unique to the proposed LNP site that could pose a significant impediment to the development of emergency plans. Therefore, the staff finds the information provided in Appendix 6 to the LNP Emergency Plan and in its responses to RAIs acceptable because they meet the requirements of 10 CFR 52.81, 10 CFR 100.1(c), and 10 CFR 100.21(g). The staff's review of the ETE Report is in Section 13.3C.18, "Evacuation Time Estimates (ETE) Analysis," of this SER.

13.3B.9 Post Combined License Activities

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

13.3B.10 Conclusion

The staff reviewed the EP information required by regulations to be in the application, but not required to be part of the LNP Emergency Plan provided in Part 5, "Emergency Plan," of the LNP COL application. The staff concludes that the information provided is acceptable and meets the applicable requirements and guidance in 10 CFR 50.33; 10 CFR 50.34(b)(6)(v); 10 CFR 50.34(f)(1), (2), and (3); 10 CFR 50.47(c)(2); 10 CFR 52.73; 10 CFR 52.77; 10 CFR 52.79; 10 CFR 52.81; 10 CFR 100.1(c); 10 CFR 100.21(g); and the applicable portions of Appendix E to 10 CFR Part 50 as discussed above.

ATTACHMENT 13.3C - ONSITE EMERGENCY PLAN

13.3C Introduction

The NRC evaluates emergency plans for nuclear power reactors to determine whether the plans are adequate and there is reasonable assurance that the plans can be implemented. This attachment to the SER provides the results of the staff's review of the onsite emergency plan for the proposed reactors (Units 1 and 2) at the LNP site.

The LNP COL FSAR states in Section 13.3, "Emergency Planning," that the LNP Emergency Plan is included in Part 5 of the COL application. Also included as part of the onsite emergency plan are seven appendices, which provide additional information regarding various aspects of the LNP Emergency Plan (e.g., List of Emergency Plan Supporting Procedures, ETE Study Summary, and Certification Letters). In addition, Part 10 of the COL application includes a set of ITAAC related to the LNP Emergency Plan.

The following section describes the staff's evaluation of the onsite emergency plan for the LNP site and conforms to the evaluation criteria in NUREG-0654/FEMA-REP-1 and Interim Staff Guidance NSIR/DPR-ISG-01, "Emergency Planning for Nuclear Power Plants," associated with the Final Rule on Enhancements to Emergency Preparedness Regulation effective November 23, 2011. Conformance with the guidance in NUREG-0654/FEMA-REP-1 and ISG NSIR/DPR-ISG-01 provides the basis for meeting the requirements of the planning standards in 10 CFR 50.47(b), and Appendix E to 10 CFR Part 50, including the requirements of the Final Rule on Enhancements to Emergency Preparedness Regulation effective November 23, 2011.

13.3C.1 Assignment of Responsibility (Organizational Control)

13.3C.1.1 *Regulatory Basis*

In determining whether the proposed emergency plan met the applicable regulatory requirements in 10 CFR 50.47(b)(1), the staff evaluated it against the detailed evaluation criteria[4] in NUREG-0654/FEMA-REP-1, Revision 1, and guidance in NSIR/DPR-ISG-01. The staff also evaluated the proposed emergency plan against applicable regulatory requirements related to the area of "Assignment of Responsibility (Organization Control)," in Appendix E to 10 CFR Part 50.[5]

[4] The bracketed, alphanumeric designations used throughout this SER section identify the corresponding NUREG-0654/FEMA-REP-1 evaluation criteria used by the staff to determine compliance with 10 CFR 50.47(b).
[5] Braces identify requirements in Appendix E to 10 CFR Part 50.

13.3C.1.2 *Overall Response Organization*

Technical Information in the Emergency Plan: [A.1.a]
Section A.1.a, "State, Local, Federal, and Private Organizations," and Table A-1, "Primary Emergency Response Organizations," of the LNP Emergency Plan provide a listing of principal organizations, including points of contact, participating in emergency response activities within the 10-mile EPZ (plume exposure pathway). The principal organizations include the applicant; State of Florida and government offices of Department of Community Affairs (Division of Emergency Management (DEM)) and Department of Health (Bureau of Radiation Control); the local county Emergency Management offices and municipal entities (Fire and Medical support) from Citrus, Levy, and Marion counties; certain Federal government agencies, including the U.S. Department of Energy (DOE), NRC, and U.S. Department of Homeland Security (DHS), and FEMA; and the Electric Power Research Institute (EPRI), Institute of Nuclear Power Operations (INPO), and Westinghouse.

Figure A-1, "Interrelationships between Key Response Organizations," illustrates the interfaces among functional areas of LNP emergency response activity, Progress Energy corporate support, and the affected State, local, and Federal government response organizations.

In RAI 13.3-17(A)(1), the staff requested the applicant address inconsistencies between Figure A-1 and Section A.1.a of the LNP Emergency Plan which excludes three EROs: the Federal Bureau of Investigation, National Weather Service, and Department of Natural Resources. The applicant's response confirmed that these three organizations could be asked to participate in emergency response activities within the LNP 10-mile EPZ and committed to revise Sections A.1.a and A.1.b of the LNP Emergency Plan.

{Appendix E, Section IV.A.8}
Section A.1.b.1, "State of Florida," of the LNP Emergency Plan identifies the State of Florida as having the primary responsibility for the local population and environs, including the possible need for evacuation. The DEM is identified as being responsible for coordinating Federal, State, and local radiological emergency response activities, and for preparing and maintaining the State of Florida plan. The DEM would also initiate protective action responses that could include the evacuation of radiologically affected areas.

Technical Evaluation: [A.1.a]
The staff finds the additional information and proposed textual revisions provided in the applicant's response to RAI 13.3-17(A)(1) to be acceptable because they conform to the guidance in NUREG-0654/FEMA-REP-1. The staff confirmed the applicant made the referenced changes as discussed above in the LNP Emergency Plan. The staff finds that the LNP Emergency Plan adequately provides a general discussion of the assignment of responsibilities and addresses protective actions. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and meets the requirements of 10 CFR Part 50, Appendix E.

{Appendix E, Section IV.A.8}:
The staff finds that the LNP Emergency Plan adequately identifies State and/or local officials responsible for planning for, ordering, and controlling appropriate protective actions, including evacuations when necessary. This is acceptable because it meets the requirements in Appendix E to 10 CFR Part 50.

13.3C.1.3 *Concept of the Operations*

Technical Information in the Emergency Plan: [A.1.b]
Section A, "Assignment of Responsibility (Organizational Control)," of the LNP Emergency Plan provides a list of participating organizations and a discussion of their respective concepts of operation. Figures A-1 and A-2, "Communications and Interrelationships between Key Response Organizations," illustrate the interrelationships between the organizations participating in an emergency response, and the onsite and offsite ERFs. Figure A-3, "State Organization for Radiological Response," illustrates the relationship between State agencies with emergency response duties. Section A.1.b.9, "Progress Energy – LNP Emergency Response Organization (ERO)," describes the LNP ERO as having the immediate and continuing responsibility for emergency response and control of emergency activities onsite. Section A.1.b.12 in the LNP Emergency Plan describes the LNP ERO for Duke Energy.

{Appendix E, Section III}
LNP COL FSAR Section 13.3 states that the emergency plan describes the plans for coping with emergency situations, including communications interfaces and staffing of the EOF. Section A of the LNP Emergency Plan provides supporting information regarding the concept of operations and emergency response roles of supporting organizations and offsite agencies. In addition, the LNP Emergency Plan describes the facilities, emergency response measures, and functional interfaces with offsite agencies which can be used to respond to a broad range of emergencies. The LNP Emergency Plan has also been coordinated with the plans of affected government agencies and private sector support organizations.

Technical Evaluation: [A.1.b] {Appendix E, Section III}
The staff finds that the LNP Emergency Plan adequately describes the applicant's operational role, its concept of operations, and its relationship to the total effort of emergency response. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and the requirements in Appendix E to 10 CFR Part 50.

13.3C.1.4 *Organizational Interrelationships*

Technical Information in the Emergency Plan: [A.1.c]
Section 13.3C.1.3 in this SER includes discussion regarding organizational interrelationships illustrated in Figures A-1, A-2, and A-3, and Section A of the LNP Emergency Plan.

Technical Evaluation: [A.1.c]
The staff finds that the LNP Emergency Plan adequately illustrates the interrelationships of the participating organizations in emergency response in a block diagram and in text. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.1.5 *Individual in Charge of Emergency Response*

Technical Information in the Emergency Plan: [A.1.d]
The LNP Emergency Plan, Section A.1.b.12, "Progress Energy – LNP Emergency Response Organization (ERO)," identifies the Shift Manager (SM) (formerly Nuclear Shift Manager (NSM)) as the individual who has the responsibility and authority to declare an emergency classification and initiate appropriate actions pursuant to written procedures to mitigate the consequences of that emergency. The SM will assume the role of the EC until relieved by the Plant Manager (PM) (formerly Plant General Manager (PGM)), or designated alternate. The EC is responsible for the direction of all activities at the plant site during any emergency, including evacuation of the site, if necessary, and placing site generating units in a safe shutdown condition. Section B.5.1, "Nuclear Shift Manager," provides a description of the affected unit NSM as assuming the role of the EC, unless a site-wide emergency (e.g., security event or natural phenomena) is declared in which case the Unit 1 NSM would assume the role of the EC. Section B.4, "Emergency Coordinator Responsibilities," of the LNP Emergency Plan provides a detailed discussion regarding the specific responsibilities of the EC, including those responsibilities that the EC is not authorized to delegate. Section B.5.2, "Off-Site Emergency Response Organization," defines the EOF Director as being responsible for overall command and control of the LNP response to the emergency once the offsite ERO is activated. The EOF Director provides information to, and interfaces with, offsite authorities. Additional activities under the purview of the EOF Director include the monitoring of offsite results from the event, protecting plant personnel located outside of the protected area (PA), supporting the onsite organization, and coordinating the flow of information to the public. Revision 6 of the LNP Emergency Plan identifies the Shift Manager (SM) as the individual who has the responsibility and authority to declare an emergency classification and states the SM serves as the EC until the Plant Manager, or designated alternate, arrives to assume the position of EC.

In RAI 13.3-39 (Bullet 4), the staff asked the applicant to incorporate into the emergency plan its description of PEF's response to a simultaneous emergency at LNP and CR3 as it pertains to activation and operation of the EOF. In response to RAI 13.3-39 (Bullet 4), the applicant committed to revise the emergency plan to discuss the specific roles and responsibilities of the EOF facility lead in the event of a simultaneous emergency at both LNP and the CR3 nuclear plant, owned and operated by PEF.

Technical Evaluation: [A.1.d]

The staff finds the additional information and proposed textual revisions provided in response to RAI 13.3-39 (Bullet 4) acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1. The staff confirmed that information provided in response to RAI 13.3-39 (Bullet 4) is incorporated into the LNP Emergency Plan.

By letter dated April 18, 2013, from PEF to NRC regarding impacts from retirement of the CR3 nuclear plant, the applicant renamed the EOF to remove any reference to the Crystal River Training Center. The EOF is now referred to in the LNP Emergency Plan as the LNP EOF. In addition, the applicant added a conditional statement to address when the EOF is required for use by CR3. As discussed in Section 13.3.4, CR3 has been granted an exemption from the need for an EOF. The assignment of the EOF Director as the facility lead for command and control of the EOF response remained unchanged.

On the basis of the staff's review of the LNP Emergency plan, and deletion of reference to the Crystal River Training Center, and added conditional statement, the staff finds that the LNP Emergency Plan adequately identifies a specific individual by title that will be in charge of the emergency response to an event at the LNP site. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.1.6 *24-Hour Response Capability*

Technical Information in the Emergency Plan: [A.1.e]
Section A.1.b, Concept of Operations," of the LNP Emergency Plan identifies 24-hour communication capabilities, including titles of responsible individuals, for the LNP site, the State of Florida, counties of Levy, Marion, and Citrus and various private and Federal organizations. Section F, "Emergency Communications," describes the capability at LNP for 24-hour communications between the CRs or TSCs and the EOF, State and county EOCs, via the State of Florida Hot Ringdown Telephone System. The applicant proposed EP ITAAC 1.1 to verify that EPIPs provide for 24-hour per day emergency response staffing and manning of communication links, including continuous operations for a protracted period.

Technical Evaluation: [A.1.e]
The staff finds that the LNP Emergency Plan describes provisions for 24-hour per day emergency response, including 24-hour per day manning of communications links. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.1.7 *Written Agreements*

Technical Information in the Emergency Plan: [A.3] {Appendix E, Section IV.A.7}
Section A.3, "Written Agreements," of the LNP Emergency Plan states, in part, that DEF has established agreements with local emergency response support services, including firefighting, medical and hospital services listed in Appendix 3. Appendix 3, "Certification Letters," of the LNP Emergency Plan includes a listing of written agreements between DEF (formerly PEF) and associated emergency support organizations. Appendix 3 states, in part, that copies of the original agreements are kept on file by LNP Emergency Preparedness organization or with DEF Contract Services. The original written agreements are included in Part 5, "Emergency Plan," of the COL application. In RAIs 13.3-17(B)(1), 13.3-17(B)(2), and Supplemental RAIs 13.3-28(1), and 13.3-28(2), the staff requested finalized letters of agreement (LOAs) from Federal, State,

and local agencies, and other support organizations having an emergency response role within the LNP EPZs. In its response, in part, the applicant proposed a revised description of the primary function and responsibility of local law enforcement agencies (LLEAs), including response to a hostile action event at the LNP site, in the LNP Emergency Plan. In addition, the applicant proposed a license condition requiring updated LOAs to be in place for all organizations listed in Appendix 3 prior to the full participation exercise to be conducted in accordance with Appendix E to 10 CFR Part 50. Specifically, the applicant proposed License Condition 11(B):

> B. Prior to the full-participation exercise to be conducted in accordance with the requirements of Appendix E to 10 CFR Part 50, DEF will have available for NRC inspection LOAs with entities listed on Appendix 3 of the LNP COL application Part 5, Emergency Plan. These LOAs will detail each entity's specific emergency planning responsibilities and certify the entity's concurrence with their responsibilities.

Technical Evaluation: [A.3] {Appendix E, Section IV.A.7}

The staff finds the additional information and proposed textual revisions provided in response to supplemental RAIs 13.3-28(1) and 13.3-28(2), in consideration of RAIs 13.3-17(B)(1) and 13.3-17(B)(2), to be acceptable because they conform to the guidance in NUREG-0654/FEMA-REP-1. The staff confirmed that the proposed revisions to the LNP Emergency Plan and Part 10 of the COL application provided in response to the above RAIs were included in Part 10 of the LNP COL application.

The applicant proposed License Condition 11(B) in response to supplemental RAI 13.3-28(2):

> B. Prior to the full-participation exercise to be conducted in accordance with the requirements of Appendix E to 10 CFR Part 50, PEF will have available for NRC inspection LOAs with entities listed on Appendix 3 of the LNP COL application Part 5, Emergency Plan. These LOAs will detail each entity's specific emergency planning responsibilities and certify the entity's concurrence with their responsibilities.

Pursuant to the requirements of 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50, and the guidance in NUREG-0654/FEMA-REP-1 and NSIR/DPR-ISG-01, the staff has revised the language in License Conditions 11(A) and (C) to incorporate the requirement for State and local review and agreement of the LNP initial EALs, and development of finalized letters of agreement, originally proposed, in part, in License Condition 11(B) as stated above. These revisions are as follows:

> A. Progress Energy Florida shall submit a fully developed set of site-specific EALs for LNP Units 1 [Unit 2] to the NRC in accordance with NEI 07-01, Revision 0, with no deviations. <u>These EALs shall have been discussed and agreed upon with State and</u>

local officials. These fully developed EALs shall be submitted to the NRC for confirmation at least 180 days prior to initial fuel load.

C. Prior to the full-participation exercise to be conducted in accordance with the requirements of Appendix E to 10 CFR Part 50, PEF will have available for NRC inspection the LOAs established with the following entities:

a. State of Florida Division of Emergency Management
b. Citrus County, Florida Emergency Management Agency
c. Levy County, Florida Emergency Management Agency
d. Marion County, Florida Emergency Management Agency
e. Citrus Memorial Hospital
f. Seven Rivers Regional Medical Center
g. Citrus County, Department of Public Safety Fire Rescue Division
h. Nature Coast Emergency Medical Services Fire Department

These Letters of Agreement shall specify the emergency measures to be provided in support of the LNP emergency organization to include response to a hostile action event at the site; the mutually acceptable criteria and availability of adequate resources for their implementation; and arrangements for the exchange of information.

With the staff's revisions to License Conditions 11(A) and 11(C), the staff finds 11(B) to be redundant. Therefore, License Condition 11(B) has been deleted. With the modifications identified above, the staff finds License Conditions 11(A) and 11(C) to be acceptable.

The staff finds that with the above license condition, the LNP Emergency Plan will include written agreements with support organizations having an emergency response role within its EPZs, including consideration for the availability of adequate resources and response to a hostile action event at the LNP site, prior to the full participation exercise. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and NSIR/DPR-ISG-01 and meets the requirements in Appendix E to 10 CFR Part 50.

13.3C.1.8 *Operations for a Protracted Period*

Technical Information in the Emergency Plan: [A.4]
Section A.4, "Continuous Operations," of the LNP emergency Plan states that DEF maintains the capability for continuous operations through training of multiple responders for key emergency response positions. Section A.1.b.12 of the LNP Emergency Plan states that the LNP ERO is prepared to function on a 24-hour basis. The EC or EOF Director is responsible for ensuring continuity of technical, administrative, and material resources during emergency operations. The applicant proposed EP ITAAC 1.1 to verify that EPIPs provide for 24-hour per day emergency response staffing and manning of communication links, including continuous operations for a protracted period.

Technical Evaluation: [A.4]
The staff finds that the LNP Emergency Plan describes the capability for continuous (24-hour) operation for a protracted period and identifies the individual in the principal organization that will be responsible for continuity of resources. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.1.9 *Conclusion*

On the basis of its review of the onsite emergency plan as described above for assignment of responsibility, the staff concludes that the information provided in the LNP Emergency Plan is acceptable and meets the requirements of 10 CFR 50.47(b)(1) because it conforms with the guidance in Evaluation Criterion A of NUREG-0654/FEMA-REP-1 and NSIR/DPR-ISG-01 and meets the applicable requirements of Appendix E to 10 CFR Part 50 as described above.

13.3C.2 Onsite Emergency Organization

13.3C.2.1 *Regulatory Basis*

In determining whether the proposed emergency plan met the applicable regulatory requirements in 10 CFR 50.47(b)(2), the staff evaluated it against the detailed evaluation criteria in NUREG-0654/FEMA-REP-1 and NSIR/DPR-ISG-01. The staff also evaluated the proposed emergency plan against applicable regulatory requirements related to the area of "Onsite Emergency Organization," in Appendix E to 10 CFR Part 50.

13.3C.2.2 *Normal Plant Operating Organization*

Technical Information in the Emergency Plan: **{Appendix E, Section IV.A.1}**
Section B.1, "On-Site Emergency Organization," and Section B.7, "Corporate Support for the Plant Staff," of the LNP Emergency Plan provide an overview of the normal plant operating organization. In addition, Section B.7 provides a brief description of the organizations reporting hierarchy. Section B.7 further states that the Nuclear Generation organization consists of organizational elements that provide additional administrative and technical support to ensure continued safe plant operation. These elements include Engineering, Support Services, Training and Nuclear Oversight. The corporate structure of DEF is provided in the LNP Emergency Plan.

Technical Evaluation: {Appendix E, Section IV.A.1}
The staff finds that the LNP Emergency Plan adequately describes the normal plant operating organization. This is acceptable because it meets the requirements in Appendix E to 10 CFR Part 50.

13.3C.2.3 *Onsite Emergency Organization*

Technical Information in the Emergency Plan: [B.1] {Appendix E, Section IV.A.2.b} {Appendix E, Section IV.A.9} Section B.1 of the LNP Emergency Plan describes the onsite emergency organization available to respond to a declared emergency at the LNP site. Figures A-1, B-1, B-2, “On-Site Emergency Response Organization (CR, TSC, OSC),” and B-3, “Off-Site Emergency Response Organization (EOF/ENC),” illustrate the interrelationships between the LNP ERO, and associated onsite and offsite ERFs, including their communication interfaces and lines of authority. The narrative in Section B.1 states that plant staff will fill the roles in the ERO that align with their normal staff functions. Table B-1, “Minimum Staffing Requirements for Emergencies,” identifies the minimum staff available onsite, and within a short period to perform key emergency activities. In RAIs 13.3-18(A)(1) and 13.3-18(A)(2)(A) through 13.3-18(A)(2)(E), the staff requested the applicant resolve discrepancies between the narratives in Section B, Figures B-1 and B-2, and Table B-1 of the LNP Emergency Plan. These discrepancies excluded various ERO members from the figures and text in the LNP Emergency Plan, and included inconsistencies between various ERO members and their respective ERF locations. In its response, the applicant provided clarification of the responsibilities for various ERO positions to resolve the identified discrepancies.

In response to the Final Rule on Enhancements to Emergency Preparedness Regulation effective November 23, 2011, the applicant proposed the following addition to License Condition 11 in Part 10 of the COL application:

F. At least two (2) years prior to scheduled initial fuel load, DEF 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.

Technical Evaluation: [B.1] {Appendix E, Section IV.A.2.b} {Appendix E, Section IV.A.9} The staff finds the additional information and proposed textual revisions to the emergency plan provided in response to RAI 13.3-18(A)(1) and RAIs 13.3-18(A)(2)(A) through 13.3-18(A)(2)(E) to be acceptable because they conform to the guidance in NUREG-0654/FEMA-REP-1 and meet the applicable requirements of Appendix E to 10 CFR Part 50. The staff confirmed the proposed changes in these RAIs were incorporated into the LNP Emergency Plan.

By letter dated November 8, 2012, “LNP Emergency Plan Revision 5 Submittal and Proposed License Condition for the On-Shift Staffing Analysis”, from PEF to the NRC, the applicant proposed a license condition to perform an assessment of its emergency response staffing pursuant to the guidance contained in NEI 10-05. Specifically, the assessment will include a detailed analysis to validate whether on-shift personnel are assigned emergency plan implementation functions that would prevent the timely performance of their assigned functions as described in the LNP Emergency Plan. NRC issued associated guidance in Interim Staff Guidance NSIR/DPR-ISG-01 as part of the issuance of the Final EP Rule that endorsed NEI 10-05 as an acceptable methodology for a licensee to perform the required staffing analysis pursuant to Appendix E to 10 CFR Part 50, Section IV.A.9. The staff finds the proposed license condition to be acceptable and verified that Part 10 of the COL application was updated to incorporate this license condition.

Therefore, the staff finds that the LNP Emergency Plan provides an adequate description of the onsite emergency organization of plant staff personnel for all shifts and its relation to the responsibilities and duties of the normal staff complement. This is acceptable because it meets the requirements of Appendix E to 10 CFR Part 50 and conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.2.4 *Designation of an Emergency Coordinator*

Technical Information in the Emergency Plan: [B.2]
Section B.2, "Emergency Coordinator," of the LNP Emergency Plan states that the Nuclear Shift Manager will assume the position of EC of the affected unit until relieved by the PM or an alternate. The EC will assume duties of the position until relieved or upon termination of the emergency. The EC has the responsibility and authority to initiate emergency response actions, including notification of affected State, local, and Federal authorities and providing protective action recommendations (PARs) to offsite authorities. In the LNP Emergency Plan, DEF states that the Shift Manager will assume the position of EC until relieved by the Plant Manager or designated alternate.

Technical Evaluation: [B.2]
The staff finds that the LNP Emergency Plan adequately identifies a designated individual as emergency coordinator, who shall be on shift at all times, and who shall have the authority and responsibility to immediately and unilaterally initiate any emergency actions, including providing protective action recommendations to authorities responsible for implementing offsite emergency measures. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and meets the requirements of Appendix E to 10 CFR Part 50.

13.3C.2.5 *Line of Succession for the Emergency Coordinator*

Technical Information in the Emergency Plan: [B.3]
Section B.3, "Emergency Coordinator Line of Succession," of the LNP Emergency Plan describes the EC line of succession. A designated alternate will assume the responsibilities of the EC if the SM is unable to fulfill his or her duties and responsibilities. The PM or designated alternate will assume the EC role as soon as possible after an emergency classification is determined. Section B.5.1.F, "Emergency Coordinator – CR," of the LNP Emergency Plan states that the assigned alternates to assume the role of the EC during the initial stages of an emergency are on-shift licensed Senior Control Operators designated in accordance with operations' procedures.

Technical Evaluation:
The staff finds that the LNP Emergency Plan adequately identifies a line of succession for the emergency coordinator position, and identifies the specific conditions for higher level utility

officials assuming this function. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.2.6 *Responsibilities of the Emergency Coordinator*

Technical Information in the Emergency Plan: [B.4] {Appendix E, Section IV.A.2.c}
Section B.4 of the LNP Emergency Plan describes the role and responsibilities of the EC. The EC shall not delegate the responsibility for decisions related to:

- Emergency Classification;
- Notifications to State, counties, and the NRC;
- PARs to State and local authorities responsible for offsite emergency measures;
- Approval of planned radiation exposures for LNP personnel in excess of 5 rem total effective dose equivalent (TEDE) or entry into radiation fields greater than 25 rem/hour;
- Review and approval of deviations from Technical Specifications or license conditions if the EC-TSC is a Nuclear Shift Manager (NSM), or ensure that such deviations are approved by a NSM;
- Authorization of the administration of potassium iodide to on-site emergency workers; and
- Termination of the emergency.

Section B.5.1 of the emergency plan states that the SM assumes the role of EC-CR, on the affected unit in an emergency, until relieved by the PM or designated alternate. Following activation of the TSC, overall command and control of the onsite response to the emergency is assumed by the EC-TSC. The EOF Director assumes responsibility for overall command and control of the LNP response to the emergency following activation of the EOF.

{Appendix E, Section IV.A.2.a}
Section B, "On-Site Emergency Organization," of the LNP Emergency Plan describes the onsite ERO. The authorities, responsibilities and duties of individuals who will take charge within this organization are discussed in Sections B.4 through B.5.1 and described in Figures B-1 and B-2, and Table B-1.

Technical Evaluation: [B.4] {Appendix E, Section IV.A.2.c}
The LNP Emergency Plan establishes the functional responsibilities assigned to the emergency coordinator, and clearly specifies which responsibilities may not be delegated to other elements of the emergency organization. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and requirements of Appendix E to 10 CFR Part 50.

{Appendix E, Section IV.A.2.a}:
The staff finds the LNP Emergency Plan adequately describes the onsite ERO with a detailed discussion of the authorities, responsibilities, and duties of the individual(s) who will take charge during an emergency. This is acceptable because it meets the requirements of Appendix E to 10 CFR Part 50.

13.3C.2.7 *On-shift and Augmentation Emergency Response Staff*

Technical Information in the Emergency Plan: [B.5] {Appendix E, Section IV.A.9}
Section B.5, "Plant Emergency Response Staff," of the LNP Emergency Plan provides a description of the positions, titles, and major tasks of onsite and offsite personnel assigned to functional areas of emergency activities. Minimum on-shift staffing requirements, including augmented staffing times for LNP are identified in Table B-1 of the LNP Emergency Plan. Figures B-2 and B-3 of the LNP Emergency Plan illustrate augmented staffing to support activation of ERFs, including minimum staffing and support positions.

In RAI 13.3-18(D)(6), the staff requested additional information from the applicant regarding the basis for its ERO staffing levels. In its response, the applicant stated, in part, that its basis for the staffing composition identified in Table B-1 of the LNP Emergency Plan is Table B-1 of NUREG-0654/FEMA-REP-1, Revision 1. The applicant further stated that Table B-1 includes positions needed for most types of emergencies and is not an all-inclusive list of ERO members that will respond to an event. In supplemental RAI 13.3-45(1), the staff requested that the applicant address the availability of digital instrumentation and controls (I&C) maintenance personnel as part of its staffing basis for Table B-1, and to discuss whether Table B-1 meets its site-specific needs to effectively respond, on-shift and for an extended period of time, to a declared emergency event. In response, the applicant stated, in part, that digital components can be affected during an emergency and Electrical/I&C personnel will be trained in repair and corrective action tasks associated with digital components. One individual capable of performing this function must be on-shift at all times and three additional personnel will augment the shift staffing upon declaration of an Alert or higher emergency. By letter dated June 20, 2011, the applicant supplemented its response to RAI 13.3-45(1) to clarify that the on-shift Electrical/I&C personnel and at least one additional augmented staff member for this position will be trained in digital component repair and corrective action tasks.

Several positions (e.g., Shift Technical Advisor (STA), Unit Senior Control Operators, Control Operators, Dose Projection Team Leader, and maintenance personnel – mechanical, electrical, and I&C) were identified in Table B-1, Figure B-2, or Figure B-3 as being a part of the ERO; however, there was no discussion provided in the LNP Emergency Plan regarding their emergency support functions. In RAIs 13.3-18(A)(1), 13.3-41(1), and supplemental RAI 13.3-29(3)(b), the staff requested that the applicant provide a description of the emergency support functions and responsibilities in the emergency plan for each of the above identified positions. In response, the applicant provided a brief discussion of the primary responsibilities for each position stated above and committed to incorporating this information into the LNP Emergency Plan. The applicant further stated in response to RAI 13.3-42 (Bullet 3) that each ERF (e.g., operational support center (OSC), TSC, EOF, and Emergency News Center (ENC)) will have a corresponding activation and operation EPIP that includes the minimum and augmented staff roles and responsibilities associated with each facility. Appendix 5, "List of Emergency Plan Supporting Procedures," to the LNP Emergency Plan includes the titles of the EPIPs described above.

The augmented staffing times identified in Table B-1 of the LNP Emergency Plan are represented as a range of time, 30-45 minutes and 60-75 minutes, respectively, versus 30 minutes and 60 minutes as identified in NUREG-0654/FEMA-REP-1. In RAI 13.3-18(D)(1), the staff asked the applicant to provide augmented staffing times consistent with the guidance in NUREG-0654/FEMA-REP-1 or explain why extended augmentation times are acceptable. The applicant stated, in part, that notification of the ERO typically occurs within the first 15 minutes of an event. Once notified, ERO members are expected to respond to their respective ERFs within 30 or 60 minutes and be ready to assume responsibility for their ERO function within approximately 15 minutes. Therefore, the ranges of 30-45 minutes and 60-75 minutes shown on Table B-1 include the initial ERO notification time, not to exceed 15 minutes and turnover time to assume the ERO role and responsibility for their respective Table B-1 function. In addition, the applicant provided operating experience from the Crystal River Nuclear Facility (now being decommissioned), owned and operated by PEF, which is located approximately 9 miles from the LNP. The applicant stated that experience from Crystal River has shown that based on local demographics, weather, traffic, and housing availability for station employees, it is achievable to augment staffing within 30 to 60 minutes after notification of an emergency. In supplemental RAI 13.3-45(2), the staff requested the applicant clarify inconsistencies in augmentation times (e.g., the addition of 15 minutes to the 60-75 minute augmentation time) as described in the responses to RAIs 13.3-21B, 13.3-44(2) and 13.3-18(D)(1), or include the response to RAI 13.3-18(D)(1) in the LNP Emergency Plan. In response, the applicant stated, in part, that they will replace the ranges of time (30-45 and 60-75 minutes) for staff augmentation provided in Table B-1 and Section H.4 of LNP Emergency Plan with goals of 30 and 60 minutes to improve the overall clarity of response times for ERO personnel. A 15 minute briefing and turnover time will continue to be used in the facility activation times as described in Section H.4 of the plan. By letter dated June 20, 2011, the applicant supplemented its response to RAI 13.3-45(2) to clarify its ERO augmentation and ERF activation goals.

In RAIs 13.3-18(D)(3), 13.3-18(D)(4), 13.3-18(D)(5), 13.3-18(D)(7), and supplemental RAIs 13.3-29(2) and 13.3-29(3)(a), the staff requested additional clarification regarding collateral and potentially competing duties for the following ERO positions identified in Table B-1 of the LNP Emergency Plan: mechanical, electrical and I&C maintenance, fire brigade, emergency communicator, and the STA. The applicant's response to these RAIs included the following key points:

- Current staffing plans are such that each maintenance discipline will fill their own respective vacancies (e.g., mechanical maintenance positions will be filled with mechanical maintenance personnel) with the exception of fire brigade members performing the functions of first aid and rescue operations. During emergency situations, the mechanical and electrical maintenance shift members do not have collateral duties. Any staffing decisions made for LNP that are different than stated above will be in compliance with Table B-1, and staff will be trained and qualified personnel that do not have collateral emergency response duties.

- The fire brigade will consist of at least five onsite (per shift) trained and qualified members in accordance with the FSAR. The exact composition of the fire brigade may vary per shift among qualified responders, and personnel assigned to the fire brigade will not have collateral duties that compete or conflict with fire brigade responsibilities. The fire brigade is typically composed of operations personnel; however, if other personnel assume brigade responsibilities they will be trained and qualified to the same qualifications described in the LNP Emergency Plan. The LNP fire brigade members are trained in first aid and rescue operations. In an emergency situation that does not involve a fire, the fire brigade members are readily available for any needed first aid and rescue operations. In the event of a fire, the fire brigade will be on scene and handle any injured personnel near the fire as instructed per routine fire training and response. The ability to handle and address injured personnel in a fire is standard for fire fighters. Onsite support will be augmented by offsite fire rescue that would handle fire fighting and first aid activities beyond the capability of the onsite team.

- Typically a non-licensed operator will be assigned the role of Emergency Communicator, and the non-licensed operator will not have any collateral duties. In lieu of a non-licensed operator, a trained and qualified licensed operator may fill the role of Emergency Communicator if the shift complement could accommodate this assignment without any collateral duties. The Emergency Communicator position will not be augmented with operations personnel once the TSC and/or EOF are operational. Personnel assigned the role of Emergency Communicator will be trained and qualified to do so without collateral duties.

- The responsibility for an STA during transients or accident situations is to assess plant conditions and provide technical assistance and advice to mitigate an event. No additional collateral duties will be added to the STA or Senior Reactor Operator/STA position.

In RAI 13.3-29(1), the staff asked the applicant to discuss the inconsistency between the radiological control team members staffing for on-shift protective actions (in-plant) specified in Table B-1 of the LNP Emergency Plan versus the associated shift staffing levels identified in Table B-1 of NUREG-0654/FEMA-REP-1. In addition, the staff asked the applicant to clarify whether the staffing in Table B-1 of the LNP Emergency Plan is applicable to Unit 1 only, or Units 1 and 2 combined. In response, the applicant stated that Table B-1 of the LNP Emergency Plan will be revised to be consistent with Table B-1 of NUREG-0654/FEMA-REP-1. The LNP Table B-1 will show 2 members of the radiological control team on-shift for Unit 1 with an additional member on-shift for Units 1 and 2. A footnote allowing the function to be performed by shift personnel assigned other functions will also be added to these positions. In supplemental RAI 13.3-45(3), the staff asked the applicant to revise Table B-1 of the LNP Emergency Plan to correct the total staffing for radiological control team members consistent with its response to RAI 13.3-29(1), and to address the footnote added to this position by discussing any collateral duties or competing priorities that could have an impact on performing the positions' emergency response function. In response, the applicant stated, in part, that Table B-1 will be revised to be consistent with the response to RAI 13.3-29(1) as described above. The footnote provided for this position is consistent with NUREG-0654/FEMA-REP-1. In addition, the applicant stated, in part, that LNP radiological control team personnel will not have collateral duties during emergency situations, and any on-shift personnel required to perform in-plant protective actions will be trained and qualified to do so.

In supplemental RAI 13.3-45(4), the staff requested that the applicant clarify whether the Radiation Monitoring Team personnel described in Section I.4.1, "On-site Dose Assessment," of the LNP Emergency Plan are the same as the Environmental Monitoring Team personnel identified in LNP Table B-1. In response, the applicant referred to its response for RAI 13.03-47 in which it proposed, in part, to revise Table B-1 to identify the Radiological Monitoring Team as being responsible for performing the major task of Off-site Surveys. The applicant stated that this change in nomenclature should more appropriately align with NUREG-0654/FEMA-REP-1 and distinguish between LNP and State Monitoring Teams.

In response to the Final Rule on Enhancements to Emergency Preparedness Regulation effective November 23, 2011, the applicant proposed the following addition to License Condition 11 in Part 10 of the COL application:

F. At least two (2) years prior to scheduled initial fuel load, DEF 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.

The applicant proposed EP ITAAC 2.1 to verify EPIPs exist that provide for minimum and augmented on-shift staffing levels consistent with Table B-1 of the LNP Emergency Plan.

Technical Evaluation: [B.5] {Appendix E, Section IV.A.9}
The staff finds the clarifying information and proposed textual revisions to the LNP Emergency Plan provided in response to RAIs 13.3-18(A)(1), 13.3-18(D)(2), 13.3-18(D)(5), 13.3-18(D)(7) through 13.3-18(D)(10), and supplemental RAIs 13.3-29(3)(b), 13.3-41(1), 13.3-42 (Bullet 3), 13.3-45(4), and 13.3-47 to be acceptable because they conform to the guidance in NUREG-0654/FEMA-REP-1 and meet the requirements in Appendix E to 10 CFR Part 50.

The staff confirmed that the proposed revisions provided in response to RAIs 13.3-18(A)(1), 13.3-18(D)(2), 13.3-18(D)(3), 13.3-18(D)(7), 13.3-18(D)(9), and 13.3-18(D)(10) have been incorporated into the LNP Emergency Plan. The staff also confirmed that the proposed revisions provided in response to RAIs 13.3-29(3)(b), 13.3-41(1), and 13.3-42 (Bullet 3), have been incorporated into the LNP Emergency Plan.

The staff created Confirmatory Item 13.3-47 to track the proposed change to LNP Table B-1 consistent with the applicant's response to RAI 13.3-47.

The staff finds the applicant's response to supplemental RAI 13.3-41(2), in consideration of its responses to RAI 13.3-18(D)(3) and supplemental RAI 13.3-29(3)(a), to be acceptable because it corrects inconsistencies regarding collateral duties for Maintenance personnel in Table B-1 of the LNP Emergency Plan, conforms to the guidance in NUREG-0654/FEMA-REP-1, and meets the requirements in Appendix E to 10 CFR Part 50. The staff confirmed that Table B-1 of the

LNP Emergency Plan clarified that the maintenance personnel will not have collateral duties during an emergency.

The staff finds the applicant's response to supplemental RAI 13.3-29(2), in consideration of its response to RAI 13.3-18(D)(4), to be acceptable because it provides clarification regarding the fire brigade composition and collateral duties, conforms to the guidance in NUREG-0654/FEMA-REP-1, and meets the requirements in Appendix E to 10 CFR Part 50. The staff confirmed the applicant revised Section B.5.1 of the LNP Emergency Plan to reflect that fire brigade members will not have collateral emergency response duties that compete or conflict with fire brigade response.

The staff finds that the additional information and proposed revisions to the minimum staff augmentation and activation goals provided in response to RAI 13.3-45(2) and its supplement, in consideration of its prior response to RAIs 13.3-21(B) and 13.3-18(D)(1) and 13.3-44(2), to be acceptable because it describes provisions for a timely staff augmentation and activation of the ERFs, and conforms to the guidance in NUREG-0654/FEMA-REP-1. The staff created Confirmatory Item 13.3-45(2**)** to track the proposed textual revisions to the emergency plan consistent with the Applicant's RAI responses.

The staff finds the applicant's response to supplemental RAI 13.3-45(1), in consideration of its response to RAI 13.3-18(D)(6), to be acceptable because it identifies on-shift personnel who will be trained and qualified to work on digital components, as needed, when performing repair and corrective actions during an emergency. This conforms to the guidance in NUREG-0654/FEMA-REP-1 and meets the requirements in Appendix E to 10 CFR Part 50. The staff created Confirmatory Item 13.3-45(1) to track the proposed textual revision of the emergency plan consistent with the Applicant's RAI responses.

The staff finds the applicant's response to supplemental RAI 13.3-45(3), in consideration of its response to RAI 13.3-29(1), to be acceptable because it includes a proposed revision to Table B-1 of the LNP Emergency Plan, consistent with its response to RAI 13.3-29(1), that aligns with the minimum shift staffing number (3 versus 1) of radiological control team members supporting the major task of on-shift protective actions. In addition, the applicant proposed a revision to the LNP Emergency Plan clarifying that the radiological control team members described above will be qualified to perform their tasks identified in Table B-1 without collateral duties that compete or conflict with their ERO responsibilities. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1. The staff created Confirmatory Item 13.3-45(3) to track the applicant's inclusion of its response into the LNP Emergency Plan.

By letter dated November 8, 2012, from PEF to the NRC, the applicant proposed a license condition to perform an assessment of its emergency response staffing pursuant to the guidance contained in NEI 10-05. Specifically, the assessment will include a detailed analysis to validate whether on-shift personnel are assigned emergency plan implementation functions that would prevent the timely performance of their assigned functions as described in the LNP Emergency Plan. NRC issued associated guidance in Interim Staff Guidance NSIR/DPR-ISG-01 as part of the issuance of the Final EP Rule that endorsed NEI 10-05 as an acceptable methodology for a licensee to perform the required staffing analysis pursuant to Appendix E to 10 CFR Part 50, Section IV.A.9.

The staff finds the proposed license condition (11F) to be acceptable with the exception of the reference to the scheduled date for initial fuel load. License Condition (13-7) is modified to be consistent with the completion of EP ITAAC 2.0.

Resolution of Confirmatory Items 13.3-45(1), 13.3-45(2), 13.3-45(3), and 13.3-47

Confirmatory Items 13.3-45(1), 13.3-45(2), 13.3-45(3), and 13.3-47 are applicant commitments to update the LNP Emergency Plan. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). As a result, Confirmatory Items 13.3-45(1), 13.3-45(2), 13.3-45(3), and 13.3-47 are now closed.

On the basis of its review, the staff finds that the LNP Emergency Plan specifies the positions or titles and major tasks to be performed by the persons to be assigned to the functional areas of emergency activity. For emergency situations, specific assignments were made for all shifts and for plant staff members, both onsite and away from the site. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and NSIR/DPR-ISG-01 and meets the requirements in Appendix E to 10 CFR Part 50. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.2.8 *Interfaces Between Functional Areas*

Technical Information in the Emergency Plan: [B.6]
Section B.6, "Interfaces Between Functional Areas," of the LNP Emergency Plan states that Figure A-1 illustrates the interfaces among functional areas of LNP emergency response activity, Progress Energy corporate support, and the affected State and local, and Federal government response organizations. In addition, Figure B-1 of the LNP Emergency Plan further illustrates the interrelationship and interface between the LNP ERO, associated onsite and offsite ERFs, Federal, State and county government response organizations, and local support services. The staff requested additional clarification from the applicant in RAI 13.3-18(A)(3), regarding the identification of Federal agencies, other than the NRC Headquarters, that interface with the LNP site. The applicant's response included an updated Figure A-1 revising its illustrated interface with the NRC Regions, the DHS/FEMA, DOE, Federal Bureau of Investigation (FBI), and National Weather Service (NWS).

Technical Evaluation:
The staff finds the additional information and proposed textual revisions to the emergency plan provided in response to RAI 13.3-18(A)(3) to be acceptable because they conform to the guidance in NUREG-0654/FEMA-REP-1. The staff confirmed that the revision to Figure A-1 provided in this RAI response is included in the LNP Emergency Plan. The staff finds that the LNP Emergency Plan adequately specifies the interfaces between and among the onsite functional areas of emergency activity, licensee headquarters support, local services support, and State and local government response organization. The interfaces were illustrated in a

block diagram, and included the onsite TSC, OSC, and the applicant's EOF. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.2.9 *Corporate Support*

Technical Information in the Emergency Plan: [B.7] {Appendix E, Section IV.A.3}
Section B.7, "Corporate Support for Plant Staff," of the LNP Emergency Plan states that Progress Energy's Nuclear Generation organization consists of organizational elements that provide additional administrative and technical support to ensure continued safe plant operation. In the LNP Emergency Plan, the DEF corporate structure that directly controls and supports the operation of LNP is described. Upon declaration of an emergency, as conditions warrant, management, technical, and administrative personnel staff the ERFs and provide support as shown in Table B-1. In the event of an emergency at LNP that requires personnel and other support resources beyond those available within the LNP ERO, augmentation support is available from offsite sources (e.g., Nuclear Generation organization) and further described in plant procedures. The following areas receiving corporate support during an emergency include:

a. logistics support for emergency personnel (e.g., transportation, communications, temporary quarters, food and water, sanitary facilities in the field, and special equipment and supplies procurement)

b. technical support for planning and reentry/recovery operations

c. management level interface with governmental authorities

d. release of information to news media during an emergency (coordinated with governmental authorities)

In RAI 13.3-18(B), the staff requested that the applicant clarify in the emergency plan which support personnel will augment logistics support for emergency personnel. In response, the applicant stated that the EOF Facility Manager is responsible for logistics support during an emergency. Administrative staff in the EOF will assist the Facility Manager in procuring needed supplies and resources. Specifics regarding the responsibilities of the EOF Facility Manager and administrative staff are included in the implementing procedures. The applicant committed to revise Figure B-3 to clarify the responsibility of the EOF Facility Manager.

In RAI 13.3-65, the staff requested the applicant to clarify in the emergency plan the support role of CR3 and describe how it is notified of an emergency at LNP. In response, the applicant stated, in part, that CR3 is an extension of the corporate support provided by Progress Energy and therefore is notified of an emergency at LNP in accordance with EPIPs. The applicant proposed a revision to the LNP Emergency Plan that would clarify this information.

Technical Evaluation: [B.7] {Appendix E, Section IV.A.3}
The staff finds the additional information and proposed textual revisions to the emergency plan provided in response to RAIs 13.3-18(B) and 13.3-65 to be acceptable because they conform to the guidance in NUREG-0654/FEMA-REP-1 and meet the requirements of Appendix E to 10 CFR Part 50. The staff confirmed the changes proposed in response to RAI 13.3-18(B) and RAI 13.3-65 were incorporated into the LNP Emergency Plan. Therefore, the staff finds that the LNP Emergency Plan adequately describes who in the corporate management, administrative, and technical support personnel will augment the plant staff during emergency events. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and the requirements of Appendix E to 10 CFR Part 50.

13.3C.2.10 *Contractor and Private Organizations Support*

Technical Information in the Emergency Plan: [B.8] {Appendix E, Section IV.A.5}
Section B.8, "Support from Contractor and Private Organizations," of the LNP Emergency Plan lists contractor and private organizations that are available to assist in emergency response at the LNP site. In RAI 13.3-18(C), the staff requested that the applicant provide additional information identifying, by position and function to be performed, other employees of the licensee or consultants with special qualifications for coping with emergency conditions that may arise, including the special qualifications of those persons. In its response, the applicant committed to revise Section B.8 of the LNP Emergency Plan to include a discussion of services provided by INPO, American Nuclear Insurers (ANI), DOE Radiation Emergency Assistance Center/Training Site (REAC/TS), and Westinghouse Electric Company, LLC.

Technical Evaluation: [B.8] {Appendix E, Section IV.A.5}
The staff finds the additional information and proposed textual revisions to the emergency plan provided in response to RAI 13.3-18(C) to be acceptable because they conforms to the guidance in NUREG-0654/FEMA-REP-1 and meet the requirements in Appendix E to 10 CFR Part 50. The staff confirmed the changes proposed in response to RAI 13.3-18(C) were incorporated in the LNP Emergency Plan. The staff finds that the LNP Emergency Plan adequately specifies the contractor and private organizations that may be requested to provide technical assistance to, and augmentation of, the emergency organization. The staff also finds that the LNP Emergency Plan adequately identifies, by position and function to be performed, other employees of the licensee with special qualifications for coping with emergency conditions that may arise or other persons with special qualifications, such as consultants, who are not employees of the licensee, and who may be called upon for assistance for emergencies. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and meets the requirements in Appendix E to 10 CFR Part 50.

13.3C.2.11 *Local Emergency Response Support*

Technical Information in the Emergency Plan: [B.9] {Appendix E, Section IV.A.6}
Sections A.1.b, "Concept of Operations," of the LNP Emergency Plan describes local services (e.g., fire departments, hospitals, and LLEA) available to support the LNP ERO. This section includes a description of the support role of Citrus, Levy, and Marion County emergency

management organization. CR3 is listed in Table C-1 of the LNP Emergency Plan as having radiological laboratories available to support the processing of highly radioactive samples, if necessary. Table L-1, "Summary of Actions for Emergency Medical Treatment," identifies local offsite medical facilities that are utilized depending upon the type of injury sustained and degree of contamination, if any. Additional information regarding written agreements of support organizations having an emergency response role within the LNP EPZs is in Section 13.3C.1.7 of this SER.

Technical Evaluation: [B.9] {Appendix E, Section IV.A.6}
By letter dated April 18, 2013, from PEF to NRC regarding impacts from retirement of CR3 nuclear plant, the applicant stated they would delete the reference to Crystal River 3 nuclear plant from Table C-1 of the LNP Emergency Plan as providing local support to analyze highly radioactive samples during an emergency at LNP. The staff confirmed that Table C-1 of the LNP Emergency Plan does not reference CR3.

In consideration for the above CR3 reference deletion, the staff finds that the LNP Emergency Plan adequately identified, or provided reference to, the services to be provided by local agencies for handling emergencies (e.g., police, ambulance, medical, hospital, and fire-fighting organizations). The staff also finds that the LNP Emergency Plan adequately incorporates, or provides reference to, information about the emergency response roles of supporting organizations and offsite agencies. The information in the onsite emergency plan is sufficient to provide assurance of coordination among the support groups and with the licensee. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and meets the requirements in Appendix E to 10 CFR Part 50.

13.3C.2.12 *Conclusion*

On the basis of its review of the LNP Emergency Plan as described above for the onsite emergency organization, the staff concludes that the information provided in the LNP Emergency Plan is acceptable and meets the requirements of 10 CFR 50.47(b)(2) because it conforms with the guidance in Evaluation Criterion B of NUREG-0654/FEMA-REP-1 and NSIR/DPR-ISG-01 and complies with the applicable requirements of Appendix E to 10 CFR Part 50 as described above.

13.3C.3 Emergency Response Support and Resources

13.3C.3.1 *Regulatory Basis*

In determining whether the proposed emergency plan met the applicable regulatory requirements in 10 CFR 50.47(b)(3), the staff evaluated it against the detailed evaluation criteria in NUREG-0654/FEMA-REP-1 and guidance in NSIR/DPR-ISG-01. The staff also evaluated the proposed emergency plan against applicable regulatory requirements related to the area of "Emergency Response Support and Resources," in Appendix E to 10 CFR Part 50.

13.3C.3.2 *Individual Authorized to Request Federal Support*

Technical Information in the Emergency Plan: [C.1.a]
Sections A.1.b.12 and B.4 of the LNP Emergency Plan describe the responsibilities of the EC. Specifically, should the EC determine that extreme measures need to be taken in order to maintain control of an emergency situation, the EC has the authority to direct personnel to evacuate the LNP site, direct a safe shutdown, initiate accountability activities, notify all applicable agencies of the plant status or required outside assistance. Section C.1, "Federal Response Capability," of the LNP Emergency Plan states, in part, that under some complex circumstances, the EOF Director may request assistance directly or through the NRC (Federal coordinating agency).

Technical Evaluation: [C.1.a]
The staff finds that the LNP Emergency Plan adequately addresses the individuals authorized to request Federal support because the description conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.3.3 *Expected Assistance from State, Local, and Federal Agencies*

Technical Information in the Emergency Plan: [C.1.b] {Appendix E, Section IV.A.7}
Section A.1.b of the LNP Emergency Plan describes the primary State, local, and Federal organizations and expected emergency response support to be provided to PEF during an event at the LNP site. Section C of the LNP Emergency Plan states that support from State, local, and Federal agencies includes assistance for onsite activities in response to a hostile action event that is sufficient to cope with potential events. Section C.1 of the LNP Emergency Plan states that the NRC, acting as the cognizant Federal agency, will initiate and coordinate Federal response for the emergency under the National Response Framework (NRF).
Section C.1.b of the LNP Emergency Plan states that PEF estimates that NRC support would arrive at the site within 3-4 hours (based on driving time; shorter if using aircraft) following the notification to deploy. Furthermore, PEF expects NRC assistance from NRC offices in Atlanta, Georgia, will arrive in the LNP site vicinity within 7 to 8 hours following notification. This time may also be reduced using aircraft. Federal radiological monitoring assistance may be provided by the NRC. By letter dated February 16, 2011, "Response to Request for Additional Information Letter No. 100 Related to Emergency Planning," to the NRC from PEF, the applicant provided additional information to clarify the NRC's expected response time to an LNP emergency. The applicant removed the reference to the NRC providing radiological monitoring assistance from the emergency plan and stated, in part, that NRC assistance is expected at the LNP site within approximately 8 hours following notification and drive time. The team may reduce this time by use of aircraft. Section A.1.b.11, "Department of Homeland Security [DHS/Federal Emergency Management Agency (FEMA)]," states that DHS and its subordinate agency FEMA are assigned lead responsibility for Federal offsite nuclear EP and response. DHS/FEMA Region IV and the Federal Bureau of Investigation will provide assistance to the LNP as needed.

In response to the Final Rule on Enhancements to Emergency Preparedness Regulation effective November 23, 2011, the applicant provided additional information in Section C, "Emergency Response Support and Resources," of the LNP Emergency Plan, which states that support from State, local, and Federal agencies includes assistance for onsite activities in response to a hostile action event that is sufficient to cope with potential events. In RAI 13.3-59, the staff requested the applicant to discuss whether acknowledgement of a State, local and Federal response to a hostile action event had been incorporated into Section A of the LNP Emergency Plan, which describes the emergency responsibilities of various support organizations having an operational role with the LNP EPZs. In response, the applicant provided a discussion that concluded, in part, that although the information in Section A of the LNP Emergency Plan does not explicitly mention hostile action support, the information contained in Section A is adequate. By letter dated April 26, 2013, "Supplemental Response to NRC RAI Letter 111 Related to SRP Section 13.3", the applicant supplemented its response to clarify in the LNP Emergency Plan that hostile action response is one of the emergency responsibilities for local law enforcement agencies.

Technical Evaluation: [C.1.b] {Appendix E, Section IV.A.7}
The staff finds the applicant's clarification regarding NRC's expected response time during an emergency and the roles of offsite response agencies during hostile actions, in consideration of its supplemental response, to be acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and NSIR/DPR-ISG-01 and meets the requirements of Appendix E to 10 CFR Part 50. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). Therefore, on the basis of its review, the staff finds that the LNP Emergency Plan adequately identifies the assistance expected from appropriate State, local, and Federal agencies with responsibilities for coping with emergencies. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and NSIR/DPR-ISG-01 and meets the requirements in Appendix E to 10 CFR Part 50.

13.3C.3.4 *Resources to Support the Federal Response*

Technical Information in the Emergency Plan: [C.1.c]
Section C.1.c of the LNP Emergency Plan states that DEF will provide facilities and resources needed to support the Federal response through the EOF. DEF will provide office space and telephone communications for NRC personnel in the TSC, EOF, and ENC. Section A.1.b.2, "State of Florida Department of Community Affairs, Division of Emergency Management (DEM)," of the LNP Emergency Plan states that the DEM provides personnel and equipment to ERFs, and provides needed supplies to State and local political subdivisions. The State Emergency Management Communications Network, the State Hot Ringdown Telephone System, and the Florida Emergency Satellite Communications System (ESATCOM) communication systems are also available to the DEM. Section H.3, "State/County Emergency Operations Centers," lists the State Emergency Operations Center (SEOC), the State Warning Point-Tallahassee (SWPT), and the Citrus, Levy, and Marion County EOCs as facilities utilized in the event of an LNP emergency. Section H.3, "State/County Emergency Operations Centers," also states that implementing procedures describe the inter-relationship of DEF with these centers and Federal agencies. Appendix 3 identifies certification letters with

organizations that may be required to provide support during an emergency at LNP. Signed copies of the letters are provided.

Technical Evaluation: [C.1.c]
The staff finds that the LNP Emergency Plan adequately 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. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.3.5 *Representatives to Offsite Governments*

Technical Information in the Emergency Plan: [C.2.b]
Section C.2, "Off-Site Organization Representation in the Emergency Operations Facility," of the LNP Emergency Plan states that the EOF organization will dispatch a representative to principal offsite State and local EOCs to provide technical expertise and assistance to these organizations. Section B.5.f, "Representatives to the State/County EOCs," states that representatives sent to the State/County EOCs are located in the Florida State EOC State Administrative Building in Tallahassee, Florida; the Citrus County EOC in Lecanto, Florida; the Levy County EOC in Bronson, Florida; and the Marion County EOC in Ocala, Florida.

Technical Evaluation: [C.2.b]
The staff finds that the LNP Emergency Plan adequately addresses the dispatch of a representative to principal offsite governmental EOCs. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.3.6 *Radiological Laboratory Support*

Technical Information in the Emergency Plan: [C.3]
Table C-1, "Radiological Laboratories – Capabilities," of the LNP Emergency Plan identifies three radiological laboratories and their capabilities: post-accident analyses and monitoring of radioactive samples. In addition, the LNP Emergency Plan states that the LNP ERO is authorized to use these laboratories in an emergency situation, which are expected to respond once resources become available. Section C.3, "Radiological Laboratories," states that the Department of Health, Bureau of Radiation Control (DHBRC) will provide services for low-level radioactivity samples and environmental monitoring.

Technical Evaluation: [C.3]
By letter dated April 18, 2013, from PEF to NRC regarding impacts from retirement of CR3 nuclear plant, the applicant removed reference to Crystal River 3 nuclear plant from Table C-1 of the LNP Emergency Plan as providing local support to analyze highly radioactive samples during an emergency at LNP. The staff confirmed that CR3 is not referenced in Table C-1 of the LNP Emergency Plan.

In consideration for the above CR3 reference deletion, the staff finds that the LNP Emergency Plan adequately identifies radiological laboratories, their general capabilities, and expected availability to provide radiological monitoring and analyses services which can be used in an emergency. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.3.7 *Other Sources of Assistance*

Technical Information in the Emergency Plan: [C.4]
Section A of the LNP Emergency Plan includes a listing of State and county facilities available to provide assistance to LNP during an emergency. Section B.8 provides a listing of contractor and private organizations that are considered part of the overall response organization. Radiological laboratories and their general capabilities are identified in Table C-1. Section C.4, "Other Supporting Organizations," of the LNP Emergency Plan states, in part, that Oak Ridge Associated Universities is available to provide backup medical care and treatment of personnel. Appendix 3 includes Letters of Certification and Agreement with organizations that may be required to provide support to LNP during a classified emergency. Signed copies of these letters were provided.

{Appendix E, Section III}
The LNP FSAR Section 13.3-2 states that the emergency plan describes the plans for coping with emergency situations, including communications interfaces and staffing of the EOF. Section A, "Assignment of Responsibility (Organizational Control)," of the LNP Emergency Plan provides supporting information regarding the concept of operations and emergency response roles of supporting organizations and offsite agencies.

Technical Evaluation: [C.4]
The staff finds that the LNP Emergency Plan adequately identifies the other sources of assistance expected to support any emergency response. This is acceptable because it conforms to the guidance in NUREG 0654/FEMA-REP-1.

{Appendix E, Section III}:
The staff finds that the LNP Emergency Plan adequately describes the applicant's operational role, its concept of operations, and its relationship to the total effort. This is acceptable because it meets the requirements in Appendix E to 10 CFR Part 50.

13.3C.3.8 *Conclusion*

On the basis of its review of the onsite emergency plan as described above for emergency response support and resources, the staff concludes that the information provided in the LNP Emergency Plan is acceptable and meets the requirements of 10 CFR 50.47(b)(3) because it conforms with the guidance in Evaluation Criterion C of NUREG-0654/FEMA-REP-1 and NSIR/DPR-ISG-01 and complies with applicable requirements of Appendix E to 10 CFR Part 50 as described above.

13.3C.4 Emergency Classification System

13.3C.4.1 *Regulatory Basis*

In determining whether the proposed emergency plan met the applicable regulatory requirements in 10 CFR 50.47(b)(4), the staff evaluated it against the detailed evaluation criteria in NUREG-0654/FEMA-REP-1 and guidance in NSIR/DPR-ISG-01. The staff also evaluated the proposed emergency plan against applicable regulatory requirements related to the area of "Emergency Classification System," in Appendix E to 10 CFR Part 50.

13.3C.4.2 *Emergency Classification System*

Technical Information in the Emergency Plan: [D.1 and D.2] {Appendix E, Section IV.B and IV.C}
Section D, "Emergency Classification System," of the LNP Emergency Plan describes four emergency classes and includes a brief statement of purpose for each emergency classification level: Notification of Unusual Event (NOUE), Alert, Site Area Emergency (SAE), and General Emergency (GE). Section D.2, "Emergency Action Levels," and Appendix 4, "Emergency Action Levels," incorporate by reference NEI 07-01, "Methodology for Development of Emergency Action Levels for Advanced Passive Light Water Reactors," Revision 0, as the basis for the LNP EAL scheme. Section D.2 states that Appendix 4 provides the parameter values and equipment status that will be used in classifying emergencies at LNP. In addition, Appendix 4 includes five recognition category matrices, and a statement to support that the emergency classification and EAL scheme has been reviewed by the State of Florida and local counties of Citrus, Levy, and Marion, and will continue to be reviewed by the State and local authorities on an annual basis.

The applicant proposed EP ITAAC 3.1 to verify that the specified parameters (facility system and effluent) are retrievable in the CRs, TSC, and EOF, and the ranges of displays encompass the values specified in the emergency classification and action level scheme. Appendix 5, "List of Emergency Plan Supporting Procedures," identifies an EPIP entitled, "Emergency Classification."

In RAI 13.3-01, the staff requested the applicant address its plans to finalize the LNP emergency classification and action level scheme and provided them with two options. Option 1 was the submission of an entire EAL scheme, which includes all site-specific information. Option 2 had four parts (critical elements) that addressed the submission of an overview of the EAL scheme using NEI 07-01, Revision 0, and the proposal of a license condition that addresses EAL completion and submission to the NRC. In response, the applicant selected Option 2. The applicant provided the following information: a definition and statement of purpose for each emergency class; a license condition committing to the use NEI 07-01 or an equivalent NRC endorsed EAL scheme with no deviations; a State and local government review and approval of the proposed EALs; and a statement indicating that the fully developed EAL scheme will be incorporated into an EPIP or the LNP Emergency Plan controlled pursuant to 10 CFR 50.54(q). The applicant supplemented its response, which provided additional information to clarify the revision of NEI 07-01 (Revision 0) to be used as the technical basis for its EALs, and changed the license condition submittal date of its EAL scheme to the NRC. In addition, the applicant proposed revisions to the emergency plan, which, in part, removed a requirement for the applicant to collaborate with, and obtain approval of its EAL scheme, from State and local government authorities.

Revision 1 to Part 10, "Proposed License Conditions (Including ITAAC)," of the COL application includes the following License Condition (No. 11, Emergency Planning Actions):

A. Progress Energy Florida shall submit a fully developed set of site-specific Emergency Action Levels (EALs) for Levy Units 1 (Unit 2) to the NRC in accordance with NEI 07-01 Revision 0, with no deviations. These fully developed EALs shall be submitted to the NRC for confirmation at least 180 days prior to initial fuel load.

In supplemental RAI 13.3-30, the staff asked the applicant to provide the revised language to the LNP Emergency Plan that includes a general list of licensee actions for each emergency classification, and a license condition to ensure that the final version of the initial EALs will be discussed with, and agreed upon, by State and local governmental authorities at least 180 days prior to fuel load. In response, the applicant provided revised language to Section D of the LNP Emergency Plan and a reference to an EPIP where the fully developed EAL scheme will be included. In addition, the applicant proposed to remove Appendix 4 from the emergency plan and mark it as "Not Used" since the LNP EALs have not been fully developed, and revised a proposed license condition developed in response to supplemental RAI 13.3-28(2) to include the concurrence of the State and local governments with the LNP EALs. The applicant submitted a supplemental response to this RAI which incorporated the State and local government review requirement into the emergency plan that had been previously deleted during the removal of Appendix 4.

Revision 2 to Part 10, "Proposed License Conditions (Including ITAAC)," of the COL application includes the following License Condition, in part (No. 11, Emergency Planning Actions):

C. These Letters of Agreement will certify each agency's concurrence with the emergency action levels described in LNP Units 1 and 2 COLA Part 5 Emergency Plan.

In response to the Final Rule on Enhancements to Emergency Preparedness Regulation effective November 23, 2011, the applicant provided additional information in Section D.3, "Emergency Declaration," of the LNP Emergency Plan that states that LNP maintains the capability to assess, classify and declare an emergency condition within 15 minutes following the availability of indications to cognizant facility staff that an emergency action level has been exceeded. Section D.3 further states that the Shift Manager/Emergency Coordinator is responsible for promptly declaring the emergency condition following identification of the appropriate emergency classification level, consistent with the need to provide for public health and safety. The LNP Emergency Plan contains additional information in Section D.3 to clarify that this 15 minute criterion is not to be construed as a grace period for restoring plant conditions to avoid declaring an EAL, and should not limit response actions necessary to protect public health and safety. Additional details describing the timeliness of emergency declaration are contained in LNP EPIPs.

Technical Evaluation: [D.1 and D.2] {Appendix E, Section IV.B and IV.C}
The staff finds the applicant's definition of the four emergency classifications (NOUE, Alert, SAE, GE) introduced in Section D of Revision 1 to the LNP Emergency Plan acceptable because they are consistent with the emergency classifications described in Appendix E to 10 CFR Part 50 and defined verbatim with NRC endorsed guidance NEI 07-01, Revision 0, which includes security-based events.

In Section D.2 of the LNP Emergency Plan, the staff finds the applicant's reference to NEI 07-01, Revision 0, as the technical basis for development of the LNP site-specific EALs to be acceptable since NEI 07-01, Revision 0 was reviewed by NRC staff and found acceptable for use, as documented in a letter to NEI dated August 12, 2009. NEI 07-01 includes the critical elements specified in 10 CFR 50.47(b)(4), and Sections IV.B and IV.C of Appendix E to 10 CFR Part 50. The staff recognizes that the response to supplemental RAI 13.3-30 alters the text in Section D.2 and deletes the reference to NEI 07-01, Revision 0, as the technical basis for development of EALs. However the staff's determination of acceptability remains valid since the revised Section D.2 introduces an EPIP, "Emergency Classification," that will include the fully developed set of EALs, and a license condition proposed by the applicant that refers to the site-specific EALs as being developed in accordance with NEI 07-01, Revision 0, with no deviations. The staff has confirmed that Revision 1 to Part 10 of the COL application incorporates this license condition as described in this section of the SER.

The staff requested additional information from the applicant in supplemental RAI 13.3-30 because the applicant's initial and supplemental response to RAI 13.3-1 did not fully address all of the critical elements outlined in Option 2 (e.g., licensee actions for each emergency classification were not provided consistent with NUREG 0654/FEMA-REP-1, Appendix 1; Appendix 4 of the emergency plan includes an incomplete EAL scheme). The staff finds the applicant's response to supplemental RAI 13.3-30 to be acceptable because it addresses the critical elements outlined in Option 2 to RAI 13.3-1 and conforms to the guidance in NEI 07-01, Revision 0, and NUREG-0654/FEMA-REP-1, Appendix 1. However, the revision to Appendix 4 removed an Appendix E to 10 CFR Part 50 requirement (E.IV.B) to review the LNP's EALs with State and local authorities on an annual basis. The applicant revised the emergency plan to add this requirement in a supplemental response to RAI 13.3-30, which the staff finds acceptable because it meets the requirements in Appendix E to 10 CFR Part 50.

The staff confirmed that the applicant's proposed revisions to the LNP Emergency Plan provided in response to RAI 13.3-30 and its supplement were incorporated into Revision 2 of the LNP Emergency Plan and Part 10 of the COL application. In its further review of License Condition 11(A) and (C) in Part 10 of the COL application, Revision 2, the staff finds the proposed language to be ambiguous with regards to the State and local review and acceptance of LNP's initial EALs as required by Section IV.B of Appendix E to 10 CFR Part 50. Therefore, the staff revised License Condition 11(A) as follows to address this requirement:

A. Progress Energy Florida shall submit a fully developed set of site-specific EALs for LNP Units 1 [Unit 2] to the NRC in accordance with NEI 07-01, Revision 0, with no deviations. These 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 at least 180 days prior to initial fuel load.

The staff removed the underlined language described above from License Condition 11(C) and added it to License Condition 11(A). The staff finds that the proposed EAL scheme and license condition as modified by the staff to be acceptable because they meet the requirements of Appendix E to 10 CFR Part 50 and conform to the guidance in NUREG-0654/FEMA-REP-1. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

By letter dated November 8, 2012 from PEF to the NRC, the applicant provided its response to address the Final Rule on Enhancements to Emergency Preparedness Regulation effective November 23, 2011, regarding emergency declaration timeliness. The staff finds the applicant's description of emergency declaration timeliness in Section D.3 of the LNP Emergency Plan and reference to additional detail contained in an EPIP, to be acceptable since it conforms to the guidance in NSIR/DPR-ISG-01 and meets the requirements of Appendix E.IV.C.2 to 10 CFR Part 50.

13.3C.4.3 *Emergency Action Level Review by State and Local Authorities*

Technical Information in the Emergency Plan: {Appendix E, Section IV.B}
As previously described in Section 13.3C.4.2 of this SER, License Condition 11(A) includes provisions to ensure that the finalized EALs for LNP have been discussed and agreed upon with State and local officials. In addition, Section P.4, "Plan Review and Updates," of the LNP Emergency Plan includes the annual requirement for the licensee to review its EALs with the State and local governments.

Technical Evaluation: {Appendix E, Section IV.B}
The staff finds the proposed License Condition 11(A), as modified by the staff, to be acceptable because it meets the requirements of Appendix E to 10 CFR Part 50. The staff finds that the LNP Emergency Plan provides for the annual review of EALs by State and local officials. This is acceptable because it meets the requirements of Appendix E, Section IV.B to 10 CFR Part 50.

13.3C.4.4 *Conclusions*

On the basis of its review of the LNP Emergency Plan as described above for the emergency classification system, the staff concludes that the information provided in the LNP Emergency Plan is acceptable and meets the requirements of 10 CFR 50.47(b)(4) because it conforms with the guidance in Evaluation Criterion D of NUREG-0654/FEMA-REP-1 and NSIR/DPR-ISG-01 and meets the applicable requirements of Appendix E to 10 CFR Part 50 as described above.

13.3C.5 Notification Methods and Procedures

13.3C.5.1 *Regulatory Basis*

In determining whether the proposed emergency plan met the applicable regulatory requirements in 10 CFR 50.47(b)(5) for notification methods and procedures, the staff evaluated it against the detailed evaluation criteria in NUREG-0654/FEMA-REP-1. The staff also evaluated the proposed emergency plan against applicable regulatory requirements related to the area of "Notification Methods and Procedures," in Appendix E to 10 CFR Part 50 and 10 CFR 50.72, "Immediate notification requirements for operating nuclear power reactors."[6]

13.3C.5.2 *Notification Procedures, Capabilities, and Agreements*

Technical Information in the Emergency Plan: [E.1] {Appendix E, Sections IV.D.1 and D.3}

Section E, "Notification Methods and Procedures," of the LNP Emergency Plan states that mutually agreeable methods and procedures for notification of offsite response organizations are consistent with the emergency classification and action level scheme and have been established between DEF and State and local agencies. Appendix 5 of the LNP Emergency Plan includes an EPIP titled, "Notification and Communication," that provides details regarding notification responsibilities, communication systems, and information required to be transmitted to offsite agencies, including provisions for message verification. The means used to notify local, State, and Federal officials and agencies is described in Section E.1, "Notification and Mobilization of Emergency Response Personnel," and Section F, "Emergency Communications," of the LNP Emergency Plan. Points of contact for participating agencies and organizations are outlined in Table A-1. Appendix 7, "Public Alert and Notification System," Section 2.0, "Design Objective/Basis," states design parameters of the Alert and Notification System (ANS) are intended to meet or exceed the applicable criteria in Appendix 3 of NUREG-0654/FEMA-REP-1. Section E.5, "Instruction to the Public in the Plume Exposure Pathway EPZ," states notification of the public is the responsibility of State and local Emergency Management authorities. The applicant proposed EP ITAAC 4.1 and 12.1.1.B.2 to test the capabilities of the system used to notify the State of Florida and counties of Levy, Citrus, and Marion within 15 minutes after an emergency is declared. In addition, the applicant proposed EP ITAAC 9.3 and 12.1.1.B.4 to test the capability of the Public Alert and Notification System to successfully initiate a broadcast message to notify and protect all segments of the transient and resident populations.

Technical Evaluation: [E.1] {Appendix E, Section IV.D.1 and D.3}

The staff finds that the LNP Emergency Plan adequately refers to procedures which describe the mutually agreeable bases for notification of response organizations and conforms to the emergency classification scheme consistent with Appendix 1 to NUREG-0654/FEMA-REP-1, and NEI 07-01, Revision 0. These procedures will include the means for verification of

---

[6] Parentheses identify other applicable regulatory requirements

messages. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and meets the requirements in Appendix E to 10 CFR Part 50. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.5.3 *Notification and Activation of the Emergency Response Organization*

Technical Information in the Emergency Plan: **[E.2] {Appendix E, Section IV.C}**
Section E.1.1, "Progress Energy Emergency Response Organization," of the LNP Emergency Plan states that notification and mobilization of onsite and offsite personnel will be directed by the EC once an event has been classified. The public address system will be used as the primary means for notification of personnel within the PA. Audible and visual alarms specific to the nature of the emergency, will be used to alert site staff. ERO members are requested to respond, as directed by the EC. Offsite ERO staff will be contacted via a dedicated notification system. Commercial telephone and/or telephone-activated pager will be used as a backup means to notify ERO members who are offsite. Telephone numbers will be available in the Emergency Telephone Directory. Corporate personnel will be notified in accordance with implementing procedures. The applicant proposed EP ITAAC 4.2 to test the capability of the primary and back-up ERO notification systems. In addition, the applicant proposed EP ITAAC 12.1.1.B.1 and 12.1.1.B.2 to demonstrate the ability to alert, notify, and mobilize site emergency response personnel, and notify the NRC, and State and local governments in accordance with implementing procedures.

Technical Evaluation: [E.2] {Appendix E, Section IV.C}
The staff finds that the LNP Emergency Plan adequately addresses procedures for alerting, notifying, and mobilizing emergency response personnel. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and meets the requirements in Appendix E to 10 CFR Part 50. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.5.4 *Initial Message Content to Offsite Response Organizations*

Technical Information in the Emergency Plan: [E.3] {Appendix E, Section IV.A.4 and IV.C}
Section E.2, "Message Content," of the LNP Emergency Plan states that the content of the messages to Offsite-Response Organizations (OROs) have been established in conjunction with the State and local governments. The messages include the initial emergency classification (or classification escalation), whether a release is taking place, basic meteorological data, potentially affected population/areas, and any recommended protective actions. Supplemental messages containing more detail may be released once additional information is available. The applicant proposed EP ITAAC 12.1.1.B.2.a to test the capabilities of the LNP site to transmit information to State and local agencies within 15 minutes of event classification consistent with implementing procedures.

Technical Evaluation: [E.3] {Appendix E, Section IV.A.4 and IV.C}
The staff finds that the LNP Emergency Plan, in conjunction with State and local government authorities, adequately established the contents of the initial emergency messages to be sent

from the plant. This is acceptable because it meets the requirements of Appendix E, to 10 CFR Part 50 and conforms to the guidance in NUREG-0654/FEMA-REP-1. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.5.5 *Follow-up Messages to Offsite Response Organizations*

Technical Information in the Emergency Plan: [E.4]
Section E.3, "Follow-up Messages to Off-Site Authorities," of the LNP Emergency Plan states follow-up messages will be issued to the affected State and local authorities to describe the emergency. Follow-up messages will include the following information: incident location, name of caller and contact information; date and time; emergency classification; information regarding a potential or actual release; estimates of quantities and concentrations of radioactive particulate; meteorological conditions; projected doses at prescribed locations; emergency response actions underway; protective action recommendations; requests for any onsite support needed by offsite organizations; and any prognosis for changes in event classification or other conditions based on the current plant assessment.

Technical Evaluation: [E.4]
The staff finds that the LNP Emergency Plan adequately provides for follow-up messages from the facility to offsite authorities, and the content of these messages is consistent with the guidance in NUREG-065/FEMA-REP-1, Revision 1. In addition, the staff verified that the nature of the information provided is consistent with the requirements of the State and local emergency plans. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.5.6 *Notification of the Public*

Technical Information in the Emergency Plan: [E.6] {Appendix E, Section IV.D.3}
Section E.5 of the LNP Emergency Plan states that the primary means for alerting the public in the 10-mile plume exposure pathway EPZ to initiate protective actions is by sounding the ANS. In supplemental RAI 13.3-38, the staff requested the applicant provide additional information related to a secondary capability to promptly alert and notify the public of an emergency should the primary system (ANS) become unavailable. In its response, the applicant stated that Sections 2.1 and 2.2 of Appendix 7, "Public Alert and Notification System," to the LNP Emergency Plan identifies mobile sirens as the alternate method of notifying the public when offsite locations 5 miles from the site are not suitable for fixed siren placement. Section J.10.c, "Protective Measure Implementation," describes warnings to the public as being the responsibility of State and local officials. Section E.5 of the LNP Emergency Plan states, in part, that in the event of an emergency, the public will be advised to tune to local televisions or radio stations for instructions. General information regarding the nature of potential emergencies will be disseminated through news or press releases from the ENC. The Public Information Director is responsible for the coordination and dissemination of this information. This process is discussed in Section G, "Public Education and Information," of the LNP Emergency Plan. Appendix 7 of the LNP Emergency Plan provides detailed information regarding the design objectives of the public ANS, including the ability to alert the population within the plume exposure pathway EPZ within 15 minutes. Appendix 5 provides an EPIP titled, "Notification and

Communication," which implements this section of the LNP Emergency Plan. The applicant proposed EP ITAAC 9.3 to test the capability of the Public ANS to successfully initiate a broadcast message to notify and protect all segments of the transient and resident populations.

By letter dated November 8, 2012, from PEF to the NRC, the applicant provided its response to address the Final Rule on Enhancements to Emergency Preparedness Regulation effective November 23, 2011, regarding its backup ANS capability should the primary means for performing this function become unavailable. The staff requested additional information from the applicant in RAI 13.3-60 to clarify in the LNP Emergency Plan the administrative means used by DEF for alerting the populace within the 10-mile EPZ, and to clearly identify whether tone alert radios were the designated backup means for performing this activity. In addition, the staff requested the applicant describe the primary means for notifying the populace within the 10-mile EPZ should mobile sirens become cost-prohibitive, and clarify in the LNP Emergency Plan whether the 15-minute timeliness goal includes both the capability to alert and notify the public of an emergency at the LNP site. In its initial and supplemental response to address this RAI, the applicant committed to revise Section E.5 of the LNP Emergency Plan to reference Appendix 7 which describes the ANS; clarify that route alerting is the backup means for alerting the public while deleting its reference to tone alert radios; add a reference to Appendix 5 in Section E to clarify that the "Notification and Communication" implementing procedures apply to Section E of the emergency plan; and clarify sections E.5, J.10.c, and Appendix 7 to remove mobile sirens from the emergency plan as a primary means of alerting the public.

Technical Evaluation: [E.6] {Appendix E, Section IV.D.3}
The staff finds the additional information and proposed textual revisions to the emergency plan provided in response to supplemental RAI 13.3-38 and RAI 13.3-60, in consideration of its supplemental response, acceptable because they provide clarification that an alternate means of alerting the public exists for an emergency at LNP and conforms to the guidance in NUREG-0654/FEMA-REP-1 and NSIR/DPR-ISG-01 and meets the requirements of Appendix E to 10 CFR Part 50. The staff confirmed that the information provided in response to supplemental RAI 13.3-38 has been incorporated into Revision 2 of the LNP Emergency Plan. The staff confirmed that the information provided in response to RAI 13.3-60, including its supplemental response, has been incorporated into Revision 6 of the LNP Emergency Plan. Therefore, the staff finds that the LNP Emergency Plan adequately establishes the administrative and physical means and the time required for notifying and providing prompt instructions to the public in the plume exposure pathway EPZ. This is acceptable because it conforms to the guidance of NUREG-0654/FEMA-REP-1 and NSIR/DPR-ISG-01 and meets the requirements of Appendix E to 10 CFR Part 50. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.5.7 *Written Messages to the Public*

Technical Information in the Emergency Plan: [E.7]
Section E.6, "Written Messages to the Public," of the LNP Emergency Plan states that written, pre-planned messages or Emergency Alert System (EAS) messages are released to the media by the State or local Director of Emergency Management consistent with the emergency

classification scheme. These messages provide instructions on specific actions to be taken by the public, including information on the nature of the emergency and recommended protective actions (e.g., sheltering, evacuation, potassium iodide).

Technical Evaluation: [E.7]
The staff finds the LNP Emergency Plan adequately discusses written messages intended for the public developed by the State of Florida. In particular, draft messages to the public giving instructions with regard to specific protective actions to be taken by occupants of affected areas, were prepared. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.5.8 *Notification of the NRC*

Technical Information in the Emergency Plan: {Appendix E, Section IV.A.4} (10 CFR 50.72(a)(3)) and **(10 CFR 50.72(c)(3))**
Section E.1.2.b, "Nuclear Regulatory Commission (NRC)," of the LNP Emergency Plan states the NRC will be notified using the Emergency Notification System (ENS) as soon as possible and within one hour of emergency classification. Commercial telephone lines are available as a backup method for notification. Section F.1.c, "Description of Communication Links," states separate telephone lines are dedicated for communications with the NRC. Section F.1.c.1 states that the ENS will be used to provide initial notifications to the NRC, as well as ongoing information about plant systems, status, and parameters. The EC and EOF Director, when the EOF is operational are responsible for direct interface with offsite authorities. Additional information regarding the timely notification to the NRC during a security-based event can be found in Section 13.3C.17.3 of this SER. In RAI 13.3-46, the staff requested the applicant clarify in the LNP Emergency Plan whether an open, continuous channel for communication with the NRC will exist, if requested. In response, the applicant stated that the LNP will maintain an open, continuous communication channel with the Nuclear Regulatory Commission Operations Center (NRCOC) upon request by the NRC per 10 CFR 50.72(c)(3) over the ENS and/or Health Physics Network (HPN) circuits. The EC has accountability to ensure the channel remains open upon request. The applicant proposed to revise the LNP Emergency Plan to reflect this information.

Technical Evaluation: {Appendix E, Section IV.A.4} (10 CFR 50.72(a)(3))
The staff finds that the LNP Emergency Plan provides for prompt notification (as soon as possible, within one hour) of the NRC after declaration of an emergency. This is acceptable because it meets the requirements in 10 CFR 50.72(a)(3) and applicable portions of Appendix E to 10 CFR Part 50.

(10 CFR 50.72(c)(3))
The staff finds the applicant's response to RAI 13.3-46 to be acceptable because it describes the means by which the licensee will maintain an open line with the NRC upon request and meets the requirements of 10 CFR 50.72(c)(3). The staff created Confirmation Item 13.3-46 to track the applicant's proposed changes to the LNP Emergency Plan in response to RAI 13.3-46.

Resolution of Confirmatory Item 13.3-46

Confirmatory Item 13.3-46 is an applicant commitment to update the LNP Emergency Plan. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). As a result, Confirmatory Item 13.3-46 is now closed.

On the basis of its review of the LNP Emergency Plan, the staff finds that the LNP Emergency Plan includes provisions for the licensee to maintain an open, continuous communication channel with the NRCOC upon request by the NRC. This is acceptable because it meets the requirements of 10 CFR 50.72(c)(3).

13.3C.5.9 *Conclusion*

On the basis of its review of the LNP Emergency Plan, the staff concludes that the information provided in the LNP Emergency Plan regarding notification methods and procedures is acceptable and meets the requirements of 10 CFR 50.47(b)(5) because it conforms to the guidance in Evaluation Criterion E of NUREG-0654/FEMA-REP-1 and NSIR/DPR-ISG-01 and meets the applicable requirements of Appendix E to 10 CFR Part 50 and 10 CFR 50.72(a)(3) and (c)(3) as described above.

13.3C.6 Emergency Communications

13.3C.6.1 *Regulatory Basis*

In determining whether the proposed emergency plan met the applicable regulatory requirements in 10 CFR 50.47(b)(6) for emergency communications, the staff evaluated it against the detailed evaluation criteria in NUREG-0654/FEMA-REP-1. The staff also evaluated the proposed emergency plan against applicable regulatory requirements related to the area of "Emergency Communications," in Appendix E to 10 CFR Part 50 and GL 91-14, “Emergency Telecommunications.”

13.3C.6.2 *Content of the Emergency Communications Plan*

Technical Information in the Plan: [F.1.a]
Section F, “Emergency Communications,” of the LNP Emergency Plan describes the communication systems and provisions for communications between the LNP site ERFs and principal response organizations, including State, local, and Federal agencies. Section F further states that details describing the operation and testing of communication systems is located in EPIPs. Section F.1.a states that Progress Energy maintains capabilities for 24-hour notification to the State and county emergency response network and all State/county warning points are manned 24-hours per day. Appendix 5 identifies an EPIP, “Notification and Communication,” that supports and implements Section F of the LNP Emergency Plan.

Technical Evaluation: [F.1.a]
The staff finds that the LNP Emergency Plan adequately addresses communication plans for emergencies, provides for 24-hour per day notification to, and activation of, the State/local emergency response network; and at a minimum, a telephone link and alternate, including 24-hour per day manning of communications links that initiate emergency response actions. This is acceptable because they conform to the guidance described in NUREG-0654/FEMA-REP-1. Additional information regarding emergency communications and the staff's evaluation is located in SER Section 9.5.2, "Communications Systems."

Technical Information in the Plan: [F.1.b]
Sections F.1.b and F.1.d of the LNP Emergency Plan identify various communication links (e.g., State of Florida Hot Ringdown Telephone System, Florida ESATCOM, private telephone, satellite telephone, and dedicated radio networks) available from the CRs, TSCs, and EOF used to provide a primary and alternate means of communicating with State and local governments within the EPZs.

Technical Evaluation: [F.1.b]
The staff finds that the LNP Emergency Plan adequately addresses provisions for continuous communications with State and local governments within the plume exposure pathway EPZ. This is acceptable because it meets the guidance in NUREG-0654/FEMA–REP-1, Revision 1.

Technical Information in the Plan: [F.1.c]
Section F.1.c of the LNP Emergency Plan lists separate telephone lines dedicated for communications with the NRC including the ENS, HPN, Reactor Safety Counterpart Link (RSCL), Protective Measures Counterpart Link (PMCL), Emergency Response Data System (ERDS), Management Counterpart Link (MCL), and NRC Remote Access link.

Technical Evaluation: [F.1.c]
The staff finds that the LNP Emergency Plan adequately addresses provisions for communications, as needed, with Federal EROs. This is acceptable because it conforms to the guidance described in NUREG-0654/FEMA-REP-1.

Technical Information in the Plan: [F.1.d]
Section F.1.d of the LNP Emergency Plan describes the communication links to be used for communications between the LNP ERFs (e.g., CRs, TSCs, and EOF), State and county EOCs, and the Florida DHBRC, via the State of Florida Hot Ringdown Telephone System and conference-line phone systems. In RAI 13.3-47(1), the staff requested that the applicant clarify in the LNP Emergency Plan the provisions for communications from the OSC and EOF to the PEF radiological monitoring teams, which are dispatched, as needed, prior to the arrival of the State of Florida DHBRC support. In response, the applicant stated, in part, that a separate radio communications channel exists for communications from the EOF, TSC, and CR to the PEF radiological monitoring teams that are dispatched for offsite monitoring, as needed, prior to the arrival of the State of Florida DHBRC support. Commercial cell phones, satellite phones, or other means are available as backup to the primary field team communications system. The applicant proposed EP ITAAC 5.1 to demonstrate the capability of both the primary and

secondary communications systems/methods between the LNP ERFs, radiological field monitoring teams, and State/county warning points and EOCs. Table A-1 provides the point of contact, by title, for primary organizations in the ERO. Tables F-1, "On-Site Communications," and F-2, "Interfacility/Organization Communications," identify communication systems and the title of the primary communicators within each ERF and its respective organization.

Technical Evaluation: [F.1.d]
The staff finds the applicant's response to RAI 13.3-47(1) to be acceptable because it clarifies the provisions for communications between the ERFs and the licensee's radiological monitoring teams, and conforms to the guidance in NUREG-0654/FEMA-REP-1. The staff created Confirmatory Item 13.3-47(1) to track the applicant's proposed changes to the LNP Emergency Plan provided in response to RAI 13.3-47(1).

<u>Resolution of Confirmatory Item 13.3-47(1)</u>

Confirmatory Item 13.3-47(1) is an applicant commitment to update the LNP Emergency Plan. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). As a result, Confirmatory Item 13.3-47(1) is now closed.

On the basis of its review of the LNP Emergency Plan, the staff finds that the LNP Emergency Plan adequately describes the communication plans that included provisions for emergency communications between the nuclear facility and the EOF, State and local EOCs, and radiological monitoring teams. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

Technical Information in the Plan: [F.1.e]
Section 13.3C.5.3 of this SER provides information regarding the primary and backup means of notification and activation of the onsite and offsite ERO.

Technical Evaluation: [F.1.e]
The staff's evaluation of the information provided by the applicant regarding the provision for alerting or activating emergency personnel in each response organization can be found in Section 13.3C.5.3 of this SER.

Technical Information in the Plan: [F.1.f]
Section F.1.f of the LNP Emergency Plan states that communications between the LNP CRs, TSCs, and EOF, to the NRCOC is via the Emergency Telephone System (ETS) or private phone. Communications from these facilities to the NRC Regional Office is via private telephone. Section I.7, "Field Monitoring Capability," identifies the Radiological Emergency Team assembly area as the EOF. The applicant proposed EP ITAAC 5.2 to verify that a test will be performed to demonstrate communications between LNP ERFs and the NRC offices (regional and headquarters). The test will include the HPN and ERDS.

Technical Evaluation: [F.1.f]

The staff finds that the LNP Emergency Plan adequately describes the communication plans for emergencies and addresses provisions for communication by the licensee with NRC Headquarters and NRC Regional Office EOCs and the EOF and radiological monitoring team assembly area. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

Technical Information in the Plan: {Appendix E, Section IV.E.9}
Section F of the LNP Emergency Plan describes onsite and offsite communication systems. Section F.3, "Communication System Reliability," of the LNP Emergency Plan states that the variety of onsite communication networks ensures the availability and reliability of communications. Failure of normal power supplies will not prevent offsite communication capability since backup power is provided. Communications systems equipment is located in different areas to prevent incapacitation of all communication systems during an accident. Dedicated telephone lines are checked according to specified schedules. Additional information regarding individuals by title, and alternates for those in charge at both ends of the communication links can be found in Sections 13.3C.5 and 13.3C.6.2 [F.1.d] of this SER.

{Appendix E, Section IV.E.9(a)}
Section F, "Emergency Communications," of the LNP Emergency Plan describes the communication links used to notify and activate State/local agencies. Section F.3, "Communication System Reliability," states that monthly tests are conducted between the LNP to State and local warning points, and the State EOCs within the plume exposure pathway EPZ.

{Appendix E, Section IV.E.9(b)}
Section N.2.a, "Communication Drills," of the LNP Emergency Plan states that PEF tests communications with Federal EROs and States within the ingestion pathway EPZ monthly. Testing includes that of the ETS and the ERDS.

{Appendix E, Section IV.E.9(c)}
Section F.1.d, including related EP ITAAC, as discussed in Section 13.3C.6.2 in this SER and in LNP Emergency Plan describes the provisions for communication between the LNP ERFs (e.g., CRs, TSCs, and EOF) and State and county EOCs, and the State of Florida DHBRC. In RAI 13.3-47(1), the staff requested additional information regarding provisions for communication with the LNP radiological monitoring team as addressed above in this section of the SER. Section F.1.d of the LNP Emergency Plan further describes three separate conference line phone systems that have been established to facilitate communications between the CRs, TSCs, and the EOF, including the establishment of a quarterly test frequency. Section F.3 of the LNP Emergency Plan states that communication tests between the LNP site and State and county warning points, and the State EOCs within the plume exposure pathway EPZ are performed monthly. This communication test includes an aspect of understanding the content of messages. In addition, Section F.3 states communication tests between the LNP site and State and local EOCs, and the environmental monitoring teams are tested annually. In RAI 13.3-47(2), the staff requested the applicant provide clarification in the LNP Emergency Plan regarding the communications test frequency between the LNP site (e.g., CRs, TSCs, and

EOF), and the LNP radiological control teams that precede the offsite survey support provided by the State of Florida DHBRC (environmental monitoring team). The applicant stated, in part, that a future revision to the LNP Emergency Plan will state that the communication test frequency between the LNP EOF, TSC, and CR to the radiological monitoring team shall be annual.

{Appendix E, Section IV.E.9(d)}
Section F.3 of the LNP Emergency Plan states that quarterly communication tests are conducted between the LNP ERFs (e.g., CRs, TSCs, and EOF) to the NRC Headquarters Operations Center. In RAI 13.3-19(B), the staff requested a discussion on why the LNP ERFs communication test with NRC Headquarters is quarterly instead of monthly. In its response, the applicant committed to change the frequency of this communications test to monthly consistent with the regulations. In RAI 13.3-47(3), the staff requested that the applicant clarify in the LNP Emergency Plan the frequency of testing communications between the LNP CRs, TSCs, and EOF and the appropriate NRC Regional Office. In response, the applicant stated, in part, that a future revision to the LNP Emergency Plan will state the frequency for testing communications between the LNP CRs, TSCs, EOF, and appropriate NRC Regional Office will be on a monthly basis.

Technical Evaluation: {Appendix E, Section IV.E.9(a)-(d)}
The staff finds the additional information and proposed textual revision provided in response to RAI 13.3-19(B) to be acceptable because it meets the requirements of Appendix E to 10 CFR Part 50. However, the applicant did not identify the testing of communications between the LNP ERFs and the appropriate NRC Regional office. The staff requested this information in supplemental RAI 13.3-47(3). The staff confirmed the changes proposed to the LNP Emergency Plan in response to RAI 13.3-19(B) were incorporated in Revision 1 to the LNP Emergency Plan.

The staff finds the additional clarification and textual revisions provided in the applicant's responses to RAIs 13.3-47(2) and (3) to be acceptable because it clarifies in the LNP Emergency Plan the frequency of testing communications between the LNP ERFs, radiological control teams, and the appropriate NRC Regional Office, which meets the applicable requirements of Appendix E to 10 CFR Part 50. The staff created Confirmatory Items 13.3-47(2) and 13.3-47(3) to track the applicant's proposed changes to the LNP Emergency Plan provided in response to RAIs 13.3-47(2) and (3).

<u>Resolution of Confirmatory Items 13.3-47(2) and 13.3-47(3)</u>

Confirmatory Items 13.3-47(2) and 13.3-47(3) are applicant commitments to update the LNP Emergency Plan. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). As a result, Confirmatory Items 13.3-47(2) and 13.3-47(3) are now closed.

On the basis of its review of the LNP Emergency Plan, the staff finds that the LNP Emergency Plan adequately states that at least one onsite and one offsite communications systems exists, and that each system has a backup power source. This is acceptable because it meets the requirements described in Appendix E to 10 CFR Part 50.

In addition, 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. Consistent with the function of the governmental agency, these arrangements included:

a. Provisions for communications with contiguous State/local governments within the plume exposure pathway EPZ. Such communications shall be tested monthly.

b. Provisions for communications with Federal EROs. Such communications systems shall be tested annually. However, the LNP has committed to a monthly testing frequency.

c. Provisions for communications among the nuclear power reactor CR, the onsite TSC, and the EOF; and among the nuclear facility, the principal State and local EOCs, and the field assessment teams. Such communications systems shall be tested monthly.

d. Provisions for communications by the licensee with NRC Headquarters and the appropriate NRC Regional Office Operations Center from the nuclear power reactor CR, the onsite TSC, and the EOF. Such communications shall be tested monthly.

These provisions for onsite and offsite communications are acceptable because they meet the requirements in Appendix E to 10 CFR Part 50.

Technical Information in the Plan: (GL 91-14)
Section 13.3C.6.2 [F.1.c] of this SER and F.1.C of the LNP Emergency Plan describes communication pathways (e.g., ENS, HPN, RSCL, etc.) dedicated for communications with the NRC. In addition, Section 13.3C.6.2 [Appendix E.IV.E.9] of this SER and Section F.3 of the LNP Emergency Plan describe, in general, communication system reliability through the use of dedicated phone lines, normal and backup power supplies, and periodic testing. Additional information regarding the adequacy of emergency telecommunications systems is provided in Section 9.5.2 of this SER.

Technical Evaluation: (GL 91-14)
The staff finds that the LNP Emergency Plan adequately includes provisions for communications with the NRC. This is acceptable because it conforms to the guidance in GL 91-14.

13.3C.6.3 *Communications with Medical Facilities*

Technical Information in the Plan: [F.2.]
Section F.2, "Communication with Fixed and Mobile Medical Support Facilities," of the LNP Emergency Plan states that the LNP maintains communication systems which allow for communication between LNP and fixed and mobile medical support facilities. These systems

include both commercial telephone communications for fixed facilities and radio communications for ambulance contact.

Technical Evaluation: [F.2.]
The staff finds that the LNP Emergency Plan adequately ensures that a coordinated communication link exists for fixed medical support facilities and ambulance service(s). This is acceptable because it conforms to the guidance described in NUREG-0654/FEMA-REP-1.

13.3C.6.4 *Periodic Testing of the Emergency Communications System*

Technical Information in the Plan: [F.3]
Section F.3, "Communication System Reliability," of the LNP Emergency Plan provides periodic test frequencies for communications between the LNP ERFs, State and local warning points and EOCs, radiological monitoring teams, and the NRC. Appendix 7 of the LNP Emergency Plan provides a description of the design for the public ANS that includes periodic system tests (i.e., silent test, growl test, and complete cycle test) to be performed and their associated test frequencies (i.e., silent – every two weeks, growl – quarterly and after preventative maintenance). Additional information regarding communication test frequencies is in Section 13.3C.14.10, "Communication Drills," of this SER. Section F of the LNP Emergency Plan states that details regarding the operation and testing of communication systems is located in EPIPs. Appendix 5 identifies an EPIP titled, "Notification and Communication," that supports and implements Section F of the LNP Emergency Plan.

Technical Evaluation: [F.3]
The staff finds that the LNP Emergency Plan adequately describes the conduct of periodic testing of the entire emergency communications system. This is acceptable because it conforms to the guidance described in NUREG-0654/FEMA-REP-1.

13.3C.6.5 *Conclusion*

On the basis of its review of the LNP Emergency Plan, the staff concludes that the information provided in the LNP Emergency Plan regarding emergency communications is acceptable and meets the requirements of 10 CFR 50.47(b)(6) because it conforms with the guidance in Evaluation Criterion F of NUREG-0654/FEMA-REP-1 and GL 91-14 and meets the applicable requirements of Appendix E to 10 CFR Part 50.

13.3C.7 Public Education and Information

13.3C.7.1 *Regulatory Basis*

In determining whether the proposed emergency plan met the applicable regulatory requirements in 10 CFR 50.47(b)(7) for public education and information, the staff evaluated it against the detailed evaluation criteria in NUREG-0654/FEMA-REP-1. The staff also evaluated the proposed emergency plan against applicable regulatory requirements related to the area of "Public Education and Information," in Appendix E to 10 CFR Part 50.

13.3C.7.2 *Content of Public Information*

Technical Information in the Plan: [G.1]
Section G.1, "Public Information Program," of the LNP Emergency Plan describes the program designed to educate and inform the public of emergency notification methods and actions to take in the event of an emergency at LNP. DEF, in coordination with State and county officials, will provide this information to residents, businesses, and transients in the 10-mile plume exposure pathway EPZ at least annually. This information includes educational material on radiation; contacts for additional information; protective measures (e.g., evacuation routes, relocation centers, sheltering, and respiratory protection); and special needs of the handicapped. PEF states that the means for accomplishing dissemination of this information will be via a publication, in the form of brochures, calendars, and/or phone book pages that will be distributed to the residents of Citrus, Levy, and Marion Counties within a 10-mile radius of LNP, and that will be available to the general public within the same area. In RAI 13.3-20, the staff requested the applicant provide a discussion in the LNP Emergency Plan regarding its efforts to coordinate public education and information with the CR3 site, specifically in areas where the CR3 and LNP EPZs overlap. In its response, the applicant stated that the public education and information programs for the two sites will be coordinated by PEF. Development and distribution of public safety information materials to resident, business, and transient populations will be shared between the two sites. Due to the proximity of the sites and overlapping EPZs, PEF will develop and distribute one set of public information materials describing the 10-mile EPZs for both the LNP and CR3. The applicant also provided revised text for Section G.1 of the LNP Emergency Plan for clarification. In supplemental RAI 13.3-48, the staff requested that the applicant commit to develop and distribute the initial public information publications, in coordination with CR3, within 180 days prior to fuel load at LNP. In response, the applicant proposed a license condition to ensure that the initial LNP public information publications are distributed within 180 days prior to fuel load at LNP. Specifically, the applicant proposed License Condition 11(E):

E. PEF will distribute the initial LNP public information publications, developed in coordination with CR3 and consistent with the LNP Emergency Plan, to the public within 180 days prior to fuel load.

In the LNP Emergency Plan, reference to CR3 in the discussion of the public information program has been eliminated. Appendix 5 to the LNP Emergency Plan includes an administrative procedure titled, "Public Information."

Technical Evaluation: [G.1]
The staff finds the additional information and proposed textual revisions provided by the applicant in response to RAI 13.3-20 to be acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1. The staff confirmed that the changes referenced in response to RAI 13.3-20 were included in Revision 1 to the LNP Emergency Plan. The staff finds the applicant's response to RAI 13.3-48 to be acceptable because it provides License Condition 11(E) to ensure that the initial public information developed in coordination with CR3

is distributed prior to fuel load (plant operation). Specifically, the staff finds License Condition 11(E) as stated below acceptable:

E. PEF will distribute the initial LNP public information publications, developed in coordination with CR3 and consistent with the LNP Emergency Plan, to the public within 180 days prior to fuel load.

The staff created Confirmatory Item 13.3-48 to track the applicant's proposed changes to the LNP COL application provided in response to RAI 13.3-48.

<u>Resolution of Confirmatory Item 13.3-48</u>

Confirmatory Item 13.3-48 is an applicant commitment to update the LNP Emergency Plan. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). As a result, Confirmatory Item 13.3-48 is now closed.

By letter dated April 18, 2013, from PEF to NRC regarding impacts from retirement of the CR3 nuclear plant, the applicant removed the language proposed in response to RAI 13.3-20, which included revisions to the LNP Emergency Plan for the coordination of the public education and information programs between CR3 and LNP, and the commitment to publish a consolidated set of public safety information for all CR3 and Levy EPZ stakeholders. By letter dated January 10, 2014, "LNP Emergency Planning Impacts from Retirement of CR3 Supplement," from DEF to NRC, the applicant proposed revised language in License Condition 11(E) to incorporate the commitment for a coordinated dissemination of public safety information with CR3 until CR3 is no longer required to provide this information. This revision is as follows:

E. DEF shall distribute the initial LNP public information publications, consistent with the LNP Emergency Plan, within 180 days prior to fuel load at LNP. DEF must coordinate the development, initial and annual redistribution, and maintenance of this information with CR3 as long as the NRC requires CR3 to distribute public information publications.

As discussed in Section 13.3.4, CR3 was granted exemptions from specific EP standards. The staff revised License Condition 11(E) since the NRC exempted CR3 from the requirement to annually disseminate general information.

On the basis of its review of the LNP Emergency Plan, the staff finds that the LNP Emergency Plan adequately provides for a coordinated periodic (at least annually) dissemination of information to the public regarding how they will be notified and what their actions should be in an emergency. In addition, the means for accomplishing this dissemination are also adequately described. This is acceptable because it conforms to the guidance described in NUREG-0654/FEMA-REP-1.

13.3C.7.3 *Dissemination and Maintenance of Public Information*

Technical Information in the Plan: [G.2] {Appendix E, Section IV.D.2}
A general discussion regarding the LNP Public Education and Information Program is provided in Section 13.3C.7.2 of this SER. In addition, Section G.2, "Distribution and Maintenance of Public Information," of the LNP Emergency Plan states that Progress Energy will support, but not necessarily be limited to, publications (referenced above) to be provided in quantity at key locations, such as motels and various business locations, in order to reach any new or transient individuals in the area. These publications will provide the appropriate information that will be helpful if an emergency or accident occurs. This information will refer new or transient individuals to the telephone directory or other source of local emergency information, and provide guidance as to the appropriate radio and television frequencies in which information can be obtained. Section 13.3C.7.2 of this SER describes a license condition proposed by the applicant in response to supplemental RAI 13.3-48, to ensure that the initial LNP public information publications are distributed within 180 days prior to fuel load at LNP.

Technical Evaluation: [G.2] {Appendix E, Section IV.D.2}
The staff's evaluation of the applicant's response to supplemental RAI 13.3-48 is provided in Section 13.3C.7.2 of this SER. The staff created Confirmatory Item 13.3-48 to track the applicant's proposed changes to the LNP COL application provided in response to this RAI.

<u>Resolution of Confirmatory Item 13.3-48</u>

Confirmatory Item 13.3-48 is an applicant commitment to update the LNP Emergency Plan. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). As a result, Confirmatory Item 13.3-48 is now closed.

By letter dated April 18, 2013, from PEF to NRC regarding impacts from retirement of the CR3 nuclear plant, the applicant removed the language proposed in response to RAI 13.3-20, which included revisions to the LNP Emergency Plan for the coordination of the public education and information programs between CR3 and LNP, and the commitment to publish a consolidated set of public safety information for all CR3 and Levy EPZ stakeholders. By letter dated January 10, 2014, "LNP Emergency Planning Impacts from Retirement of CR3 Supplement," from DEF to NRC, the applicant proposed revised language in License Condition 11(E) to incorporate the commitment for a coordinated dissemination of public safety information with CR3 until CR3 is no longer required to provide this information. This revision is as follows:

E. DEF shall distribute the initial LNP public information publications, consistent with the LNP Emergency Plan, within 180 days prior to fuel load at LNP. DEF must coordinate the development, initial and annual redistribution, and maintenance of this information with CR3 as long as the NRC requires CR3 to distribute public information publications.

With the staff's proposed revision to License Condition 11(E), the staff finds License Condition 11(E) acceptable.

On the basis of its review of the LNP Emergency Plan, the staff finds that the LNP Emergency Plan adequately describes a public information program that provides the permanent and transient population within the plume exposure EPZ an opportunity to become aware of the information annually. The program includes provisions for written material that is likely to be available in a residence during an emergency. This is acceptable because it conforms to the guidance described in NUREG-0654/FEMA-REP-1 and meets the requirements in Appendix E to 10 CFR Part 50.

13.3C.7.4 *Points of Contact for the News Media*

Technical Information in the Plan: [G.3.a]
Section G.3, "News Media Coordination," and G.4, "Information Exchange," of the LNP Emergency Plan states that the ENC will be the principal point of contact with the news media during an emergency. The ENC is identified as being co-located with the LNP EOF. Section G.4 states that a news coordinator in the ENC will have access to all required information and provide plant status and company information during scheduled news conferences and media briefings. Section B.5.2.g, "Emergency News Center (ENC)," states that the ENC staff is responsible for the dissemination of information to the public and news media under the direction of the Public Information Director (PID). Section H.2.2, "Emergency News Center," states, in part, that the PID is responsible for dissemination of information by Progress Energy.

Technical Evaluation: [G.3.a]
By letter dated April 18, 2013, from PEF to NRC regarding impacts from retirement of the CR3 nuclear plant, the applicant renamed the EOF to remove any reference to the Crystal River Training Center. The EOF is now referred to in the LNP Emergency Plan as the LNP EOF. Therefore, the ENC is co-located with the LNP EOF. The staff confirmed these changes were made in the LNP Emergency Plan.

The staff finds that the LNP Emergency Plan adequately designates the points of contact and physical locations for use by news media during an emergency and that the LNP Emergency Plan also describes space, which may be used for a limited number of the news media at the EOF. This is acceptable because it conforms to the guidance provided in NUREG-0654/FEMA-REP-1.

13.3C.7.5 *Space for News Media*

Technical Information in the Plan: [G.3.b]
Section H.2.2, "Emergency News Center," states, in part, that the ENC provides a near-site location for the local dissemination of information to the public and news media.

Technical Evaluation: [G.3.b]
Initially, the applicant proposed to use the existing CR3 EOF for support of emergency planning for LNP Units 1 and 2. Since the ENC is co-located with the EOF, the staff's review of the EOF and ENC focused on the extension of the existing facility as it applies to the proposed reactor units at the LNP site. Currently, the EOF and ENC is an existing NRC-approved facility for CR3 that meets the requirements of 10 CFR 50.47 and Appendix E to Part 50, and conforms to the guidance in NUREG-0696 and NUREG-0737, Supplement No. 1. The staff determined the EOF and ENC was acceptable for use at LNP Units 1 and 2 because: 1) the NRC performs oversight of emergency preparedness, including the EOF and ENC, by monitoring performance indicators; 2) the EOF and ENC is inspected periodically during routine inspections, drills and exercises; and 3) any changes to the EOF and ENC are reviewed in accordance with the established inspection program and requirements for operating reactors.

However, by letter dated April 18, 2013, from PEF to NRC, the applicant proposed a revision to the LNP Emergency Plan to address the future state of CR3 as it relates to decommissioning activities and the anticipated relaxation of offsite EP responsibilities for CR3. In consideration of these circumstances, the applicant anticipates the EOF and ENC will no longer be required for response to an emergency event at CR3. In LNP Emergency Plan, the EOF has been renamed the LNP EOF and is expected to support the future needs of LNP only. The staff anticipates a lapse in time for which the readiness capabilities of the EOF and ENC will no longer be required. By letter dated January 10, 2014, from DEF to the NRC, the applicant proposed EP ITAAC 7.2.3 through 7.2.5 to address regulatory guidance criteria in NUREG-0696 and Supplement 1 to NUREG-0737 that are not addressed in the LNP Emergency Plan. Prior to fuel load, these EP ITAAC will provide staff assurance that the EOF continues to comply with the uniform building code; the EOF is environmentally controlled to provide room air temperature, humidity, and cleanliness appropriate for personnel and equipment; and the EOF is provided with industrial security when it is activated to exclude unauthorized personnel and when it is idle to maintain its readiness. Given that the EOF and ENC may not be required to maintain its functionality for some time prior to LNP operations, the staff found these ITAAC necessary to ensure that the EOF and ENC is constructed as designed, as required by 10 CFR 52.80. Therefore, the staff finds the applicant's proposed EP ITAAC 7.2.3 through 7.2.5 acceptable since they conform to the guidance in NUREG-0696 and Supplement 1 to NUREG-0737 and meet the requirements in 10 CFR 52.80. The staff subsequently finds the LNP EOF and ENC are acceptable.

The applicant is proposing to use the existing CR3 ENC for LNP Units 1 and 2. The staff's review focused on the extension of the existing facility as it applies to the proposed reactor units at the LNP site. The ENC is an existing NRC approved facility for CR3 that conforms to the guidance in NUREG-0654/FEMA-REP-1 as it pertains to limited space for news media at the near-site EOF. Therefore, the staff finds the ENC acceptable for use at LNP Units 1 and 2 because: 1) the NRC performs oversight of emergency preparedness, including the ENC, by monitoring performance indicators; 2) the ENC is inspected periodically during routine inspections and drills and exercises; and 3) any changes to the ENC are reviewed in accordance with the established inspection program and requirements for operating reactors.

13.3C.7.6 *Designated Spokesperson*

Technical Information in the Plan: [G.4.a]
Section G.4 of the LNP Emergency Plan states that a News Coordinator in the ENC will have access to all required information and provide plant status and company information during news conferences and media briefings.

Technical Evaluation: [G.4.a]
The staff finds that the LNP Emergency Plan adequately identifies a spokesperson that has access to all necessary information. This is acceptable because it conforms to the guidance provided in NUREG-0654/FEMA-REP-1.

13.3C.7.7 *Timely Exchange of Information*

Technical Information in the Plan: [G.4.b]
Section G.4 of the LNP Emergency Plan states that LNP personnel who are designated in implementing procedures will meet periodically and/or have timely exchanges of information. These information exchanges will extend to include other designated spokespersons of local, State, and Federal agencies, and will include the awareness of media releases. Appendix 5 to the LNP Emergency Plan includes an Administrative Procedure titled, “Public Information” that implements this commitment.

Technical Evaluation: [G.4.b]
The staff finds the LNP Emergency Plan adequately describes established arrangements for timely exchange of information among designated spokespersons. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.7.8 *Rumor Control*

Technical Information in the Plan: [G.4.c]
Section G.4 of the LNP Emergency Plan states that the timely exchange of information among spokespersons will dispel most rumors. Additional rumor control is accomplished through obtaining and disseminating accurate information through representatives of the ENC. Progress Energy Customer Service Centers would handle customer inquiries.

Technical Evaluation: [G.4.c]
The staff finds that the LNP Emergency Plan adequately describes coordinated arrangements for dealing with rumors. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.7.9 *Annual Media Orientation*

Technical Information in the Plan: [G.5]
Section G.5, “News Media Training,” of the LNP Emergency Plan states that Progress Energy, in cooperation with State and county Emergency Management, conducts an annual program to

acquaint the news media with the emergency plans, information concerning radiation and operation of LNP, and points of contact for release of public information during any emergency. These briefings may be conducted in the form of a group presentation or documented individual contacts throughout the year.

Technical Evaluation: [G.5]
The staff finds that the LNP Emergency Plan adequately describes coordinated programs that will be conducted at least annually to acquaint news media with the emergency plans, information concerning radiation, and points of contact for release of public information in an emergency. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.7.10 *Conclusion*

On the basis of its review of the LNP Emergency Plan, the staff concludes that the information provided in the LNP Emergency Plan regarding public education and information is acceptable and meets the requirements of 10 CFR 50.47(b)(7) because it conforms to the guidance in Evaluation Criterion G of NUREG-0654/FEMA-REP-1 and meets the applicable requirements of Appendix E to 10 CFR Part 50.

13.3C.8 Emergency Facilities and Equipment

13.3C.8.1 *Regulatory Basis*

In determining whether the proposed emergency plan met the applicable regulatory requirements in 10 CFR 50.47(b)(8) for emergency facilities and equipment, the staff evaluated it against the detailed evaluation criteria in NUREG-0654/FEMA-REP-1, Supplement 1 to NUREG-0737, NUREG-0696, and NSIR/DPR-ISG-01. The staff also evaluated the proposed emergency plan against applicable regulatory requirements related to the area of "Emergency Facilities and Equipment," in Appendix E to 10 CFR Part 50, 10 CFR 50.34, and 10 CFR 50.72.

<u>Technical Support Center</u>

13.3C.8.2 *Technical Support Center Functions*

Technical Information in the Emergency Plan: [H.1] {Appendix E, Section IV.E.8} (8.2.1.a)
Section H.1.2, “Technical Support Centers [TSC],” of the LNP Emergency Plan describes the establishment of a TSC for each unit. These facilities include necessary supplies and communications equipment to permit effective direction and control during an emergency. Details of facility operation are provided in implementing procedures. Appendix 5 identifies a procedure for activation and operation of the TSC. Duties of the EC that will be transferred from the CR to the TSC following activation are discussed in Section B.5.1.d, “Emergency Coordinator-TSC.” Section E.1.2, “Off-Site Emergency Response Organizations,” states the TSC is responsible for notifying State and local agencies until the EOF is operational. Functions

to be performed by the TSC, discussed in Section H.1.2.b, “Functions,” of the LNP Emergency Plan include:

1. Command and communications center for EC and assigned staff upon activation.

2. Perform emergency classification, notification of offsite agencies (including the NRC), and provide PARs to offsite agencies.

3. Provide plant management and technical support to plant operations personnel.

4. Prioritize emergency response team (ERT) activities in the plant.

5. Assist the CR in accident assessment.

Technical Evaluation: [H.1] {Appendix E, Section IV.E.8} (8.2.1.a)
The staff finds that the LNP Emergency Plan adequately describes the TSC functions. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and Supplement 1 to NUREG-0737 and meets the applicable requirements of Appendix E to 10 CFR Part 50.

13.3C.8.3 *TSC Location*

Technical Information in the Emergency Plan: (8.2.1.b) (50.34(f)(2)(xxv))
Section H.1.2.a.1, “Characteristics,” of the LNP Emergency Plan states that the TSCs are located within the protected area in the passage from the Annex Building to the CRs (of Units 1 and 2). The applicant proposed EP ITAAC 7.1.2 to verify that the TSC is close to the CR, and the walking distance from the TSC to the CR does not exceed two minutes. The applicant proposed EP ITAAC 7.1.6 to verify that the TSC is separate from the OSC.

Technical Evaluation: (8.2.1.b) (50.34(f)(2)(xxv))
The staff finds that the LNP Emergency Plan adequately describes the TSC location consistent with the TSC location in the referenced AP1000 DCD. The TSC is located within the site protected area (onsite) to facilitate necessary interaction with the CR, OSC, EOF and other personnel involved with the emergency. This is acceptable because it conforms to the guidance in Supplement 1 to NUREG-0737 and meets the applicable requirements of 10 CFR 50.34. The staff’s evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.8.4 *TSC Size and Staffing Requirements*

Technical Information in the Emergency Plan: (8.2.1.c and j)
Section H.1.2.a.1 of the LNP Emergency Plan states that each TSC command room covers 2144 square (sq) feet (ft), with 4 adjoining conference rooms that cover 988 sq ft. The TSCs are sized to accommodate approximately 25 persons, including 20 persons designated by Progress Energy and 5 NRC personnel. The applicant proposed EP ITAAC 7.1.1 to verify each

TSC has at least 1875 sq ft of floor space (75 sq ft per person to accommodate a minimum of 25 persons).

Figure B-2 illustrates minimum staffing positions and other designated positions necessary to support activation and operation of the TSC. Section B.5., "Plant Emergency Response Staff" describes the positions, titles, and major tasks of personnel assigned to the functional areas of emergency activities within the TSC. Major tasks, functional areas and positions within the TSC are also outlined in Table B-1 of the LNP Emergency Plan. Section H.4, "Activation and Staffing of Emergency Response Facilities," of the LNP Emergency Plan states that ERFs are staffed and declared operational in accordance with EPIPs. In RAI 13.3-21(B), the staff requested that the applicant explain whether the TSC will be operational within one hour following activation of the facility. In response, the applicant provided a discussion regarding staff augmentation times (30-45 and 60-75 minutes) consistent with the minimum staffing augmentation times identified in Table B-1 of the LNP Emergency Plan. The applicant also stated, in part, that a goal of 60 minutes, once notified, has been established for minimum staffing of the TSC. The TSC will be declared operational within 15 minutes of achieving minimum staffing. This time is used as turnover time. The applicant committed to revise Section H.4 of the LNP Emergency Plan for clarification of activation goals for the TSC. In its prior response to RAI 13.3-18(D)(1), the applicant provided its justification for extended augmentation and ERF activation times. The applicant stated, in part, that operating experience from Crystal River Nuclear Facility, located approximately 9 miles from LNP, has shown that based on local demographics, weather, traffic, and housing availability for station employees, it is achievable to augment staffing within 30 to 60 minutes after notification of an emergency. Therefore, since Crystal River is in close proximity to LNP, it is reasonable to conclude the same response time will be achieved for the LNP ERO. In its subsequent response to RAI 13.3-45(2), the applicant removed the reference to augmentation times of 30-45 and 60-75 minutes in Table B-1 and Section H.4 of the LNP Emergency Plan, restating its proposed goal of 60 minutes for achieving minimum staffing of the TSC following notification of ERO personnel. Additional discussion regarding augmentation times applicable to the TSC can be found in Section 13.3C.2.7 of this SER. Appendix 5 contains an EPIP, "Activation and Operation of the Technical Support Center," that supports and implements this section of the LNP Emergency Plan.

Technical Evaluation: (8.2.1.c and j)
The staff finds that the proposed revisions to the minimum staff augmentation and TSC activation goals provided in response to Supplemental RAI 13.3-45(2), in consideration of its prior response to RAIs 13.3-21(B) and 13.3-18(D)(1), to be acceptable because it describes provisions for a timely staff augmentation and activation of the TSC, and conforms to the guidance in Supplement 1 to NUREG-0737. The staff created Confirmatory Item 13.3-45(2) to track the proposed textual revision to the emergency plan consistent with the applicant's RAI responses. Additional staff technical evaluation regarding ERO staff augmentation times can be found in Section 13.3C.2.7 of this SER.

Resolution of Confirmatory Item 13.3-45(2)

Confirmatory Item 13.3-45(2) is an applicant commitment to update the LNP Emergency Plan. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). As a result, Confirmatory Item 13.3-45(2) is now closed.

The staff finds that the LNP Emergency Plan adequately describes the TSC size and staffing requirements. This is acceptable because it conforms to the regulatory guidance in Supplement 1 to NUREG-0737. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.8.5 *TSC Structure*

Technical Information in the Emergency Plan: (8.2.1.d)
Section H.1.2.a.2 of the LNP Emergency Plan states the TSC exterior walls, roof, and floor are built to Seismic Category II requirements. The applicant proposed EP ITAAC 7.1 to verify that an inspection of the as-built TSCs will be performed, including a test of the capabilities.

Technical Evaluation: (8.2.1.d)
The applicant stated in the LNP Emergency Plan that the TSC is built to Seismic Category II requirements. This exceeds the criterion in Supplement 1 to NUREG-0737, which states that the TSC should be built in accordance with the uniform building code. Therefore, the staff finds that the LNP Emergency Plan adequately describes the TSC structure. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.8.6 *TSC Environmental Controls*

Technical Information in the Emergency Plan: (8.2.1.e)
Section H.1.2.a.4 of the LNP Emergency Plan states the TSC is environmentally controlled to provide room air temperature, humidity and cleanliness appropriate for personnel and equipment.

Technical Evaluation: (8.2.1.e)
The staff finds that the LNP Emergency Plan adequately describes the TSC environmental controls. This is acceptable because it meets the applicable regulatory guidance in Supplement 1 to NUREG-0737.

13.3C.8.7 *TSC Radiological Protection*

Technical Information in the Emergency Plan: (8.2.1.f)
Section H.1.2.a.3 of the LNP Emergency Plan states the TSC is provided with radiation protection equivalent to CR habitability requirements, such that the dose to an individual in the TSC for the duration of a design basis accident is less than 5 roentgen equivalent man (rem) total effective dose equivalent (TEDE). The applicant proposed EP ITAAC 7.1.4 to verify that

the TSC ventilation systems include a high efficiency particulate air (HEPA) and charcoal filter, and that radiation monitors are installed.

Technical Evaluation: (8.2.1.f)
The staff finds that the LNP Emergency Plan adequately describes the TSC radiological protection. This is acceptable because it meets the applicable regulatory guidance in Supplement 1 to NUREG-0737. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.8.8 *TSC Communications*

Technical Information in the Emergency Plan: (8.2.1.g)
Section H.1.2.c.1, "Emergency Equipment and Supplies," of the LNP Emergency Plan states, in part, that the TSC maintains reliable voice communications with the CRs, EOF, OSCs, NRCOC, State and local warning points, and State EOCs. Additional information related to communication systems can be found in Section 13.3C.6, "Emergency Communications," of this SER. The applicant proposed EP ITAAC 7.1.3 to verify that communications equipment is installed, and voice transmission and reception are accomplished between the CRs, TSCs, OSCs, and EOFs.

Technical Evaluation: (8.2.1.g) The staff finds that the LNP Emergency Plan adequately describes the TSC communications. This is acceptable because it meets the applicable regulatory guidance in Supplement 1 to NUREG-0737. Evaluation of communication equipment can be found in Section 13.3C.6.2 of this SER. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.8.9 *TSC Data Collection, Storage, and Analysis*

Technical Information in the Emergency Plan: (8.2.1.h)
Section H.5, "On-Site Monitoring Systems," of the LNP Emergency Plan states Progress Energy maintains and operates onsite monitoring systems to provide data essential for initiating emergency measures and performing accident assessment. Section H.1.2.c.2, "Emergency Equipment and Supplies," states that the TSCs contain a visual display system capable of displaying plant data, safety parameter display systems (SPDSs), and radiation monitoring systems (RMSs) information. Section H.8, "Meteorological Instrumentation and Procedures," states, in part, that real time meteorological data with provisions for computerized historical storage and retrieval, for use in accident scenarios will be available in the TSCs. In addition, by letter dated December 21, 2010, the applicant proposed to revise the LNP Emergency Plan to include a statement that the TSC has been established consistent with NUREG-0696 guidelines. The applicant proposed EP ITAAC 7.1.5 to verify that the TSC receives, stores, processes, and displays plant and environmental information, which enables the initiation of emergency measures and the performance of emergency assessment. These capabilities are demonstrated during testing and acceptance activities. Additional information regarding the availability of meteorological information and data, including atmospheric diffusion estimates,

can be found in Section 2.3.3, " Onsite Meteorological Measurement Program," and Section 7.5, "Safety-Related Display Information," of this SER.

Technical Evaluation: (8.2.1.h)
The staff finds that the applicant's proposed reference to the TSC being established consistent with NUREG-0696 guidance to be acceptable. The staff created Confirmatory Item 13.3-2 to track the applicant's inclusion of the information as stated above in the next revision to the LNP Emergency Plan.

Resolution of Confirmatory Item 13.3-2

Confirmatory Item 13.3-2 is an applicant commitment to update the LNP Emergency Plan. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). As a result, Confirmatory Item 13.3-2 is now closed.

The staff finds that the LNP Emergency Plan adequately describes the TSC functions of data collection, storage, and analysis. This is acceptable because it meets the applicable regulatory guidance in Supplement 1 to NUREG-0737. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.8.10 *TSC Human Factors Engineering*

Technical Information in the Emergency Plan: (8.2.1.h and k)
Section H.1.2.a.7, "Emergency Equipment and Supplies," of the LNP Emergency Plan states the TSC is designed using human factors criteria in APP-GW-GLR-136, "AP1000 Human Factors' Program Implementation for the Emergency Operations Facility and the Technical Support Center." In addition, by letter dated December 21, 2010, the applicant proposed to revise the LNP Emergency Plan to include a statement that the TSC has been established consistent with NUREG-0696 guidelines. In response to RAI 13.03-49(4)(b), in part, the applicant proposed additional EP ITAAC acceptance criteria (12.1.1.D.2.d) that states, in part, the applicant will demonstrate the capability of the TSC equipment and data displays to clearly reflect the affected unit during an emergency. Additional information regarding human factors engineering (HFE) for the TSC can be found in Chapter 18, "Human Factors Engineering," of the AP1000 DCD and its supplements, and Section 18.2 of this SER.

Technical Evaluation: (8.2.1.h and k)
The staff created Confirmatory Item 13.3-2 in Section 13.3C.8.9 of this SER to track the applicant's inclusion of its reference to NUREG-0696 in a future revision to the LNP Emergency Plan. The staff's evaluation of the TSC HFE pursuant to Supplement 1 to NUREG-0737 is addressed in Section 18.2 of this SER.

Resolution of Confirmatory Item 13.3-2

Confirmatory Item 13.3-2 is an applicant commitment to update the LNP Emergency Plan. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). As a result, Confirmatory Item 13.3-2 is now closed.

The staff finds that the LNP Emergency Plan adequately addresses the LNP HFE Program. This is acceptable because it meets the guidance in Supplement 1 to NUREG-0737. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.8.11 *TSC Plant Records*

Technical Information in the Emergency Plan: (8.2.1.i)
Section H.1.2.c.3, "Emergency Equipment and Supplies," of the LNP Emergency Plan states that the TSCs contain reference materials that include: mechanical and electrical systems drawings; the plant operating manual; the FSAR; and corporate, plant, State, and local emergency plans that are available in hardcopy or online. In addition, by letter dated December 21, 2010, the applicant proposed to revise the LNP Emergency Plan to include a statement that the TSC has been established consistent with NUREG-0696 guidelines.

Technical Evaluation: (8.2.1.i)
The staff finds that the applicant's proposed reference to the TSC being established consistent with NUREG-0696 guidance to be acceptable. The staff created Confirmatory Item 13.3-2 in Section 13.3C.8.9 of this SER to track the applicant's inclusion of its reference to NUREG-0696 in a future revision to the LNP Emergency Plan.

Resolution of Confirmatory Item 13.3-2

Confirmatory Item 13.3-2 is an applicant commitment to update the LNP Emergency Plan. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). As a result, Confirmatory Item 13.3-2 is now closed.

The staff finds that the LNP Emergency Plan adequately describes the TSC plant records availability. This is acceptable because it conforms to the guidance in Supplement 1 to NUREG-0737.

13.3C.8.12 *TSC Activation*

Technical Information in the Emergency Plan: [H.4]
Section H.4 of the LNP Emergency Plan states that Progress Energy's staffing and activation of the TSC is required upon declaration of an emergency classification of alert, SAE, or GE. In addition, the TSC is staffed and declared operational in accordance with an EPIP identified in Appendix 5 to the LNP Emergency Plan titled, "Activation and Operation of the TSC."

Technical Evaluation: [H.4]
In Section 13.3C.2.7 and 13.3C.8.4 of this SER, the staff created Confirmatory Item 13.3-45(2) to track a textual revision to the LNP Emergency Plan that clarifies the language justifying untimely augmentation of the ERO and activation of its ERFs.

Resolution of Confirmatory Item 13.3-45(2)

Confirmatory Item 13.3-45(2) is an applicant commitment to update the LNP Emergency Plan. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). As a result, Confirmatory Item 13.3-45(2) is now closed.

The staff finds that the LNP Emergency Plan adequately provides for timely activation and staffing of facilities and centers described in the plan. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

Operations Support Center

13.3C.8.13 *Operations Support Center Functions*

Technical Information in the Emergency Plan: [H.1] (8.3.1.a)
Functions to be performed by the OSC, described in Section H.1.3.b, "Functions," of the LNP Emergency Plan include:

1. Assembly location for the OSC manager and operational support (i.e., Maintenance, Operations, Radiation Protection, and Chemistry) personnel for receipt of equipment and assignments to aid in response to an emergency.

2. Briefing and dispatch of emergency teams.

Section B.5.1.n, "OSC Manager," of the emergency plan states the OSC manager is responsible for providing direction to the total onsite maintenance and equipment restoration effort, including coordinating the dispatch of OSC teams.

Technical Evaluation: [H.1] (8.3.1.a)
The staff finds that the LNP Emergency Plan adequately describes the OSC functions. This is acceptable because it meets the applicable regulatory guidance in Supplement 1 to NUREG-0737 and conforms to the guidance described in NUREG-0654/FEMA-REP-1.

13.3C.8.14 *OSC Location*

Technical Information in the Emergency Plan: (8.3.1.b) (50.34(f)(2)(xxv))
Section H.1.3.a.1, "Characteristics," of the LNP Emergency Plan states the OSC is located inside the protected area on the second floor of the Annex Building of each unit adjacent to the

CRs. The applicant proposed EP ITAAC 7.1.6 to verify that there is an OSC located inside the unit's protected area separate from the CR and TSC.

Technical Evaluation: (8.3.1.b) (50.34(f)(2)(xxv))
The staff finds that the LNP Emergency Plan adequately describes the location of the OSCs. This is acceptable because it conforms to the guidance described in Supplement 1 to NUREG-0737 and 10 CFR 50.34. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.8.15   *OSC Coordination Activities*

Technical Information in the Emergency Plan: (8.3.1.a)
Section H.1.3.b, "Functions," of the LNP Emergency Plan states that the OSC is an assembly location for the OSC Manager and support personnel (e.g., operations, maintenance, health physics, and chemistry) for receipt of equipment and assignments to aid in response to an emergency. The OSC is the location for the briefing and dispatch of emergency response teams. This location includes separate areas for coordinating and planning OSC activities.

Technical Evaluation: (8.3.1.a)
The staff finds that the LNP Emergency Plan adequately describes the OSC coordination of activities function. This is acceptable because it conforms to the regulatory guidance in Supplement 1 to NUREG-0737.

13.3C.8.16   *OSC Communications*

Technical Information in the Emergency Plan: (8.3.1.c)
Section H.1.3.c.1, "Characteristics," of the LNP Emergency Plan states the OSC maintains reliable voice communications with the CRs, TSCs, and EOF. Additional information related to communication systems can be found in Section F, "Emergency Communications," of the LNP Emergency Plan and 13.3C.6 of this SER. The applicant proposed EP ITAAC 7.1.7 to verify that communication equipment is installed, and voice transmissions and reception are accomplished between the OSC and OSC teams, the TSC, and CRs.

Technical Evaluation: (8.3.1.c)
The staff finds that the LNP Emergency Plan adequately describes the OSC communications. This is acceptable because it conforms to the applicable regulatory guidance in Supplement 1 to NUREG-0737. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.8.17   *OSC Activation and Staffing*

Technical Information in the Emergency Plan: [H.4]
In Sections 13.3C.2.7, 13.3C.8.4 and 13.3C.8.12 of this SER, the staff provided discussion regarding the timely activation and staffing of ERFs, including the OSC. Section H.4 of the LNP Emergency Plan states, in part, that a goal of 60 minutes has been established for minimum

staffing in the OSC. It is the goal of the organization to be capable of declaring the OSC operational within 15 minutes of achieving minimum staffing.

Technical Evaluation: [H.4]
In Section 13.3C.2.7 and 13.3C.8.4 of this SER, the staff created Confirmatory Item 13.3-45(2) to track a textual revision to the LNP Emergency Plan that eliminates language justifying untimely augmentation of the ERO and activation of its ERFs, including the OSC.

<u>Resolution of Confirmatory Item 13.3-45(2)</u>

Confirmatory Item 13.3-45(2) is an applicant commitment to update the LNP Emergency Plan. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). As a result, Confirmatory Item 13.3-45(2) is now closed.

The staff finds that the LNP Emergency Plan adequately provides for activation and staffing of the OSC. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.8.18   *OSC Capacity and Supplies*

Technical Information in the Emergency Plan: [H.9]
Section H.1.3, "Operations Support Centers," of the LNP Emergency Plan establishes an OSC for each unit. The total area for each OSC is approximately 2,888 square feet. Additional space is available in adjacent offices and locker rooms, as needed. Table H-1, "Typical Emergency Kit Equipment/Supplies and Locations," provides a general list of emergency equipment and supplies available in the OSC that includes personnel dosimetry, protective clothing, portable radiation monitoring equipment, and portable lighting. Section J.6, "Protective Measures," states that LNP distributes protective equipment and supplies to personnel remaining or arriving onsite, as needed, to control radiological exposure or contamination including respiratory protection. Section F of the LNP Emergency Plan states that portable ultra-high frequency (UHF) radios are available to emergency teams for limited communication. Appendix 5 identifies an implementing procedure for the OSC titled, "Activation and Operation of the Operational Support Center," which supports and implements Section H of the LNP Emergency Plan.

Technical Evaluation: [H.9]
The staff finds the LNP Emergency Plan adequately describes the OSC capacity and supplies. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

<u>Emergency Operations Facility</u>

13.3C.8.19   *Emergency Operations Facility Functions*

Technical Information in the Emergency Plan: [H.2] {Appendix E, Section IV.E.8} (8.4.1.a)
Section H.2.1.b.1, "Functions," of the LNP Emergency Plan states, in part, that the EOF:

1. Is capable of supporting extended emergency operations, including simultaneous activation with CR3;

2. Provides a near-site location for assembling EOF staff and representatives of Federal, State, county, and industry emergency response agencies;

3. Upon activation, performs offsite notification, PARs, environmental monitoring, and dose projection;

4. Emergency communications systems monitoring and control;

5. Provides technical analysis and support;

6. Receives and displays site status and parameters data;

7. Serves as the Recovery Center during recovery operations;

Section B.5.2.a, "EOF Director," of the emergency plan states that upon activation of the EOF, the EOF Director is responsible for overall command and control of the LNP response to an emergency. This includes activities for providing information to, and interfacing with, offsite authorities, monitoring offsite results of the event, protecting plant personnel outside the protected area(s), supporting the onsite organization and coordinating the flow of information to the public information ERO. In RAI 13.3-21(A), the staff requested, in part, clarification from the applicant regarding the use of a shared EOF for LNP and CR3, and its ability to accommodate a response to a simultaneous emergency at both sites. In response, the applicant stated, in part, that the LNP EOF will be a shared facility with sufficient space and equipment to handle the response to a simultaneous event at both sites. The applicant stated that equipment will be available in adequate number with connection capability to facilitate unimpeded communication with offsite agencies, onsite ERFs and the ENC. The applicant stated that the EOF will have the capability to acquire, display, and evaluate radiological, meteorological, and plant system data pertinent to offsite protective measures for both LNP and CR3 without decreasing effectiveness. The applicant committed to revise Section H.2.1 of the LNP Emergency Plan to clarify the use of the EOF for a simultaneous event. In supplemental RAI 13.3-39, the staff requested that the applicant include in the LNP Emergency Plan additional information related to the shared EOF location and functionality provided in its prior responses to RAIs 14.3.10-1(J), 13.3-21(A), 13.3-18(3)(A), and 13.3-18(3)(D). The applicant's response committed to including the associated information from these RAIs into a future revision of the LNP Emergency Plan.

In supplemental RAI 13.3-31, the staff requested that the applicant propose a license condition to demonstrate the integrated capability and functionality of the existing EOF with LNP and Crystal River TSCs, the NRC, and other Federal, State, and local coordination centers, prior to use of the EOF for LNP emergency response. In response, the applicant committed to revise proposed License Condition 11 in Part 10, "Proposed Licensing Conditions (including ITAAC)," of the COL application to state that Progress Energy will demonstrate the capability of the EOF to handle simultaneous activation for a simulated emergency condition. Integrated communication, data capability, and functionality will include the LNP and Crystal River TSC, NRC, and other Federal, State, and local coordination centers.

Technical Evaluation: [H.2] {Appendix E, Section IV.E.8} (8.4.1.a)
The applicant proposed the use of a shared EOF between LNP Units 1 and 2, and CR3, which is owned and operated by Progress Energy. The EOF is an existing facility approved for use by the NRC for CR3. The staff's evaluation of the existing EOF as a shared facility, included the consideration of past implementation practices for shared facilities pertaining to operating reactors and the associated Commissions requirements for operation. In addition, the staff's evaluation focused on the potential impact to the functionality and capability of the existing facility with the addition of the two new units.

PEF has committed in a license condition to demonstrate its integrated capability of the EOF to handle the simultaneous activation of the LNP and CR3 EROs for a simulated emergency condition. Integrated communication, data capability, and functionality will include the LNP and Crystal River TSC, NRC (site teams and incident response centers), and other Federal, State, and local coordination centers, as appropriate.

The staff finds the additional information and proposed textual revisions to the emergency plan and Part 10 of the COL application provided in response to RAIs 13.3-21(A), 13.3-31, and 13.3-39 to be acceptable because they conform to the guidance in NUREG-0654/FEMA-REP-1 and Supplement 1 to NUREG-0737 and meets the applicable requirements of Appendix E to 10 CFR Part 50. The staff confirmed that the additional information and proposed textual revisions provided in response to these RAIs have been incorporated into Revisions 1 and 2 of the LNP Emergency Plan and Part 10 of the COL application.

By letter dated November 8, 2012, from PEF to the NRC, the applicant provided its response to address the Final Rule on Enhancements to Emergency Preparedness Regulation effective November 23, 2011. In regard to implementation of the EP rule pertaining to the distance and performance based criteria of the EOF, there were no changes warranted for the LNP Emergency Plan.

By letter dated April 18, 2013, from PEF to NRC regarding impacts from retirement of the CR3 nuclear plant, the applicant renamed the EOF to remove any reference to the Crystal River Training Center. The EOF is now referred to in the LNP Emergency Plan as the LNP EOF. In addition, the applicant added a conditional statement to address when the EOF is required for use by CR3 in the LNP Emergency Plan.

As discussed in Section 13.3.4, CR3 was granted exemptions from specific EP standards including the requirement to have an EOF.

In consideration for the applicant's response to address the new EP rule, and deletion of the reference to the Crystal River Training Center with additional conditional language, the staff finds the LNP Emergency Plan adequately describes the EOF functions. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1, Supplement 1 to NUREG-0737, and NSIR/DPR-ISG-01, and meets the requirements of Appendix E to 10 CFR Part 50.

13.3C.8.20 *EOF Location*

Technical Information in the Emergency Plan: (8.4.1.b) (50.34(f)(2)(xxv)) {Appendix E, Section IV.E.8}
Section H.2.1, "Emergency Operations Facility," of the LNP Emergency Plan states, in part, that the EOF is located outside the 10-mile EPZ but within 20 miles of the LNP TSCs on West Venable Street in Crystal River, Florida. The facility is a shared EOF with CR3. In addition, the applicant stated that the EOF has been established consistent with NUREG-0696, "Functional Criteria for Emergency Response Facilities," guidelines. Section H.2.1.c.7, "Emergency Equipment and Supplies," states that radiological monitoring equipment will be provided to the EOF by Health Physics if conditions warrant.

Technical Evaluation: (8.4.1.b) (50.34(f)(2)(xxv)) {Appendix E, Section IV.E.8}
The EOF is located outside of the 10-mile plume exposure pathway EPZ and within 20 miles of the LNP TSCs. Consistent with the guidance in Supplement 1 to NUREG-0737, there is no special radiation protection factor required for the facility.

By letter dated November 8, 2012, from PEF to the NRC, the applicant provided its response to address the Final Rule on Enhancements to Emergency Preparedness Regulation effective November 23, 2011. In regard to implementation of the EP rule pertaining to the distance and performance based criteria of the EOF, there were no changes warranted for the LNP Emergency Plan.

By letter dated April 18, 2013, from PEF to NRC regarding impacts from retirement of the CR3 nuclear plant, the applicant renamed the EOF to remove any reference to the Crystal River Training Center. The EOF is now referred to in the LNP Emergency Plan as the LNP EOF. In addition, the applicant added a conditional statement to address when the EOF is required for use by CR3. As discussed in Section 13.3.4, CR3 was granted exemptions from specific EP standards including the requirement to have an EOF.

In consideration for the applicant's response to the new EP rule, and deletion of the reference to the Crystal River Training Center with additional conditional language, the staff finds the LNP Emergency Plan adequately describes the EOF location. This is acceptable because it conforms to the guidance in Supplement 1 to NUREG-0737 and NSIR/DPR-ISG-01 and meets the requirements of 50.34(f)(2)(xxv).

13.3C.8.21 *EOF Size*

Technical Information in the Emergency Plan: (8.4.1.c)
Section H.2.1.a.3, "Functions," of the LNP Emergency Plan states, in part, that the EOF provides approximately 21,000 sq. ft of working space for Progress Energy and other support personnel. Section H.2.1.b.3, "Functions," states the EOF will serve as an assembly point for

EOF staff and representatives of Federal, State, county, and industry emergency response agencies.

Technical Evaluation: (8.4.1.c)
The staff finds the LNP Emergency Plan adequately describes the EOF size requirements. This is acceptable because it conforms to the guidance in Supplement 1 to NUREG-0737.

13.3C.8.22 *EOF Structural Capabilities*

Technical Information in the Emergency Plan: (8.4.1.d)
Section H.2.1.a of the LNP Emergency Plan states, in part, that CR3 will share the existing EOF with LNP.

Technical Evaluation: (8.4.1.d)
Initially, the applicant proposed to use the existing CR3 EOF for support of emergency planning for LNP Units 1 and 2. The staff's review of the EOF focused on the extension of the existing facility as it applies to the proposed reactor units at the LNP site. Currently, the EOF is an existing NRC approved facility for CR3 that meets the requirements of 10 CFR 50.47 and Appendix E to Part 50, and conforms to the guidance in NUREG-0696 and NUREG-0737, Supplement No. 1. The staff determined the EOF was acceptable for use at LNP Units 1 and 2 because: 1) the NRC performs oversight of emergency preparedness, including the EOF, by monitoring performance indicators; 2) the EOF is inspected periodically during routine inspections, drills and exercises; and 3) any changes to the EOF are reviewed in accordance with the established inspection program and requirements for operating reactors.

However, by letter dated April 18, 2013, from PEF to NRC, the applicant proposed a revision to the LNP Emergency Plan to address the future state of CR3 as it relates to decommissioning activities and the anticipated relaxation of offsite EP responsibilities for CR3. In consideration of these circumstances, the applicant anticipates the EOF will no longer be required for response to an emergency event at CR3. In LNP Emergency Plan, the EOF has been renamed the LNP EOF and is expected to support the future needs of LNP only. The staff anticipates a lapse in time for which the readiness capabilities of the EOF will no longer be required. By letter dated January 10, 2014, from DEF to the NRC, the applicant proposed EP ITAAC 7.2.3 through 7.2.5 to address regulatory guidance criteria in NUREG-0696 and Supplement 1 to NUREG-0737 that are not addressed in the LNP Emergency Plan. Prior to fuel load, these EP ITAAC will provide staff assurance that the EOF continues to comply with the uniform building code; the EOF is environmentally controlled to provide room air temperature, humidity, and cleanliness appropriate for personnel and equipment; and the EOF is provided with industrial security when it is activated to exclude unauthorized personnel and when it is idle to maintain its readiness. Given that the EOF may not be required to maintain its functionality for some time prior to LNP operations, the staff found these ITAAC necessary to ensure that the EOF is constructed as designed, as required by 10 CFR 52.80. Therefore, the staff finds the applicant's proposed EP ITAAC 7.2.3 through 7.2.5 acceptable since they conform to the guidance in NUREG-0696 and Supplement 1 to NUREG-0737 and meet the requirements in 10 CFR 52.80. The staff subsequently finds the LNP EOF acceptable.

The applicant is proposing to use the existing CR3 EOF for LNP Units 1 and 2. The staff's review focused on the extension of the existing facility as it applies to the proposed reactor units at the LNP site. The EOF is an existing NRC approved facility for CR3 that conforms to the guidance in NUREG-0737 as it pertains to its structure. Therefore, the staff finds the EOF acceptable for use at LNP Units 1 and 2 because: 1) the NRC performs oversight of emergency preparedness, including the EOF, by monitoring performance indicators; 2) the EOF is inspected periodically during routine inspections, drills and exercises; and 3) any changes to the EOF are reviewed in accordance with the established inspection program and requirements for operating reactors.

13.3C.8.23 *EOF Environmental Controls*

Technical Information in the Emergency Plan: (8.4.1.e)
Section H.2.1.a of the LNP Emergency Plan states, in part, that CR3 will share the existing EOF with LNP.

Technical Evaluation: (8.4.1.e)
Initially, the applicant proposed to use the existing CR3 EOF for support of emergency planning for LNP Units 1 and 2. The staff's review of the EOF focused on the extension of the existing facility as it applies to the proposed reactor units at the LNP site. Currently, the EOF is an existing NRC approved facility for CR3 that meets the requirements of 10 CFR 50.47 and Appendix E to Part 50, and conforms to the guidance in NUREG-0696 and NUREG-0737, Supplement No. 1. The staff determined the EOF was acceptable for use at LNP Units 1 and 2 because: 1) the NRC performs oversight of emergency preparedness, including the EOF, by monitoring performance indicators; 2) the EOF is inspected periodically during routine inspections, drills and exercises; and 3) any changes to the EOF are reviewed in accordance with the established inspection program and requirements for operating reactors.

However, by letter dated April 18, 2013, from PEF to NRC, the applicant proposed a revision to the LNP Emergency Plan to address the future state of CR3 as it relates to decommissioning activities and the anticipated relaxation of offsite EP responsibilities for CR3. In consideration of these circumstances, the applicant anticipates the EOF will no longer be required for response to an emergency event at CR3. In LNP Emergency Plan, the EOF has been renamed the LNP EOF and is expected to support the future needs of LNP only. As discussed in Section 13.3.4, CR3 was granted exemptions from specific EP standards including the requirement to have an EOF. The staff anticipates a lapse in time for which the readiness capabilities of the EOF will no longer be required. By letter dated January 10, 2014, from DEF to the NRC, the applicant proposed EP ITAAC 7.2.3 through 7.2.5 to address regulatory guidance criteria in NUREG-0696 and Supplement 1 to NUREG-0737 that are not addressed in the LNP Emergency Plan. Prior to fuel load, these EP ITAAC will provide staff assurance that the EOF continues to comply with the uniform building code; the EOF is environmentally controlled to provide room air temperature, humidity, and cleanliness appropriate for personnel and equipment; and the EOF is provided with industrial security when it is activated to exclude unauthorized personnel and when it is idle to maintain its readiness. Given that the EOF may not be required to maintain its functionality for some time prior to LNP operations, the staff found these ITAAC necessary to ensure that the EOF is constructed as designed, as required by 10 CFR 52.80. Therefore, the staff finds the applicant's proposed EP ITAAC 7.2.3 through 7.2.5 acceptable since they conform to the guidance in NUREG-0696 and Supplement 1 to NUREG-0737 and meet the requirements in 10 CFR 52.80. The staff subsequently finds the LNP EOF acceptable.

The applicant is proposing to use the existing EOF formerly used by CR3, for LNP Units 1 and 2. The staff's review focused on the extension of the existing facility as it applies to the proposed reactor units at the LNP site. The EOF is an existing NRC approved facility formerly used by CR3 that conforms to the guidance in NUREG-0737 as it pertains to environmental controls. Therefore, the staff finds the EOF acceptable for use at LNP Units 1 and 2 because: 1) the NRC performs oversight of emergency preparedness, including the EOF, by monitoring performance indicators; 2) the EOF is inspected periodically during routine inspections and drills and exercises; and 3) any changes to the EOF are reviewed in accordance with the established inspection program and requirements for operating reactors.

EOF Voice and Data Communications and Information Collection

Technical Information in the Emergency Plan: (8.4.1.f)
Section F.1, "Description of Communication Links," of the LNP Emergency Plan states that Progress Energy maintains reliable communications links both within the plant, and between the plant and external EROs. Section H.2.1.c.2 of the LNP Emergency Plan states that the EOF is equipped with commercial telephones, the Progress Energy Voicenet system, and power based radio service for communications within the plant, with corporate facilities, and with offsite organizations. An automatic ringdown telephone system provides communications between the EOF and the TSCs. Messages, technical data, and other emergency-related information can be rapidly and efficiently communicated through facsimile equipment among and between the ERFs as well as the State and county EOCs. Special communications systems are available for non-Progress Energy support groups. The EOF is also equipped with the State Hot Ringdown Telephone System for communication with the SWPT, the Florida DHBRC, and the county EOCs. Section F.1.f of the LNP Emergency Plan states, in part, that communications between the EOF, to the NRCOC is via the ETS or private phone. Communications from these facilities to the NRC Regional Office is via private telephone. Additional information and the staff's evaluation related to emergency communication systems can be found in Section F, "Emergency Communications," of the LNP Emergency Plan and Section 13.3C.6 of this SER. The applicant proposed EP ITAAC 7.2.1 to verify that communication equipment is installed and voice transmission and reception are accomplished between the CRs, TSC, EOF, radiological monitoring teams, NRC, State and county agencies, and ENC. Section 13.3C.8.19 of this SER provides additional information regarding the availability of communication equipment to facilitate unimpeded communications during the response to an emergency at LNP.

Technical Evaluation: (8.4.1.f)
The staff finds the LNP Emergency Plan adequately describes the EOF voice and data communications and information collection capabilities. This is acceptable because it conforms to the guidance in Supplement 1 to NUREG-0737.

13.3C.8.24 *EOF Information Display, Storage and Analysis*

Technical Information in the Emergency Plan: (8.4.1.g)
Section H.2.1.b.7, "Functions," of the LNP Emergency Plan states the EOF receives and displays site status and parameter data. Section 13.3C.8.19 of this SER provides a discussion regarding the applicant's capability within the EOF to acquire, display and evaluate radiological, meteorological, and plant system data pertinent to offsite protective measures for LNP. Section H.5, "On-Site Monitoring Systems," of the LNP Emergency Plan states that Progress Energy maintains and operates onsite monitoring systems needed to provide data that is essential for initiating emergency measures and performing accident assessment. Section H.8, "Meteorological Instrumentation and Procedures," of the LNP Emergency Plan states that real time meteorological data with provisions for computerized historical storage and retrieval, for use in accident scenarios will be available in the EOF. Section I.5, "Meteorological Information," states, in part, that Progress Energy has the capability to access the NWS on a 24-hour basis to provide reliable backup meteorological data representative of site conditions. In addition, Section 13.3C.4.2 of this SER provides additional information regarding plant system and effluent parameter values characteristic of a spectrum of off-normal and accident conditions, including EP ITAAC 3.1 proposed by the applicant to verify that the specified parameters (facility system and effluent) are retrievable in the EOF, and the ranges of displays encompass the values specified in the emergency classification and action level scheme. The applicant also proposed EP ITAAC Acceptance Criteria 7.2.2 to verify that radiological data, meteorological data, and plant system data pertinent to offsite protective measures are acquired, displayed and evaluated in the EOF.

Technical Evaluation: (8.4.1.g)
The staff finds the LNP Emergency Plan adequately describes the EOF information display, storage, and analysis. This is acceptable because it conforms to the guidance in Supplement 1 to NUREG-0737.

13.3C.8.25 *EOF Plant Records*

Technical Information in the Emergency Plan: (8.4.1.h)
Section H.2.1.c.6, "Emergency Equipment and Supplies," of the LNP Emergency Plan states a selection of technical documents is stored in the EOF at all times and are available whenever the EOF is activated. By letter dated December 21, 2010, the applicant proposed to revise the LNP Emergency Plan to include a statement that the EOF has been established consistent with NUREG-0696 guidelines.

Technical Evaluation: (8.4.1.h)
The staff created **Confirmatory Item 13.3-2** in Section 13.3C.8.9 of this SER to track the applicant's inclusion of its reference to NUREG-0696 in a future revision to the LNP Emergency Plan.

Resolution of Confirmatory Item 13.3-2

Confirmatory Item 13.3-2 is an applicant commitment to update the LNP Emergency Plan. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). As a result, Confirmatory Item 13.3-2 is now closed.

The staff finds that the LNP Emergency Plan adequately describes the availability of EOF plant records. This is acceptable because it conforms to the guidance in Supplement 1 to NUREG-0737.

13.3C.8.26 *EOF Industrial Security*

Technical Information in the Emergency Plan: (8.4.1.j)
Section H.2.1.a of the LNP Emergency Plan states, in part, that CR3 will share the existing EOF with LNP when CR3 offsite response capability requires an EOF.

Technical Evaluation: (8.4.1.j)
Initially, the applicant proposed to use the existing CR3 EOF for support of emergency planning for LNP Units 1 and 2. The staff's review of the EOF focused on the extension of the existing facility as it applies to the proposed reactor units at the LNP site. Currently, the EOF is an existing NRC approved facility for CR3 that meets the requirements of 10 CFR 50.47 and Appendix E to Part 50, and conforms to the guidance in NUREG-0696 and NUREG-0737, Supplement No. 1. The staff determined the EOF was acceptable for use at LNP Units 1 and 2 because: 1) the NRC performs oversight of emergency preparedness, including the EOF, by monitoring performance indicators; 2) the EOF is inspected periodically during routine inspections, drills and exercises; and 3) any changes to the EOF are reviewed in accordance with the established inspection program and requirements for operating reactors.

However, by letter dated April 18, 2013, from PEF to NRC, the applicant proposed a revision to the LNP Emergency Plan to address the future state of CR3 as it relates to decommissioning activities and the anticipated relaxation of offsite EP responsibilities for CR3. In consideration of these circumstances, the applicant anticipates the EOF will no longer be required for response to an emergency event at CR3. In LNP Emergency Plan, the EOF has been renamed the LNP EOF and is expected to support the future needs of LNP only. As discussed in Section 13.3.4, CR3 was granted exemptions from specific EP standards including the requirement to have an EOF.

The staff anticipates a lapse in time for which the readiness capabilities of the EOF will no longer be required. By letter dated January 10, 2014, from DEF to the NRC, the applicant proposed EP ITAAC 7.2.3 through 7.2.5 to address regulatory guidance criteria in NUREG-0696 and Supplement 1 to NUREG-0737 that are not addressed in the LNP Emergency Plan. Prior to fuel load, these EP ITAAC will provide staff assurance that the EOF continues to comply with the uniform building code; the EOF is environmentally controlled to provide room air temperature, humidity, and cleanliness appropriate for personnel and equipment; and the EOF is provided with industrial security when it is activated to exclude unauthorized personnel and when it is idle, to maintain its readiness. Given that the EOF may not be required to maintain its functionality for some time prior to LNP operations, the staff found these ITAAC necessary to ensure that the EOF is constructed as designed, as required by 10 CFR 52.80. Therefore, the staff finds the applicant's proposed EP ITAAC 7.2.3 through 7.2.5 acceptable since they conform to the guidance in NUREG-0696 and Supplement 1 to NUREG-0737 and meet the requirements in 10 CFR 52.80. The staff subsequently finds the LNP EOF acceptable.

The applicant is proposing to use the EOF formerly used for CR3 for LNP Units 1 and 2. The staff's review focused on the extension of the existing facility as it applies to the proposed reactor units at the LNP site. The EOF is an existing NRC approved facility for CR3 that conforms to the guidance in NUREG-0737 as it pertains to industrial security. Therefore, the staff finds the EOF acceptable for use at LNP Units 1 and 2 because: 1) the NRC has been performing oversight of emergency preparedness, including the EOF, by monitoring performance indicators; 2) the EOF has been inspected periodically during routine inspections and drills and exercises; and 3) any changes to the EOF were reviewed in accordance with the established inspection program and requirements for operating reactors.

13.3C.8.27 *EOF Human Factors*

Technical Information in the Emergency Plan: (8.4.1.k)
By letter dated December 21, 2010, the applicant proposed to revise the LNP Emergency Plan to include a statement that the EOF has been established consistent with NUREG-0696 guidelines. In RAI 13.3-49(4)(b), the staff requested that the applicant describe the capability of the TSC and EOF equipment and data displays to clearly identify and reflect the affected unit during a declared emergency, or propose an EP ITAAC to demonstrate this capability. In response, in part, the applicant proposed additional EP ITAAC acceptance criteria (12.1.1.D.2.d) that states the applicant will demonstrate the capability of the EOF equipment and data displays to clearly reflect the affected unit. Additional information regarding human factors engineering (HFE) for the EOF can be found in Chapter 18, "Human Factors Engineering," of the AP1000 DCD and its supplements, and Section 18.2 of this SER.

Technical Evaluation: (8.4.1.k)
The staff created Confirmatory Item 13.3-2 in Section 13.3C.8.9 of this SER to track the applicant's inclusion of its reference to NUREG-0696 in a future revision to the LNP Emergency Plan. The staff's evaluation of the EOF HFE pursuant to Supplement 1 to NUREG-0737 is addressed in Section 18.2 of this SER. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

Resolution of Confirmatory Item 13.3-2

Confirmatory Item 13.3-2 is an applicant commitment to update the LNP Emergency Plan. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). As a result, Confirmatory Item 13.3-2 is now closed.

13.3C.8.28 ***EOF Activation and Staffing***

Technical Information in the Emergency Plan: [H.4] (8.4.1.i)
Section B.5.2 of the LNP Emergency Plan describes the activation of the offsite ERO and responsibilities of the EOF Director. Section H.4 of the LNP Emergency Plan states, in part, that Progress Energy staffing and activation of the EOF is required upon declaration of an emergency classification of a site area emergency or GE. A goal of 60 minutes has been established for minimum staffing of the EOF. It is the goal of the organization to declare the facility operational within 15 minutes of achieving minimum staffing. The EOF is staffed and declared operational in accordance with an EPIP identified in Appendix 5 to the LNP Emergency Plan titled, "Activation and Operation of the Emergency Operations Facility." Section B.7 states that Progress Energy management, technical, and administrative personnel staff the EOF and provide augmented support for the plant staff as outlined in Table B-1.

Sections 13.3C.2.7, 13.3C.8.4, 13.3C.8.12, and 13.3C.8.17 of this SER provide additional information relating to the activation and staffing of ERFs, including response times applicable to the EOF. Section 13.3C.8.19 of this SER provides additional information relating to the activation of the EOF in response to a simultaneous emergency at both LNP and CR3 nuclear plant, including command and control of the facility and staffing, in accordance with procedures.

Technical Evaluation: [H.4] (8.4.1.i)
In Section 13.3C.2.7 and 13.3C.8.4 of this SER, the staff created Confirmatory Item 13.3-45(2) to track a textual revision to the LNP Emergency Plan that eliminates language justifying untimely augmentation of the ERO and activation of its ERFs, including the EOF.

Resolution of Confirmatory Item 13.3-45(2)

Confirmatory Item 13.3-45(2) is an applicant commitment to update the LNP Emergency Plan. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). As a result, Confirmatory Item 13.3-45(2) is now closed.

The staff finds that the LNP Emergency Plan adequately provides for timely activation and staffing of the EOF. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and Supplement 1 to NUREG-0737.

<u>Other Emergency Facilities and Equipment</u>

13.3C.8.29 *Onsite Monitoring System*

Technical Information in the Emergency Plan: [H.5]
Section H.1, "On-Site Emergency Response Facilities," of the LNP Emergency Plan states that the Digital Display System (DDS), which is the primary plant data display system for the TSC, includes SPDS data and will provide measurement and indication of Regulatory Guide (RG) 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Revision 4 variables. Section H.5, "On-Site Monitoring Systems," of the LNP Emergency Plan provides references to the LNP COL FSAR sections containing information regarding monitoring systems for geophysical phenomena, radiological conditions, plant processes, and fire and combustion products. Sections H.7, "Off-Site Radiological Monitoring Equipment," and H.8, "Meteorological Instrumentation and Procedures," describe meteorological instrumentation and monitoring systems. Section I.2, "Plant Monitoring Equipment," describes radiological monitoring. Sections I.2.1, "Radiological Monitoring," of the LNP Emergency Plan and 11.5, "Radiation Monitoring," of the AP1000 DCD and its supplements provide a description of two radiation monitoring subsystems, one for process, airborne, and effluent radiological monitoring and sampling, and one for area radiation monitoring.

Technical Evaluation: [H.5]
The staff finds that the LNP Emergency Plan adequately describes onsite monitoring systems. This is acceptable because it conforms to the guidance provided in NUREG-0654/FEMA-REP-1.

13.3C.8.30 *Provisions to Acquire Data from Offsite Sources*

Technical Information in the Emergency Plan: [H.6]
Section H.6, "Access to Data from Environmental Monitoring Systems," of the LNP Emergency Plan states that meteorological data can be acquired from the NWS when the primary system becomes unavailable. Back-up seismic data are available from the U.S. Geological Survey. Flooding data are available from the National Oceanic and Atmospheric Administration (NOAA) Hydro-Meteorological Reports. The offsite monitoring systems are described in the LNP Offsite Dose Calculation Manual (ODCM). Environmental radiological monitoring equipment includes radioiodine and particulate monitors and thermoluminescent dosimeters (TLDs). The TLDs are posted and collected in accordance with Table 1, of NRC's Branch Technical Position for the Environmental Radiological Monitoring Program, Revision 1. Section A.1, "Emergency Organization," of the LNP Emergency Plan states, in part, that the Florida DHBRC performs offsite monitoring and performs laboratory analyses of air, water, and food samples. The DHBRC also provides radiological laboratory capability, including the use of a Mobile Emergency Radiological Laboratory (MERL) and field radiological instrumentation, equipment, and supplies. Radiological laboratories, their capabilities, and expected response times are identified in Table C-1, "Radiological Laboratories – Capabilities" of Revision 0 of the LNP Emergency Plan. In RAI 13.3-21(C), the staff requested that the applicant clarify its response times in Table C-1. In response, the applicant stated Section C.3 of the LNP Emergency Plan

will be revised to refer only to the laboratories and their capabilities listed in Table C-1. These laboratories can be used by the LNP ERO during an emergency and are expected to respond as soon as resources are available.

Technical Evaluation: [H.6]
The staff finds the additional information and proposed textual revision to the emergency plan provided in response to RAI 13.3-21(C) to be acceptable and confirmed that the change referenced above was included in the LNP Emergency Plan. Therefore, the staff finds that the LNP Emergency Plan adequately describes provisions to acquire data from, or for emergency access to, offsite monitoring and analysis equipment. This is acceptable because it conforms to the guidance provided in NUREG-0654/FEMA-REP-1.

13.3C.8.31 *Offsite Radiological Monitoring Equipment*

Technical Information in the Emergency Plan: [H.7]
Section H.7, "Off-site Radiological Monitoring Equipment," of the LNP Emergency Plan stated, in part, that Progress Energy maintains the capability and resources for field monitoring with additional dosimetry as specified in the ODCM related to the Environmental Radiological Monitoring Program. TLD stations have been placed around the site in each accessible sector at various distances. Section 13.3C.8.31 of this SER provides additional information regarding the availability of radioiodine and particulate monitors for use in the environmental radiological monitoring program. Additional support can be requested from the Florida DHBRC and the MERL. Section A.1.B.3, "Florida Department of Health, Bureau of Radiation Control," of the LNP Emergency Plan states that DHBRC provides radiological laboratory capability, including mobile laboratory facilities, such as the MERL and field radiological instrumentation, equipment, and supplies to ensure measurements are properly and effectively carried out. In addition, DHBRC Standard Operating Procedures (Chapter 8 of the State Plan) includes inventories of radiation response emergency kits, radiological laboratory equipment, and mobile laboratory equipment available through the agency. In RAI 13.3-49(1)(a), the staff requested the applicant provide additional discussion in the LNP Emergency Plan regarding the availability of offsite radiological monitoring equipment (other than environmental TLDs) in the vicinity of the nuclear facility to facilitate Progress Energy's response to a radiological emergency prior to receiving support from the State of Florida DHBRC. In response, the applicant restated its capability for field monitoring prior to receiving support from the State of Florida DHBRC. In addition, the applicant stated that all other equipment needed by Progress Energy radiological monitoring teams will be obtained from the LNP emergency kits as described in Section H of the emergency plan.

Technical Evaluation: **[H.7]**
The staff finds that the LNP Emergency Plan adequately describes the offsite radiological monitoring equipment in the vicinity of the nuclear facility. This is acceptable because it conforms to the guidance provided in NUREG-0654/FEMA-REP-1.

13.3C.8.32 *Meteorological Instrumentation*

Technical Information in the Emergency Plan: [H.8]
Section H.8, "Meteorological Instrumentation and Procedures," of the LNP Emergency Plan provides a description of available meteorological instrumentation (e.g., 60.4 meter (m) meteorological tower), the availability of meteorological data in the CRs, TSCs, and EOFs, and implementing procedures for incorporating onsite meteorological data into dose assessment calculations. Section I.5, "Meteorological Information," states that as a backup for onsite capability, meteorological data can be acquired from the NWS. Instrumentation, maintenance, and calibration of meteorological equipment are also discussed in Section 2.3.3.1, "Instrumentation," of the LNP COL FSAR. Additional information regarding the availability of meteorological information and data, including atmospheric diffusion estimates, can be found in Section 2.3.3, " Onsite Meteorological Measurement Program," and Section 7.5, "Safety-Related Display Information," of this SER.

Technical Evaluation: [H.8]
The staff finds that the LNP Emergency Plan adequately describes the meteorological instrumentation and procedures, including provisions to obtain representative current meteorological information from other sources. This is acceptable because it conforms to the guidance provided in NUREG-0654/FEMA-REP-1.

13.3C.8.33 *Inspection/Inventory of Emergency Equipment*

Technical Information in the Emergency Plan: [H.10]
Section H.9, "Emergency Equipment and Supplies," states that emergency equipment and supplies to carry out the provisions of the LNP Emergency Plan are specified in emergency plan administrative procedures. Appendix 5, "List of Emergency Plan Supporting Procedures," identifies an administrative procedure titled, "Emergency Response Facilities and Equipment," that supports this section of the LNP Emergency Plan. Section H.9 also states that provisions have been made to inspect, inventory, and operationally check emergency equipment/instruments once each calendar quarter and after drills or an actual emergency. Sufficient reserves of instruments/equipment are provided to replace those that are removed from emergency kits for calibration or repair. Calibration of instruments has been established by intervals recommended by instrument suppliers, or as required by Federal regulations.

Technical Evaluation: [H.10]
The staff finds that the LNP Emergency Plan adequately describes the provisions to inspect, inventory, and operationally check emergency equipment/instruments at least once each calendar quarter and after each use. This is acceptable because it conforms to the guidance provided in NUREG-0654/FEMA-REP-1.

13.3C.8.34 *Emergency Kits*

Technical Information in the Emergency Plan: [H.11]
Table H-1, "Typical Emergency Kit Equipment/Supplies and Locations," of the LNP Emergency Plan lists emergency supplies available at each emergency facility (e.g., CRs, OSC and TSC).

Technical Evaluation: [H.11]
The staff finds that the LNP Emergency Plan adequately describes the emergency kits available at each facility. This is acceptable because it conforms to the guidance provided in NUREG-0654/FEMA-REP-1.

13.3C.8.35 *Location to Coordinate Field Monitoring Data*

Technical Information in the Emergency Plan: [H.12]
Section H.11, "Receipt of Field Monitoring Data," of the LNP Emergency Plan states that dose assessment personnel located in the EOF are designated as the central point for the receipt of offsite monitoring data and sample media analysis results. Resources exist within the organization to evaluate this information and make recommendations.

Technical Evaluation: [H.12]
The staff finds that the LNP Emergency Plan adequately establishes a central point, dose assessment personnel in the EOF, for the receipt and analysis of all field monitoring data and coordination of sample media. This is acceptable because it conforms to the guidance provided in NUREG-0654/FEMA-REP-1.

13.3C.8.36 *Facilities and Supplies for Emergency Medical Treatment*

Technical Information in the Emergency Plan: {Appendix E, Section IV.E.4}
In RAI 13.3-49(2), the staff requested that the applicant discuss in the LNP Emergency Plan whether facilities and associated supplies exist onsite for appropriate emergency first aid treatment. In response, the applicant stated that First aid facilities at LNP are designed to provide basic first responder aid to injured or ill personnel before arrival of offsite medical support. Emergency treatment areas are located in each of the units and are located at the Health Physics area near the work exits. The first aid facilities also contain personnel contamination monitoring equipment, decontamination shower facilities, and first-aid equipment. Medical equipment and supplies are available at these locations. Additional first aid facilities and supplies will be located onsite as needed. Section L.2.2, "First Aid Kits," states that first aid kits located in various areas of the site contain equipment/items necessary to treat injured personnel until offsite support is available to transport patients to the appropriate treatment centers.

Technical Evaluation: **{Appendix E, Section IV.E.4}**
The staff finds the additional information and proposed textual revisions to the emergency plan submitted in response to RAI 13.3-49(2) acceptable because it meets the applicable requirements in Appendix E to 10 CFR Part 50. The staff created **Confirmatory Item**

13.3-49(2) to track the applicant's inclusion of its response to this RAI in the LNP Emergency Plan.

Resolution of Confirmatory Item 13.3-49(2)

Confirmatory Item 13.3-49(2) is an applicant commitment to update the LNP Emergency Plan. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). As a result, Confirmatory Item 13.3-49(2) is now closed.

The staff finds that the LNP Emergency Plan adequately describes the facilities and medical supplies at the site for appropriate emergency first aid treatment. This is acceptable because it meets the requirements provided in Appendix E to 10 CFR Part 50.

13.3C.8.37 *Maintenance of Emergency Equipment and Supplies*

Technical Information in the Emergency Plan: {Appendix E, Section IV.G}
Section 1.1 of the LNP Emergency Plan states, in part, that the Emergency Plan and implementing procedures listed in Appendix 5 outline the EP Program and includes an objective for the continued maintenance of an adequate state of EP. Section 13.3C.8.34 of this SER provides discussion regarding procedures that include provisions for the inventory, inspection, calibration, and operational checks of emergency equipment/instruments. In RAI 13.3-49(3), the staff requested that the applicant provide additional discussion in the LNP Emergency Plan regarding the maintenance of emergency equipment and supplies. In response, the applicant stated that it will revise the emergency plan to include provisions for ensuring that emergency supplies are maintained up-to-date. The applicant stated, in part, that during the inspections any emergency equipment, supplies, and parts having a shelf-life will be replaced as necessary. Inventory requirements and inspections will be delineated in LNP emergency preparedness administrative procedures.

Technical Evaluation: {Appendix E, Section IV.G}
The staff finds the additional information and proposed textual revisions to the emergency plan submitted in response to RAI 13.3-49(3) acceptable because it meets the applicable requirements in Appendix E to 10 CFR Part 50. The staff created **Confirmatory Item 13.3-49(3)** to track the applicant's inclusion of its response to this RAI in the LNP Emergency Plan.

Resolution of Confirmatory Item 13.3-49(3)

Confirmatory Item 13.3-49(3) is an applicant commitment to update the LNP Emergency Plan. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). As a result, Confirmatory Item 13.3-49(3) is now closed.

The staff finds that the LNP Emergency Plan adequately describes provisions to ensure that the emergency plan, its implementing procedures, and emergency equipment and supplies are maintained up-to-date. This is acceptable because it meets the requirements in Appendix E to 10 CFR Part 50.

13.3C.8.38 *ERDS Description, Testing, and Activation*

Technical Information in the Emergency Plan: (10 CFR 50.72(a)(4) {Appendix E, Section VI}
Section A.1.b.8, "U.S. Nuclear Regulatory Commission," of the LNP Emergency Plan describes emergency notification to the NRC and communication of operational information through dedicated phone lines for the ENS and the ERDS. Section F.1, "Description of Communication Links," states that ERDS provides a real-time transfer of plant data from LNP to the NRC. Progress Energy will activate ERDS within one hour of the declaration of an alert or higher emergency classification in accordance with implementing procedures. Section N.2.a, "Communication Drills," states that Progress Energy tests communications with Federal EROs and States within the ingestion pathway EPZ quarterly. Section F.3, "Communication System Reliability," states that communications from the CRs, TSCs, and the EOF to the NRCOC is also tested quarterly. In RAI 13.3-21(D), the staff requested the applicant clarify in the LNP Emergency Plan whether the frequency of the ERDS system testing will be quarterly. In response, the applicant committed to a monthly testing frequency and to revise the LNP Emergency Plan accordingly. In RAI 13.3-50, the staff requested that the applicant clarify in the LNP Emergency Plan whether the plant data for Units 1 and 2 transmitted from the plant computer system to the NRCOC will be representative of reactor core and coolant system conditions, reactor containment conditions, radioactivity release rates, and plant meteorological data, pursuant to the requirements of Section VI.2 of Appendix E to 10 CFR Part 50. In addition, the staff requested the applicant provide a listing of the data points that will be available for transmittal from each unit at the LNP site to the NRCOC. In response, the applicant proposed to clarify in the emergency plan the availability of data to be transmitted consistent with the staff's request in this RAI. The applicant specified that data points identified in the parameters listed in Section VI.2.a(i) for pressurized water reactors will be transmitted. The applicant proposed EP ITAAC 5.2 to verify that ERDS is established and successfully completes a transfer of data between the operating units to the NRCOC. In response to supplemental RAI 13.3-44(1), the applicant changed the proposed language in EP ITAAC 5.2 to refer to plant computer systems transmitting data to the NRCOC versus operating units.

Technical Evaluation: {Appendix E, Section VI} (10 CFR 50.72(a)(4))
The staff finds the additional information and proposed textual revisions to the emergency plan provided in response to RAI 13.3-21(D) and 13.3-50 to be acceptable because they conform to the requirements of Appendix E to 10 CFR Part 50. The staff confirmed that the proposed change provided in response to RAI 13.3-21(D) was incorporated into Revision 1 to the LNP Emergency Plan. The staff created **Confirmatory Item 13.3-50** to track the applicant's inclusion of its response to this RAI in the LNP Emergency Plan. The staff finds that the LNP Emergency Plan adequately describes the activation of ERDS and meets the regulatory requirements in 10 CFR 50.72(a)(4).

<u>Resolution of Confirmatory Item 13.3-50</u>

Confirmatory Item 13.3-50 is an applicant commitment to update the LNP Emergency Plan. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). As a result, Confirmatory Item 13.3-50 is now closed.

The staff finds that the LNP Emergency Plan adequately describes the ERDS. This is acceptable because it meets the applicable requirements in Appendix E to 10 CFR Part 50. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.8.39 *ERO Augmentation at Alternate Facility*

Technical Information in the Emergency Plan: {Appendix E, Section VI.8.d}
Section H.2.3, "Alternate Emergency Response Facility," of the LNP Emergency Plan describes the alternate facility as being located in the EOF/ENC Facility that will serve as a short-term location for ERO members to assemble and activate in the event that access to the onsite emergency response facilities is not possible with minimal equipment available for its operation. In earlier versions of the LNP Emergency Plan, Section H.2.3 stated that the alternate facility will include at a minimum communication links with the EOF, control room and security; the capability to notify offsite response organizations if the EOF is not performing this action; and computer links to the site to access plant data. In RAI 13.3-62, the staffed requested that the applicant clarify in the LNP Emergency Plan whether the alternate facility is capable of being staffed for an extended period of time with adequate equipment to support its operation; whether the purpose of the alternate facility is to stage ERO personnel to support rapid response to the LNP site to limit or mitigate site damage, or the potential for a radiological release; and whether general drawings and system information will be used to support engineering assessment activities to include damage control team planning and preparation. In response to RAI 13.3-62, the applicant provided the requested clarification as described above.

Technical Evaluation: {Appendix E, Section VI.8.d}
The applicant has designated an alternate facility (EOF) outside of the 10-mile plume exposure pathway EPZ and within 20 miles of the LNP site (TSCs) with capabilities similar to the EOF. By letter dated November 8, 2012, from PEF to the NRC, the applicant provided its response to address the Final Rule on Enhancements to Emergency Preparedness Regulation effective November 23, 2011. In regard to implementation of the EP rule and designation of an alternate facility for use during ERO augmentation should onsite facilities become unavailable, the staff requested additional clarification of the LNP Emergency Plan in RAI 13.3-62 as described above. The staff finds the additional information and proposed textual revisions to the emergency plan proposed in RAI 13.3-62 to be acceptable because they conform to the requirements of Appendix E to 10 CFR Part 50, and the guidance in NSIR/DPR-ISG-01. The applicant identified an EPIP, "Activation and Operation of the Alternate Emergency Response Facility" in Appendix 5 of the LNP Emergency Plan. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). Therefore, the staff finds that the LNP Emergency Plan adequately describes the ERO augmentation at an alternate facility to support rapid response to the LNP site in the event of an emergency, meets the regulatory requirements in

10 CFR 50.47(b)(8) and Appendix E to 10 CFR Part 50, and conforms to the guidance in NSIR/DPR-ISG-01.

13.3C.8.40 *Conclusion*

On the basis of its review of the LNP Emergency Plan as described above for Emergency Facilities and Equipment, the staff concludes that information provided in the LNP Emergency Plan regarding emergency facilities and equipment is acceptable and meets the requirements of 10 CFR 50.47(b)(8) because it conforms with the guidance in Evaluation Criterion H of NUREG-0654/FEMA-REP-1, NSIR/DPR-ISG-01, Supplement 1 to NUREG-0737, and NUREG-0696, and meets the applicable requirements of Appendix E to 10 CFR Part 50 as described above.

13.3C.9 Accident Assessment

13.3C.9.1 *Regulatory Basis*

In determining whether the proposed emergency plan met the applicable regulatory requirements in 10 CFR 50.47(b)(9), the staff evaluated it against the detailed evaluation criteria in NUREG-0654/FEMA-REP-1. The staff also evaluated the proposed emergency plan against applicable regulatory requirements related to the area of "Accident Assessment" in Appendix E to 10 CFR Part 50 and 10 CFR 50.34.

13.3C.9.2 *Initiating Conditions for Emergency Classes*

Technical Information in the Emergency Plan: [I.1]
Section I, “Accident Assessment,” of the LNP Emergency Plan describes the methods, systems, and equipment available for assessing and monitoring actual or potential offsite consequences of a radiological emergency. Section I states that the use of the equipment described in this section of the emergency plan during an emergency is detailed in EPIPs. Section I.1, “Parameters Indicative of Emergency Conditions,” of the LNP Emergency Plan states that plant system and effluent parameter values that would be observed in off-normal situations are described in Section D, “Emergency Classification System,” of the emergency plan. In addition, Section I.1 states that emergency response procedures and implementing procedures include methods for quickly assessing plant system and effluent parameter values, and classifying the emergency condition. Section I.2, “Plant Monitoring Systems,” of the emergency plan describes the monitoring systems that would be available for assessing plant conditions in an emergency.

Technical Evaluation: [I.1]
The staff finds that the LNP Emergency Plan adequately identifies plant system and effluent parameter values characteristic of a spectrum of off-normal conditions and accidents, and identifies the plant parameter values or other information which correspond to the initiating conditions for each emergency class. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.9.3 *Capability to Continuously Assess an Accident*

Technical Information in the Emergency Plan: [I.2] (10 CFR 50.34(f)(2)(xvii))
Section B.5.1.i, "Accident Assessment Coordinator," of the LNP Emergency Plan states that the Accident Assessment Coordinator is located in the TSC, reports to the EC-TSC, and is responsible for coordination of accident assessment team strategies to support accident mitigation. Section B.5.1.e, "Technical Support Coordinator," in Revision 1 of the LNP Emergency Plan states that the Technical Support Coordinator is located in the EOF, reports to the EOF Director, and is responsible for assisting the TSC Accident Assessment Team in identifying accident mitigation activities and monitoring critical safety system functions. In Revision 6 of the LNP Emergency Plan this position is filled by the Shift Technical Advisor (STA). Section F.1.d, "Description of Communication Links," describes separate conference-line phone systems available between the CRs, TSCs, and EOF to be used to communicate accident assessment, dose assessment, and emergency plant status information. Section 13.3C.8.2 of this SER provides additional information regarding one of the key TSC functions, which is to assist the CR in accident assessment. Section I.2, "Plant Monitoring Systems," of the LNP Emergency Plan states that initial values and continuing assessment of plant conditions through the course of an emergency may rely on reactor coolant sample results, radiation and effluent monitors, in-plant iodine instrumentation, and containment radiation monitoring. The LNP Emergency Plan provides reference to various sections of the FSAR, including Section 9.3.3, "Primary Sampling System," and 11.5, "Radiation Monitoring," which incorporates by reference the related sections of the AP1000 DCD and its supplements, and describe provisions for obtaining samples under accident conditions and radiation monitoring systems. Section I.2.1, "Radiological Monitoring," states that the RMS provides plant effluent monitoring, process fluid monitoring, airborne monitoring, and continuous indication of the radiation environment in plant areas where such information is needed. A listing of plant and sampling locations is also provided for each monitor type that is part of the RMS. Additional discussion related to Section H.5, "Onsite Monitoring Systems" of the LNP Emergency Plan and data, including SPDS and RG 1.97 variables, that can be retrieved in the CRs and TSC for accident assessment is located in Section 13.3C.8.30 of this SER.

Additional discussion regarding meteorological instrumentation and data that is digitally displayed in the CRs, TSCs, and EOF can be found in Section H.8 of the LNP Emergency Plan and 13.3C.8.33 of this SER. Additional information regarding the availability of meteorological information and data, including atmospheric diffusion estimates, can be found in Section 2.3.3, "Onsite Meteorological Measurement Program," and Section 7.5, "Safety-Related Display Information," of this SER.

Section I.6, "Determination of Release Rates and Projected Dose Rates," of the LNP Emergency Plan, states that there are implementing procedures which establish processes for estimating the extent of fuel damage. Section I.9, "Measuring Radioiodine Concentrations," describes the capabilities of field monitoring teams to assess radioiodine concentrations in air downwind of the site. The field monitoring equipment is capable of measuring concentrations as low as 1×10-7 $\mu Ci/cm^3$. The applicant proposed EP ITAAC 8.1 to verify that the means exist to provide initial and continuing radiological assessment throughout the course of an accident through the plant computer or communications with the CR.

Section I.4.1, "On-site Dose Assessment," of the LNP Emergency Plan states that implementing procedures provide procedural guidance for the following assessment activities: assessment and quantification of actual and potential releases; obtaining samples; performing isotopic analysis (evaluation of effluents); sampling and analyzing the containment atmosphere for radionuclide concentration under accident conditions; sampling and analyzing the containment atmosphere for hydrogen content under accident conditions; and estimating the types and quantities of radioactive material available for release. Additional discussion regarding onsite dose assessment is in Section 13.3C.9.6 of this SER. Appendix 5 of the LNP Emergency Plan provides reference to three EPIPs titled, "Core Damage," "Off-site Radiological Monitoring," and "Dose Assessment," that support and implement Section I of the LNP Emergency Plan.

Technical Evaluation: [I.2] (10 CFR 50.34(f)(2)(xvii))
The staff finds that the LNP Emergency Plan adequately describes the capability and resources to provide initial values and continuing assessment of plant conditions through the course of an accident. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and meets the requirements of 10 CFR 50.34(f)(2)(xvii). The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.9.4 *Capability to Determine Source Term*

Technical Information in the Emergency Plan: [I.3.a] {Appendix E, Section IV.E.2}
Section I.3, "Determination of Source Term and Radiological Conditions," of the LNP Emergency Plan states that implementing procedures provide the means for interpreting measured parameters (such as containment monitor readings) to determine source terms (such as the radioactive material available for release from containment). The applicant proposed EP ITAAC 8.2 to demonstrate that the means exist to determine the source term of releases of radioactive material within plant systems, and the magnitude of the release of radioactive materials based on plant system parameters and effluent monitors.

Technical Evaluation: [I.3.a] {Appendix E, Section IV.E.2}
The staff finds that the LNP Emergency Plan adequately establishes methods and techniques to be used for determining the source term of releases of radioactive material within plant systems based on plant system parameters and effluent monitors. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and meets the requirements of Appendix E to 10 CFR Part 50. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.9.5 *Capability to Determine the Magnitude of a Radiological Release*

Technical Information in the Emergency Plan: [I.3.b] {Appendix E, Section IV.B}
Section I.3, "Determination of Source Term and Radiological Conditions," of the LNP Emergency Plan states that the magnitude of the release can be determined from plant system

parameters and effluent monitor readings using implementing procedures. The applicant proposed EP ITAAC 8.2 to demonstrate that the means exist to determine the source term of releases of radioactive material within plant systems, and the magnitude of the release of radioactive materials based on plant system parameters and effluent monitors.

Technical Evaluation: [I.3.b] {Appendix E, Section IV.B}
The staff finds that the LNP Emergency Plan adequately establishes methods and techniques to be used for determining the magnitude of releases of radioactive material within plant systems based on plant system parameters and effluent monitors. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and meets the requirements of Appendix E to 10 CFR Part 50. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.9.6 *Relationship Between Effluent Monitors and Exposure*

Technical Information in the Emergency Plan: [I.4] {Appendix E, Section IV.A.4} {Appendix E, Section IV.B}
Section I.4, "Relationship between Effluent Monitor Reading and Exposure and Contamination Levels," of the LNP Emergency Plan describes dose assessment procedures which include the relationship between effluent monitor readings, and onsite and offsite exposures and contamination estimates for various meteorological conditions. Sections I.4.1, "On-Site Dose Assessment," and I.4.2, "Off-Site Dose Assessment," of the LNP Emergency Plan describe the emergency dose assessment program used at LNP both onsite and offsite. Information provided includes dose and dose rate determinations based on plant effluent monitors, and contamination estimates based on deposition assumptions and meteorological conditions. Section I.4.1 of the emergency plan describes the process by which onsite radiological surveys are performed and by whom. Survey results are forwarded to the TSCs for evaluation and assessment. The Radiation Controls Coordinator will assess survey results and advise the EC of in-plant radiological conditions. The need for additional or continuing surveys is established by the EC. Specific instructions for in-plant radiological surveys are provided in implementing procedures. In some instances, additional sampling and analysis are required for quantitative assessment of potential source terms or the magnitude of a release. Section 13.3C.9.3 of this SER provides additional discussion regarding the contents of implementing procedures on this topic. Section I.4.2 of the emergency plan states, in part, that an EPIP will be used to assess the dose to personnel downwind of an accidental radioactive release. The EPIP will account for specific criteria such as meteorological regimes (e.g., seabreeze) and other topographical effects so the dose projections will be representative of the LNP site. The EPIP will provide Operations staff (including the STA) with a rapid method of determining the magnitude of a radioactive release from LNP during an accident condition. The EPIP contains a series of tables that will be used with meteorological and radiological data displayed in the CR, to quickly generate offsite dose information. The EPIP will also provide dose assessment personnel guidance to determine the magnitude of the radioactive release and cumulative dose by distance and sector to aid in the formulation of PARs.

Section B.5.2, “Off-Site Emergency Response Organization,” describes the offsite ERO and states that the Radiation Controls Manager is responsible for providing direction for dose assessment, and the EOF Director has the responsibility for coordinating dose assessment. The EOF Director is also responsible for direct interface with offsite authorities. The applicant proposed EP ITAAC 8.3 to test that response personnel can 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.

Additional information regarding the availability of meteorological information and data, including atmospheric diffusion estimates, can be found in Section 13.3C.9.7, “Meteorological Information,” Section 2.3.3,“ Onsite Meteorological Measurement Program,” and Section 7.5, “Safety-Related Display Information,” of this SER.

Technical Evaluation: [I.4] {Appendix E, Section IV.A.4} {Appendix E, Section IV.B}
The staff finds that the LNP Emergency Plan adequately establishes the relationship between effluent monitor readings and onsite and offsite exposures and contamination for various meteorological conditions. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and meets the applicable requirements of Appendix E to Part 50. The staff’s evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.9.7 *Meteorological Information*

Technical Information in the Emergency Plan: [I.5.]
Section I.5, “Meteorological Information,” of the LNP Emergency Plan states that a permanent meteorological monitoring station is located within the Exclusion Area Boundary. It records the data that are required for performing dose projections and this information is presented in the CR, TSC, and EOF. Progress Energy has the capability to access the NWS in Tallahassee, Florida on a 24-hour basis to provide backup data should the onsite system fail. Sections E.2, “Message Content,” and E.3, “Follow-up Messages to Off-site Authorities,” of the LNP Emergency Plan states that the contents of initial and follow-up emergency messages established with State and local governments include basic meteorological data. Section F.1.b, “Description of Communication Links,” states that communications with State/county governments within the EPZs include weather service forecast offices. Section F.1.C states, in part, that the HPN and PMCL are separate telephone lines dedicated for communicating radiological and meteorological conditions, assessments, trends, and protective measures with the NRC. HPN and PMCL lines are located in the TSCs and EOF. The applicant proposed EP ITAAC 8.4 in response to supplemental RAI 13.3-32 to test the capability to display meteorological parameters (e.g., wind speed – 10 m and 60 m, wind direction – 10 m and 60 m, delta-temperature) in the TSC and CR in the format needed for the use in the appropriate EPIP. In supplemental RAI 13.3-51, the staff requested that the applicant revise EP ITAAC 8.4 (proposed in response to supplemental RAI 13.3-32) to include a test of the capability to display meteorological data in the EOF consistent with Section I.5 of the LNP Emergency Plan. The applicant revised EP ITAAC 8.4 as requested by the staff. Additional discussion regarding the transfer of plant operational data from LNP via ERDS to the NRCOC can be found in

Section 13.3C.8.39 of this SER. Additional information regarding the availability of meteorological information and data, including atmospheric diffusion estimates, can be found in Section 13.3C.8.33, "Meteorological Instrumentation," Section 2.3.3, " Onsite Meteorological Measurement Program," and Section 7.5, "Safety-Related Display Information," of this SER.

Technical Evaluation: [I.5]
The staff finds that the LNP Emergency Plan adequately describes the capability of acquiring and evaluating meteorological information. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.9.8 *Projecting Dose When Instrumentation is Inoperable*

Technical Information in the Emergency Plan: [I.6]
Section I.6, "Determination of Release Rates and Projected Doses," of the LNP Emergency Plan states that implementing procedures establish processes for estimating release rates and projected doses in the event that associated instrumentation is off-scale or inoperable. Procedures include provisions for estimating releases based on field monitoring data and surrogate instrumentation, and methods to estimate the extent of fuel damage. The applicant proposed EP ITAAC 8.5 to verify that a test will be performed of the capabilities to determine the release rate and projected doses if the instrumentation used for assessment is off-scale or inoperable. Procedures related to core damage and dose assessment are identified in Appendix 5 of the LNP Emergency Plan.

Technical Evaluation: [I.6]
The staff finds that the LNP Emergency Plan adequately establishes the methodology for determining the release rate/projected doses if the instrumentation used for assessment are off-scale or inoperable. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.9.9 *Field Monitoring Capability*

Technical Information in the Emergency Plan: [I.7]
Section I.4.1, "On-site Dose Assessment," of the LNP Emergency Plan states that the Radiological Monitoring Team performs activities to determine radioactive levels at the site boundary, and beyond, as soon as possible following an accidental release in accordance with implementing procedures. Conditions at the time of an emergency may dictate specific areas where intense radiological monitoring efforts will be required. Upon activation and preparation of the Radiological Monitoring Team, the Radiation Controls Coordinator and EC will determine specific areas to be monitored. The Radiological Monitoring Team has sole responsibility for plume monitoring until such time as the State Monitoring Teams arrive and assume this responsibility for areas beyond the site boundary. Results of surveys are appropriately recorded and reported to the TSCs via portable transceiver. The TSCs transmit the results to the EOF for coordination of analysis, as appropriate, with State survey results. Section I.7, "Field Monitoring

Capability," of the LNP Emergency Plan states that radiological surveys and monitoring of the offsite environs are coordinated by the State and conducted by the State Radiological Emergency Team. Field teams have access to the MERL, which is equipped to provide radiological laboratory services and can arrive at the EOF within two hours of notification. Equipment available to the field team by the MERL is provided in Table I-1, "Mobile Emergency Radiological laboratory – Typical Instrumentation and Equipment." Section H.7, "Off-Site Radiological Monitoring Equipment," provides additional information related to the MERL and State capabilities, and states that LNP has monitoring capabilities normally associated with the environmental monitoring program, such as environmental TLDs. The applicant proposed EP ITAAC 8.6 to ensure a test will be performed to demonstrate the capabilities for field monitoring teams to be dispatched and locate and monitor a radiological release within the plume exposure pathway EPZ.

Technical Evaluation: [I.7]
The staff finds that the LNP Emergency Plan adequately describes the capability and resources for field monitoring within the plume exposure EPZ. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.9.10 *Capability to Rapidly Assess Radiological Hazards*

Technical Information in the Emergency Plan: [I.8.]
Section I.8, "Assessment Hazards through Liquid or Gaseous Release Pathways," of the LNP Emergency Plan states that Progress Energy trains, designates, equips, dispatches, and coordinates, both radiological and environmental field teams in accordance with the LNP Emergency Plan. Field teams maintain the capability to perform sampling of offsite media samples to assess the potential magnitude and locations of radiological hazards. Additional discussion regarding the capability and resources for rapidly assessing radiological hazards can be found in Section 13.3C.9.6 and 13.3C.9.9 of this SER. The applicant proposed EP ITAAC 8.7 to ensure a drill or exercise is conducted that demonstrates the capability to activate field teams, which will make a rapid assessment of the actual or potential magnitude, and locations of radiological hazards through simulated liquid or gaseous release pathways. A qualified field team is capable of being notified, activated, briefed and dispatched from the EOF during a radiological release scenario. The team demonstrates conformance with procedural guidance for team composition, use of monitoring equipment, communication from the field, and locating specific sampling locations.

Technical Evaluation: [I.8]
The staff finds that the LNP Emergency Plan adequately describes methods, equipment, and expertise to rapidly assess radiological hazards. This is acceptable because they conform to the guidance in NUREG-0654/FEMA-REP-1. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.9.11 *Capability to Measure Radioiodine Concentrations in Air*

Technical Information in the Emergency Plan: [I.9]
Section I.9, "Measuring Radioiodine Concentrations," of the LNP Emergency Plan states that field teams are equipped with the capability to detect and measure radioiodine concentrations as low as 1×10-7 μCi/cm^3 (microcuries per cubic centimeter) in the vicinity of the site. Interference from background radiation and noble gas is minimized by moving to a low-background position before analyzing a sample cartridge. The collected air sample is measured by hand-held survey meter as an initial check of the projection derived from the plant data to determine if significant quantities of elemental iodine have actually been released. The applicant proposed EP ITAAC 8.8 to ensure a test will be performed of the capabilities to detect and measure radioiodine concentrations in air in the plume exposure EPZ, as low as 10^{-7} μCi/cc under field conditions.

Technical Evaluation: [I.9]
The staff finds that the LNP Emergency Plan adequately describes a capability to detect and measure radioiodine concentrations in air in the plume exposure EPZ as low as 10^{-7} μCi/cc under field conditions. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.9.12 *Means to Relate Various Parameters to Dose Rates*

Technical Information in the Emergency Plan: [I.10]
Section I.10, "Relating Measured Parameters to Dose Rates," of the LNP Emergency Plan states that implementing procedures establish the means for relating measured parameters to dose rates for key radioisotopes. These procedures also set the methods for determining projected dose based on projected and actual dose rates. The applicant proposed EP ITAAC 8.9 to ensure a test will be performed of the capabilities to estimate integrated dose from the projected and actual dose rates, and for comparing these estimates with the Environmental Protection Agency Protection Action Guidelines. Appendix 5 provides reference to an EPIP for making dose assessments.

Technical Evaluation: **[I.10]**
The staff finds that the LNP Emergency Plan adequately establishes means for relating the various measured parameters (e.g., contamination levels, water and air activity levels) to dose rates for key isotopes and gross radioactivity measurements. The LNP Emergency Plan also adequately describes provisions for estimating integrated dose from the projected and actual dose rates, and for comparing these estimates with the protective action guides. The detailed provisions are described in separate procedures. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.9.13 *Conclusion*

On the basis of its review of the LNP Emergency Plan as described above for Accident Assessment, the staff concludes that the information provided in the LNP Emergency Plan regarding accident assessment is acceptable and meets the requirements of 10 CFR 50.47(b)(9) because it conforms with the guidance in Evaluation Criterion I of NUREG-0654/FEMA-REP-1 and meets the applicable requirements of Appendix E to 10 CFR Part 50 and 10 CFR 50.34 as described above.

13.3C.10 Protective Response

13.3C.10.1 *Regulatory Basis*

In determining whether the proposed emergency plan met the applicable regulatory requirements in 10 CFR 50.47(b)(10) for protective response, the staff evaluated it against the detailed evaluation criteria in NUREG-0654/FEMA-REP-1 and NSIR/DPR-ISG-01. In addition, the staff evaluated the proposed emergency plan against the applicable requirements in Appendix E to 10 CFR Part 50.

13.3C.10.2 *Warning Onsite Personnel*

Technical Information in the Emergency Plan: [J.1.a-d]
Section J, "Protective Response," of the LNP Emergency Plan describes the protective actions that have been developed to limit radiation exposure to site personnel and the general public in the event of an accident at the site. In Revision 6 of the LNP Emergency Plan, protective actions developed to protect onsite personnel during a hostile action are also addressed. Section J.1, "On-Site Notification," states that methods have been established, in a timely manner, to notify all individuals within the LNP site boundary of an emergency condition requiring individual action. These individuals may include LNP personnel not having emergency assignments; visitors; contractors and construction personnel; and other individuals in the public access areas, on or passing through the site or within the owner controlled area. Notifications will be made to individuals within the PA primarily through use of the plant's public address system and audible warning systems. In areas of high noise or other areas where these systems may not be audible, other measures (e.g., visible warning signals or personal notifications) may be used. Notification to personnel located outside of the PA are through audible warnings provided by warning systems and the activities of the Security Force (e.g., vehicle-mounted public address systems) or local law enforcement, as needed. LNP provides information regarding the meaning of the various warning systems and appropriate response actions through plant training programs, visitor orientation, escort instructions, posted instructions, or within the content of audible messages. In RAI 13.3-23(A), the staff requested the applicant clarify the time required to warn or advise onsite individuals of an emergency. The applicant's response stated that personnel and others within the LNP site boundary will be notified in a timely manner (about 15 minutes). In response to RAI 13.3-44(3), the applicant proposed EP ITAAC 9.1 and 12.1.1.B.3 to ensure a test will be performed to demonstrate the

capability to warn and advise onsite individuals of emergency conditions in a timely manner (about 15 minutes) in accordance with the LNP Emergency Plan.

Technical Evaluation: [J.1.a-d]
The staff finds the clarification and textual revision to the emergency plan provided in response to RAI 13.3-23(A) to be acceptable because they conform to NUREG-0654/FEMA-REP-1. The staff confirmed that the proposed changes provided in response to this RAI were incorporated into Revision 6 to the LNP Emergency Plan. The staff also confirmed that the proposed changes to EP ITAAC 9.1 and 12.1.1.B.3 were incorporated into Revision 2 to Part 10 of the COL application. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER. Therefore, the staff finds that the LNP Emergency Plan adequately establishes the means and time required to warn or advise onsite individuals and individuals who may be in areas controlled by the operator, including employees not having emergency assignments, visitors, contractor and construction personnel, and other persons who may be in the public access areas on or passing through the site or within the owner controlled area. This is acceptable because it meets the guidance in NUREG-0654/FEMA-REP-1.

13.3C.10.3 *Evacuation Routes for Onsite Personnel*

Technical Information in the Emergency Plan: [J.2]
Section J.2, "Evacuation Routes and Transportation," of the LNP Emergency Plan states that in the event of an evacuation, onsite personnel will be evacuated to a remote offsite assembly area. In RAI 13.3-23(B), the staff requested that the applicant provide additional information regarding the location of the pre-designated main assembly area or alternate remote offsite assembly area to be used when evacuating onsite personnel in the event of an emergency. In its response, the applicant stated that since each emergency situation can be unique in regards to radiological, meteorological, plant, and security conditions, implementing procedures will provide flexibility on assignment of assembly areas, both onsite and offsite, for evacuating onsite personnel. The applicant stated that the LNP Training Building is the primary onsite, pre-designated assembly area located outside of the PA for evacuating non-essential personnel, while the EOF is the primary offsite assembly area and alternate remote offsite assembly area. Section J.2 states that evacuation of non-essential personnel could be required from either the PA or from the entire owner-controlled area. Section J.2.a of the emergency plan states that non-essential personnel (e.g., personnel not on the ERO or assisting with the emergency) shall evacuate using their respective personal transportation and follow established evacuation routes. Section J.2.d indicates that personnel without transportation will arrange for rides with others. Local evacuations for radiation control and fire protection are conducted in accordance with site procedures. Section J.10, "Protective Measures Implementation," states that evacuation routes are illustrated in Figure A.6-2, "Levy Evacuation Routes and Shelters." Appendix 5 includes an implementing procedure titled, "Evacuation and Accountability," that supports and implements Section J of the LNP Emergency Plan.

Technical Evaluation: [J.2]
The staff finds the clarifications and textual revisions to the emergency plan provided in response to RAI 13.3-23(B) to be acceptable because they clarify the locations of pre-

designated and alternate remote assembly areas, and the response conforms to the guidance in NUREG-0654/FEMA-REP-1. The staff confirmed that the changes provided in response to this RAI were included in to the LNP Emergency Plan. Therefore, the staff finds that the LNP Emergency Plan adequately describes the provisions for evacuation routes and transportation for onsite individuals to a suitable location. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.10.4 *Radiological Monitoring of Onsite Personnel*

Technical Information in the Emergency Plan: [J.3]
Section J.2, "Evacuation Routes and Transportation," of the LNP Emergency Plan states that evacuating personnel may be monitored through portal monitors as they leave the PA or by portable friskers in the evacuation monitoring area based on the current situation. If conditions warrant, they will reassemble at an offsite area, the EOF or other suitable area, until remote monitoring and decontamination stations are established. Section J.3, "Personnel Monitoring and Decontamination," states that if a radiological release has occurred or is in progress, a representative sample of vehicles will be monitored for contamination prior to dismissing personnel to relocation sites. Progress Energy has established relocation sites for personnel monitoring. Contamination monitoring of personnel, vehicles, and personal property arriving at the assembly area is directed by the Emergency Coordinator when a possibility exists that individuals may have become contaminated before or during the LNP site evacuation. Based on monitoring results, personnel will be cleared or dispatched to an offsite vehicle wash-down station. If it is necessary to dispatch personnel offsite, Progress Energy will coordinate this process with county emergency management personnel. The applicant proposed EP ITAAC 9.2 to demonstrate the capability to radiologically monitor people evacuated from the site. Equipment is available, and personnel have been assigned and trained to procedures that are approved and in place to accomplish this activity.

Technical Evaluation: [J.3]
The staff finds that the LNP Emergency Plan adequately provides for radiological monitoring of people evacuated from the site. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.10.5 *Evacuation of Non-essential Onsite Personnel*

Technical Information in the Emergency Plan: [J.4]
Section J.4, "Non-essential Personnel Evacuation and Decontamination," of the LNP Emergency Plan states that evacuation of non-essential personnel in the event of a "site area emergency" or "general emergency" is described in Section J.2, "Evacuation Routes and Transportation." Appropriate equipment and supplies are provided from the facility to the assembly areas to facilitate contamination monitoring. All members of the public who are onsite must be evacuated if there is a possibility of individual exposures. When assembly is requested, members of the general public will proceed to the pre-designated assembly area(s); and non-essential personnel will stop work, shut down potentially hazardous equipment, and

proceed to the pre-designated assembly area(s). Assembly area accountability will take place and the results will be reported to the EC when requested. Members of the general public and LNP personnel will remain in assembly area(s) until instructed to return to work, to shelter in the assembly areas, or to evacuate. Section J.2 states that non-essential personnel exiting the site will be directed to proceed either to their homes, if no radiological release has occurred, or to an assembly area, such as the EOF or other suitable location, until county monitoring and decontamination stations are in place. Non-essential personnel exiting the site may also be monitored through portal monitors as they exit the PA or by portable friskers in the evacuation monitoring area based on the situation.

Technical Evaluation: [J.4]
The staff finds that the LNP Emergency Plan adequately provides for the evacuation of onsite non-essential personnel in the event of a “site area emergency” or “general emergency” and describes a decontamination capability. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.10.6 *Onsite Personnel Accountability*

Technical Information in the Emergency Plan: [J.5]
Section J.5, “Personnel Accountability,” of the LNP Emergency Plan states that all personnel within the PA will be evacuated at a Site Area or General Emergency classification, or earlier if deemed necessary by the EC. Any remaining personnel within the PA will be accounted for within 30 minutes, and continuously thereafter during the emergency. Missing individuals will be identified by Security. Additional discussion regarding a delay in accountability due to a security-based event and onsite personnel protective decision making by the EC can be found in Section 13.3C.17.4 of this SER. Emergency procedures describe the accountability methodology. Search procedures will be implemented to locate unaccounted persons. Procedures related to evacuation and accountability are identified in Appendix 5, “List of Emergency Plan Supporting Procedures,” of the LNP Emergency Plan.

Technical Evaluation: [J.5]
The staff finds that the LNP Emergency Plan adequately describes the capability to account for all individuals onsite at the time of an emergency and ascertain the names of missing individuals within 30 minutes of its start, accounting for all onsite individuals continuously thereafter. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.10.7 *Protection for Personnel Remaining or Arriving Onsite*

Technical Information in the Emergency Plan: [J.6.a-c]
Section J.6, “Protective Measures,” of the LNP Emergency Plan states that LNP distributes protective equipment and supplies, as needed, to personnel remaining or arriving onsite during an emergency to control radiological exposure and contamination. The equipment and supplies include respiratory protection for individuals, protective clothing, and potassium iodide tablets for protection against radioactive iodine, if warranted. Other engineering controls (e.g., ventilation in TSCs and CRs) are used, as well, to control personnel exposure to radioactive material in the

air. Revision 6 of the LNP Emergency Plan addresses protective measures for onsite workers in the event of a hostile action. The protective actions are described in LNP EPIPs.

Technical Evaluation: [J.6.a-c]
The staff finds that the LNP Emergency Plan adequately provides for the use of individual respiratory protection, protective clothing, and radioprotective drugs (e.g., individual thyroid protection) including under hostile conditions. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and NSIR/DPR-ISG-01.

13.3C.10.8 *Recommending Protective Actions*

Technical Information in the Emergency Plan: [J.7]
Section J.7, "Protective Action Recommendations and Bases," of the LNP Emergency Plan provides discussion regarding who is responsible for recommending offsite protective actions in an emergency, including communications with State and local government authorities. The EOF Director or the EC (if the EOF is not activated) is responsible for making protective action recommendations (PARs) to the State and affected counties within 15 minutes of both declaring a general emergency and making any change in the PARs. Specific PARs, tied to plant and meteorological conditions, are provided in an implementing procedure. This guidance is based on Supplement 3 to NUREG-0654/FEMA-REP-1, "Criteria for Protective Action Recommendations for Severe Accidents." Appendix 5 of the LNP Emergency Plan includes reference to an EPIP titled, "Protective Action Recommendations." Section J.7 further states that public PARs are based on plant conditions, estimated offsite doses, or some combination of both. The EALs correspond to the projected dose to the population-at-risk and are determined consistent with the methodology discussed in NEI 07-01. Offsite dose projections are compared to the Protective Action Guides shown in Table J-1, which are derived from USEPA 400-R-92-001. Section J.7 states that sheltering may be appropriate when a release is controlled or terminated, or when conditions exist, such as severe weather, that would make evacuation dangerous. In addition, recommendations are made for use of potassium iodide by the public that are consistent with approved strategies.

Technical Evaluation: [J.7]
The staff finds that the LNP Emergency Plan adequately establishes a mechanism for recommending protective actions to the appropriate State and local authorities. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.10.9 *Evacuation Time Estimates*

Technical Information in the Emergency Plan: [J.8] {Appendix E, Section IV.1}
Section J.8, "Evacuation Time Estimates," of the LNP Emergency Plan states that an ETE study was performed for the LNP site, consistent with guidance in NUREG-0654, Appendix 4, "Evacuation Time Estimates within the Plume Exposure Pathway Emergency Planning Zone," and NUREG/CR-6863, "Development of Evacuation Time Estimate Studies for Nuclear Power Plants." A summary of the ETEs are provided in Table J-2, "10-Mile Emergency Planning Zone [EPZ] Evacuation Time Estimates (100 Percent) (Hr:Min)," of the emergency plan. Details

regarding this study are provided in Appendix 6, “Evacuation Time Estimate Study Summary,” of the LNP Emergency Plan, and are reviewed separately in Section 13.3C.18 of this SER. Figure A6-1, “EPZ Population Distribution (by Subzone),” presents a distribution of the population within the 10-mile plume exposure pathway EPZ.

By letter dated November 8, 2012, from PEF to the NRC, the applicant provided its response to address the Final Rule on Enhancements to Emergency Preparedness Regulation effective November 23, 2011. In regard to implementation of the EP rule pertaining to the evacuation time estimates, the applicant revised the emergency plan Section J.8 (Revision 5) to clarify the intended use of the ETE study and subsequent updates by the LNP organization and State and local authorities, including timing requirements for periodic updates by LNP and submittal to the NRC for review.

Technical Evaluation: [J.8] {Appendix E, Section IV.1}
The staff finds that the LNP Emergency Plan adequately provides time estimates for evacuation within the plume exposure EPZ. This is acceptable because it meets the guidance in NUREG-0654/FEMA-REP-1 and requirements in Appendix E to 10 CFR Part 50.

By letter dated November 8, 2012, from PEF to the NRC, the applicant provided its response to address the Final Rule on Enhancements to Emergency Preparedness Regulation effective November 23, 2011. The staff finds the revisions to the LNP Emergency Plan (Revision 5) regarding evacuation time estimate updates, including timing requirements, information sharing with State and local authorities, and submittal to the NRC to be acceptable because they conform to the guidance in NSIR/DPR-ISG-01 and meet the requirements of Appendix E to 10 CFR Part 50. At this time, revision to the ETE Study contained in Appendix 6 is not warranted.

The staff’s detailed evaluation of the LNP ETE Report is addressed in Section 13.3C.18 of this SER.

13.3C.10.10 *Plans to Implement Protective Measures*

Technical Information in the Emergency Plan: [J.10.a]
Section J.10, “Protective Measures Implementation,” of the LNP Emergency Plan states that Figure A6-2, “Levy Evacuation Routes and Shelters,” provides a map of the evacuation routes, reception centers, and shelters. Pre-selected radiological sampling and monitoring points are identified in implementing procedures. Procedures related to PARs and evacuation are identified in Appendix 5 to the LNP Emergency Plan.

Technical Evaluation: [J.10.a]
The staff finds that the LNP Emergency Plan adequately addresses evacuation routes, evacuation areas, pre-selected radiological sampling and monitoring points, relocation centers in host areas, and shelter areas. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

Technical Information in the Emergency Plan: [J.10.b]
Section J.10, "Protective Measures Implementation," of the LNP Emergency Plan states that Appendix 6, "Evacuation Time Estimate Study Summary," provides tables and maps of the plume exposure pathway EPZ illustrating population distribution. Figure A6-1, "Resident Population within the 10-Mile EPZ," provides resident population in sector format.

Technical Evaluation: [J.10.b]
The staff finds that the LNP Emergency Plan includes figures that adequately show population distribution around the nuclear facility. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

Technical Information in the Emergency Plan: [J.10.c]
Section J.10, "Protective Measures Implementation," of the LNP Emergency Plan states that warnings to the public within the EPZ are the responsibility of the State and local officials. The primary method of warning the public is by the use of the ANS. Section E.5, "Instructions to the Public in the Plume Exposure EPZ," states that the primary method of alerting the public is by sounding the ANS. In addition, Sections 2.1 and 2.2 of Appendix 7, "Public Alert and Notification System," describe mobile sirens as the alternate method of notifying the public when offsite locations five miles from the site are not suitable for fixed sirens. The applicant revised Section E.5 and J.10.c. of the LNP Emergency Plan to discuss the alternate method used for alerting the public of an emergency. Additional discussion regarding notification of the public can be found in Section 13.3C.5.6 of this SER. Revision 6 of the LNP Emergency Plan states that the primary alert system consists of sirens and the backup is route alerting and the primary notification system is the EAS and the backup is route alerting.

Technical Evaluation: [J.10.c]
The staff finds that the LNP Emergency Plan adequately describes the means for notifying all segments of the transient and resident population. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

Technical Information in the Emergency Plan: [J.10.m]
Section J.10, "Protective Measures Implementation," of the LNP Emergency Plan states that choices of recommended protected actions are based on guidance provided in EPA 400-R-92-00. Section J.8, "Evacuation Time Estimates," and Appendix 6, "Evacuation Time Estimate Study Summary," of the LNP Emergency Plan provides a summary of ETE prepared for the plume Exposure Pathway EPZ. Table J-2, "10-Mile Emergency Planning Zone Evacuation Time Estimates (100 Percent) (Hr:Min)," provides an illustrative summary of ETEs within the Plume Exposure Pathway EPZ.

Technical Evaluation: [J.10.m]
The staff finds that the LNP Emergency Plan includes the basis for recommended protective actions for the plume exposure pathway during emergency conditions. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.10.11 *Conclusion*

The staff concludes that the information provided in the LNP Emergency Plan regarding protective response is acceptable and meets the requirements of 10 CFR 50.47(b)(10) because it conforms with the guidance in Evaluation Criterion J of NUREG-0654/FEMA-REP-1 as described above.

13.3C.11 Radiological Exposure Control

13.3C.11.1 *Regulatory Basis*

In determining whether the proposed emergency plan met the applicable regulatory requirements in 10 CFR 50.47(b)(11), the staff evaluated it against the detailed evaluation criteria in NUREG-0654/FEMA-REP-1.

13.3C.11.2 *Onsite Exposure Guidelines*

Technical Information in the Emergency Plan: [K.1.a-g]
Section K, “Radiological Exposure Control,” of the LNP Emergency Plan states that exposure guidelines are consistent with the Environmental Protection Agency (EPA) Emergency Worker and Lifesaving Activity Protective Action Guides described in EPA 400-R-92-001, “Manual of Protective Action Guides and Protective Actions for Nuclear Incidents.” Section K.1, “Emergency Exposures,” of the LNP Emergency Plan states that in the event of an emergency, workers involved in: the removal of injured persons; undertaking corrective actions; performing assessment actions; providing first aid; performing personnel decontamination; providing ambulance service; or providing medical treatment services would be expected to comply with routine dose limits unless the conditions of protecting valuable property, lifesaving, or protection of large populations would require a higher exposure. The higher-dose provision would be evaluated based on the guidelines in Table K-1, “Emergency Worker Exposure Guidelines,” of the LNP Emergency Plan.

Technical Evaluation: [K.1.a-g]
The staff finds that the LNP Emergency Plan adequately describes onsite exposure guidelines for the removal of injured persons, undertaking corrective actions, performing assessment actions, providing first aid, performing personnel decontamination, providing ambulance service, and providing medical treatment services. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.11.3 *Onsite Radiation Protection Program*

Technical Information in the Emergency Plan: [K.2]
Section K.2, “Radiation Protection Program [RPP],” states that the RPP’s purpose is to ensure that radiation doses are kept as low as reasonably achievable (ALARA) and do not exceed established limits for normal operating and emergency conditions. The established methods within the RPP include access control, personnel monitoring, and contamination control. The

applicant stated that the RPP and implementing procedures include provisions for implementing emergency exposure guidelines. Section K.1 of the LNP Emergency Plan states that the EC, in consultation with facility radiation protection personnel, can authorize doses exceeding the dose limits in 10 CFR Part 20, “Standards for protection against radiation.” If consideration for exceeding the occupational dose limits provided in 10 CFR Part 20 is required, these exposures will be limited to individuals who are properly trained and knowledgeable of the tasks to be completed and the risks associated with the exposures. Selection criteria for volunteer emergency workers include consideration of those who are in good physical health, are familiar with the consequences of emergency exposure, and are not a “declared pregnant adult.” Efforts are made to maintain personnel doses ALARA. Additional discussion regarding the circumstances surrounding the extension of exposure guidelines is located in Section 13.3C.11.2 of this SER. Additional information regarding the onsite RPP is located in SER Section 12.0, “Radiation Protection.” The applicant proposed EP ITAAC 10.1 to verify that site procedures provide the means for onsite radiation protection.

Technical Evaluation: [K.2]
The staff finds that the LNP Emergency Plan adequately provides an onsite radiation protection program to be implemented during emergencies, including methods to implement exposure guidelines. This is acceptable because it conforms to the guidance of NUREG-0654/FEMA-REP-1. The staff’s evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.11.4 *Capability to Determine Dose Received by Emergency Personnel*

Technical Information in the Emergency Plan: [K.3.a] {Appendix E, Section IV.E.1}
Section K.3, “Dosimetry and Dose Assessment,” of the LNP Emergency Plan states that dosimeters are maintained by the Radiation Protection section in adequate supply for use during an emergency. Implementing procedures describe in detail the types of personal dosimeter devices (both self-reading and permanent), the manner in which they are to be used, who is to wear them, and how they are cared for. The types of dosimeters include TLDs, electronic alarming dosimeters, and special types of ring badges. In an emergency situation, special care shall be taken to assure the proper reading frequency of dosimeters. Provisions have been established, onsite and through service organizations, to provide the 24-hour per day capability to read dosimeters to determine doses received by emergency workers. The applicant proposed EP ITAAC 10.2 to verify that EPIPs provide the means for the 24-hour per day capability to determine the doses received by emergency personnel and maintaining of dose records.

Technical Evaluation: [K.3.a]
The staff finds that the LNP Emergency Plan adequately describes provisions for distribution of dosimeters and the 24-hour per day capability to determine the doses received by emergency personnel involved in any radiological emergency. This is acceptable because it conforms to the guidance of NUREG-0654/FEMA-REP-1. The staff’s evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.11.5 *Dose Records for Emergency Personnel*

Technical Information in the Emergency Plan: [K.3.b]
Section 13.3C.11.4 of this SER provides discussion regarding the frequency for reading dosimeters issued to emergency workers. Section K.3.b of the LNP Emergency Plan states, in part, that the LNP RPP requires that the individual exposure records be documented and maintained to demonstrate and facilitate compliance with procedural requirements and applicable government regulations; and for reconstruction of the doses for medical or legal purposes.

Technical Evaluation: [K.3.b]
The staff finds that the LNP Emergency Plan adequately provides for ensuring that dosimeters are read at appropriate frequencies, and includes provisions for maintaining dose records for emergency workers involved in any nuclear accident. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.11.6 *Decontamination Action Levels*

Technical Information in the Emergency Plan: [K.5.a]
Section K.5, "Decontamination Action Levels," of the LNP Emergency Plan states that LNP implements procedural requirements for personnel and area decontamination, including decontamination action levels and criteria for returning areas and items to normal use. In addition, LNP implements procedures for decontamination of onsite personnel wounds, supplies, instruments and equipment, and for waste disposal.

Technical Evaluation: [K.5.a]
The staff finds that the LNP Emergency Plan adequately addresses decontamination action levels. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.11.7 *Decontamination Facilities and Supplies*

Technical Information in the Emergency Plan: [K.5.b] {Appendix E, Section IV.E.3}
Decontamination of onsite emergency personnel wounds, supplies, instruments and equipment, and for waste disposal is discussed in Section 13.3C.11.6 of this SER. In addition, Section K.5 of the LNP Emergency Plan states that LNP provides decontamination supplies with emergency kits. Section H.1.2, "Technical Support Centers," of the LNP Emergency Plan states that TSC contains a decontamination area and monitoring area, and that the TSC is equipped with a survey meter and an area radiation monitor. Section K.7, "Decontamination of Relocated LNP Personnel," states that LNP has dedicated decontamination and clothing kits and decontamination stations onsite. Additional information regarding the existence of a decontamination facility (Room 40355) in the Health Physics area of the Annex Building for personnel decontamination, which will include two personnel showers and two sinks connected to the radioactive liquid waste system, can be found in the staff's evaluation of the AP1000 DCD, NUREG-1793 and its supplements, Section 13.3.3.1, "General Description of Facilities." In RAI 13.3-52, the staff requested the applicant provide clarification in the LNP

Emergency Plan regarding the specific location(s) of any onsite decontamination facilities, including decontamination supplies associated with these facilities that will be used for decontaminating onsite personnel. In addition, the staff requested the applicant provide additional clarification regarding the existence of a decontamination area located inside the TSC since the AP1000 DCD drawings (e.g., Figure 1.2-19) do not include such an area. In response, the applicant stated, in part, that during non-emergency and emergency conditions, decontamination showers and supplies are provided onsite in the Health Physics (HP) area located in the Annex Building of the AP1000 units along with additional personnel decontamination equipment and capabilities. Basic decontamination supplies such as soaps, shampoo, mild detergent, 3 % Hydrogen Peroxide solution, plastic bags, plastic suits, cotton swabs, oral hygiene products, and saline solution will be available in the HP area. The decontamination and monitoring station near the HP area will remain the primary location during non-emergency and emergency conditions. However, in the event of an emergency when it is no longer practical for the HP area to be used as a decontamination area for TSC personnel, the TSC will also have a temporary decontamination and monitoring area established, including supplies.

Technical Evaluation: [K.5.b] {Appendix E, Section IV.E.3}
The staff finds that the additional information and proposed textual revisions to the emergency plan provided in response to RAI 13.3-52 to be acceptable because it provides clarification regarding the applicant's reference to the TSC as a decontamination area, and reference to the Annex Building as containing a decontamination facility, including decontamination supplies. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and meets the requirements of Appendix E to 10 CFR Part 50. The staff created **Confirmatory Item 13.3-52** to track the applicant's inclusion of its RAI response into the emergency plan.

Resolution of Confirmatory Item 13.3-52

Confirmatory Item 13.3-52 is an applicant commitment to update the LNP Emergency Plan. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). As a result, Confirmatory Item 13.3-52 is now closed.

The staff finds that the LNP Emergency Plan adequately addresses decontamination of emergency personnel and equipment. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and meets the requirements of Appendix E to 10 CFR Part 50.

13.3C.11.8 *Onsite Contamination Control*

Technical Information in the Emergency Plan: [K.6.a]
Section K.2 of the LNP Emergency Plan states, in part, that the RPP establishes measures to assure personnel doses are maintained ALARA, including contamination control. Section K.6, "Contamination Control Measures," of the LNP Emergency Plan states that the strict control of access to areas is a primary means to minimize radiation exposures. Section K.6.a describes implementing procedures that exist so that hazardous radiological areas can be quickly identified and controlled, and these measures are initiated by the EC through the use of

Radiation Monitoring Teams. In addition, the LNP Emergency Plan provides discussion regarding how Radiation Work Permits and Access Control Points are used to maintain control of personnel exposures, inform workers of radiological hazards, assure appropriate precautions are taken, and prevent the spread of contamination. In supplemental RAI 13.3-42 (Bullet 5), the staff requested the applicant clarify which implementing procedure supports and implements Section K, "Radiological Exposure Control," of the LNP Emergency Plan. In response, the applicant stated, in part, that the LNP Emergency Plan includes an EPIP for radiological exposure control that includes guidance for onsite contamination control. The applicant provided EP ITAAC 10.4 to verify site procedures provide the means for onsite contamination control measures.

[K.6.b]
Section K.6.b of the LNP Emergency Plan states that contamination control is enforced with respect to potable water and food supply by routine measures. All potable water for the plant comes from approved, surveyed locations and no food or drinking is permitted in the radiation controlled area (RCA).

[K.6.c]
Section K.6.c states that LNP would permit areas or items to be returned to normal use after it has been verified that contamination levels are within levels established by the LNP RPP or its supporting procedures.

Technical Evaluation: [K.6.a-c]
The staff finds that the LNP Emergency Plan adequately addresses onsite contamination control. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.11.9 *Capability to Decontaminate Relocated Onsite Personnel*

Technical Information in the Emergency Plan: [K.7]
Section K.7, "Decontamination of Relocated LNP Personnel," of the LNP Emergency Plan describes plans for decontamination of personnel who are relocated in an emergency. Personnel who are leaving a contaminated area are monitored to ensure that their person, personal clothing, and equipment are not contaminated. LNP has dedicated decontamination and clothing kits, and decontamination stations onsite to take offsite when needed. General procedures for personal cleanliness will generally remove contaminants and minimize exposure. Stronger cleansing agents may be utilized to remove contamination from the skin avoiding risk of injury to skin surfaces.

Technical Evaluation: [K.7]
The staff finds that the LNP Emergency Plan adequately describes the capability for decontaminating relocated onsite personnel, including provisions for extra clothing and decontaminants suitable for the type of contamination expected. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.11.10 *Conclusion*

The staff concludes that the information provided in the LNP Emergency Plan regarding radiation exposure control is acceptable because it conforms with the guidance in Evaluation Criterion K of NUREG-0654/FEMA-REP-1 and meets the applicable requirements of Appendix E to 10 CFR Part 50 and 10 CFR 50.47(b)(11).

13.3C.12 Medical and Public Health Support

13.3C.12.1 *Regulatory Basis*

In determining whether the proposed emergency plan met the applicable regulatory requirements in 10 CFR 50.47(b)(12), the staff evaluated it against the detailed evaluation criteria in NUREG-0654/FEMA-REP-1. The staff also evaluated the proposed emergency plan against applicable regulatory requirements related to the area of “Medical and Public Health Support,” in Appendix E to 10 CFR Part 50.

13.3C.12.2 *Onsite Medical Support*

Technical Information in the Emergency Plan: [L.2] {Appendix E, Section IV.E.5}
Section L.2, “On-site First Aid Capability,” states that First aid assistance at LNP is designed to handle a wide range of injuries from simple first aid to injuries requiring medical assistance. This task is accomplished by Medical Response Personnel. Section L.2.1, “Medical Response Personnel,” of the LNP Emergency Plan states that First Aid assistance is provided by Medical Response personnel who are onsite individuals trained in basic medical procedures and certified by the State of Florida Department of Health, Bureau of Emergency Medical Services and Community Health Resources. Section L.2 of the LNP Emergency Plan states that Medical Response personnel are trained to handle injured personnel with or without radiological considerations in accordance with implementing procedures. Appendix 5 to the LNP Emergency Plan includes an implementing procedure titled, “Medical Response.” References to certification letters, and LOAs, are provided in Appendix 3 from offsite organizations that will provide medical support to LNP in the event of an emergency.

Technical Evaluation: [L.2] {Appendix E, Section IV.E.5}
The staff finds that the LNP Emergency Plan adequately describes onsite medical support and arrangements made for the services of physicians and other medical personnel qualified to handle radiation emergencies onsite. This is acceptable because it meets the requirements in Appendix E to 10 CFR Part 50 and conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.12.3 *Offsite Medical Services*

Technical Information in the Emergency Plan: [L.1] {Appendix E, Section IV.E.7}
Section L.1, “Hospital and Medical Support,” of the LNP Emergency Plan states, in part, that LNP has an agreement with Seven Rivers Regional Medical Center and Citrus Memorial Hospital to provide medical services to radiological and non-radiological injured individuals that

require treatment offsite. Citrus Memorial Hospital will be used when Seven Rivers Regional Medical Center is not available due to an evacuation. Section L.1.3, "Off-Site Medical Support Plans," states that the REAC/TS in Oak Ridge, Tennessee, may be used, if warranted, depending on the nature or severity of the injury or when local facilities are deemed inadequate. Section L.1.3 also describes plans that Seven Rivers Regional Medical Center and Citrus Memorial Hospital have developed for the emergency handling of radioactive cases from LNP that carry out the terms of the hospital's agreements with Progress Energy. Table L-1, "Summary of Actions for Emergency Medical Treatment" describes onsite actions to be taken and offsite medical facilities to provide medical support depending upon the type of injury sustained and degree of contamination. In RAI 13.3-24, the staff requested that the applicant clarify whether REAC/TS should also be listed in Table L-1. In its response, the applicant committed to revise Table L-1 to include a note describing the use of REAC/TS, if required. Section N.2.c, "Medical Emergency Drills," states that Duke Energy will conduct medical emergency drills that include a simulated contaminated injured individual and may involve participation by the local support services (e.g., medical transportation and offsite medical treatment facilities) annually. Additional information regarding training for offsite emergency medical responders, which includes radiation protection precautions, can be found in Section 13.3C.15.2, "Training for Off-site Emergency Organizations," of this SER.

[L.4] {Appendix E, Section IV.E.6}
Section L.4, "Medical Emergency Transportation," of the LNP Emergency Plan states that transportation of injured personnel at LNP is available by using local emergency medical services, other Duke Energy vehicles, or private vehicles. In addition, the instructions and maps to local hospitals are provided in implementing procedures. Appendix 3 of the emergency plan includes local agreements for Nature Coast Emergency Medical Services and Citrus County Fire Rescue Division of Public Safety. Nature Coast Emergency Medical Services provides ambulance transport for injured and contaminated individuals. Appendix 5 identifies an EPIP titled, "Medical Response," that supports and implements this section of the LNP Emergency Plan.

Technical Evaluation: [L.1] {Appendix E, Section IV.E.7}
The staff finds the clarification and textual revision to the emergency plan provided in response to RAI 13.3-24 to be acceptable since it identifies an additional medical facility and service available to handle contaminated injured personnel should local resources be determined inadequate. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1. The staff confirmed that the changes proposed in response to RAI 13.3-24 were incorporated into the LNP Emergency Plan. Therefore, the staff finds that the LNP Emergency Plan adequately describes arrangements made for treatment of individuals injured in support of licensed activities on the site at treatment facilities outside the site boundary. This is acceptable because it meets the requirements in Appendix E to 10 CFR Part 50 and it conforms to the guidance in NUREG-0654/FEMA-REP-1.

[L.4] {Appendix E, Section IV.E.6}
The staff finds that the LNP Emergency Plan adequately describes the arrangements made for transportation of contaminated injured individuals from the site to specifically identified treatment

facilities outside the site boundary. This is acceptable because it meets the requirements in Appendix E to 10 CFR Part 50 and it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.12.4 Conclusion

The staff concludes that the information provided in the LNP Emergency Plan regarding medical and public health support is acceptable and meets the requirements of 10 CFR 50.47(b)(12) because it conforms with the guidance in Evaluation Criterion L of NUREG-0654/FEMA-REP-1 and meets the applicable requirements of Appendix E to 10 CFR Part 50 as described above.

13.3C.13 Recovery and Reentry Planning and Post-Accident Operations

13.3C.13.1 *Regulatory Basis*

In determining whether the proposed emergency plan met the applicable regulatory requirements in 10 CFR 50.47(b)(13) for recovery and reentry planning and post-accident operations, the staff evaluated it against the detailed evaluation criteria in NUREG-0654/FEMA-REP-1. The staff also evaluated the proposed emergency plan against applicable regulatory requirements related to the area of “Recovery and Reentry Planning and Post-Accident Operations,” in Appendix E to 10 CFR Part 50.

13.3C.13.2 *Plans and Procedures for Reentry and Recovery*

Technical Information in the Emergency Plan: [M.1] {Appendix E, Section IV.H}
Section M.1, “Recovery Plans and Procedures,” of the LNP Emergency Plan,” states that Duke Energy implements recovery plans and procedures that provide guidance for a range of recovery and re-entry activities, including the recovery/re-entry organization. The recovery organization develops plans and procedures designed to address both immediate and long-term actions. The recovery organization will recommend relaxation of the protective measures based on the following conditions: site parameters of operation no longer indicate a potential or actual emergency exists; the release of radioactivity from the station is controllable, no longer exceeds permissible levels, and does not present a credible danger to the public; the site is capable of sustaining itself in a long-term shutdown condition. Reentry procedures may need to be written for specific requirements and as recovery operations progress, resources may be increased or reduced to ensure effectiveness in meeting operational needs. A procedure titled, “Recovery and Reentry,” is referenced in Appendix 5, “List of Emergency Plan Supporting Procedures,” as supporting and implementing Section M of the LNP Emergency Plan.

Technical Evaluation: [M.1] {Appendix E, Section IV.H}
The staff finds that the LNP Emergency Plan adequately describes general plans and procedures for reentry and recovery and describes the means by which decisions to relax protective measures are reached. This process considers both existing and potential conditions. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and meets the applicable requirements in Appendix E to 10 CFR Part 50.

13.3C.13.3 *Recovery Organization*

Technical Information in the Emergency Plan: [M.2]
Section M.2, "Recovery Operations," states that the EOF Director is responsible for control and direction of the recovery/re-entry operation as defined in implementing procedures. The recovery organization may be modified as required to better respond to site conditions. The EC acts as the site liaison with the recovery organization. The State of Florida will be the lead organization for offsite recovery operations in accordance with the State of Florida Radiological Emergency Management Plan (REMP). The recovery process is implemented when LNP ERO managers, with concurrence of State and Federal agencies, determine the site to be in a stable and controlled condition. Upon this determination, the EOF Director notifies the NRCOC, the State EOC, and local EOCs that the emergency has terminated and any required recovery has commenced.

Technical Evaluation: [M.2]
The staff finds that the LNP Emergency Plan adequately provides the position/title, authority, and responsibilities of individuals who will fill key positions in the facility recovery organization. The organization includes technical personnel with responsibilities to develop, evaluate, and direct recovery and reentry operations. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.13.4 *Recovery Operations Initiation*

Technical Information in the Emergency Plan: [M.3]
Section M.1.c, **"**Recovery Plans and Procedures," of the LNP Emergency Plan states that Duke Energy implements recovery plans and procedures that provide guidance for a range of recovery and re-entry activities, including the means for informing members of the ERO when recovery operations are to be initiated and any related changes in the organizational structure. The recovery process will be implemented when the LNP ERO managers have determined the site to be in a controlled and stable condition. Section 13.3C.13.2 of this SER provides discussion regarding a recovery and reentry procedures available to support and implement Section M of the LNP Emergency Plan.

Technical Evaluation: [M.3]
The staff finds that the LNP Emergency Plan adequately addresses the means for informing members of the response organizations that a recovery operation is to be initiated, and of any changes in the organizational structure that may occur. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.13.5 *Methods to Estimate Total Population Exposure*

Technical Information in the Emergency Plan: [M.4]
Section M.1,d, "Recovery Plans and Procedures," of the LNP Emergency Plan states that Duke Energy implements plans and procedures for recovery and reentry activities including methods

for periodically updating estimates of total population exposure. Section M.3, "Updating Total Population Exposure," states, in part, that the Radiological Control Manager will periodically update estimates of total population exposure using population distribution data from within EPZs. Section I.10, "Relating Measured Parameters to Dose Rates," states that implementing procedures establish the means for relating measured parameters to dose rates for key isotopes listed in Table 3 of NUREG-0654, Revision 1. Section 13.3C.13.2 of this SER provides discussion regarding a recovery and reentry procedure available to support and implement Section M of the LNP Emergency Plan. Appendix 5 of the LNP Emergency Plan also includes reference to an EPIP titled, "Dose Assessment," that supports and implements Section I of the plan.

Technical Evaluation: [M.4]
The staff finds that the LNP Emergency Plan adequately establishes a method for periodically estimating total population exposure. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.13.6 *Conclusion*

The staff concludes that the information provided in the LNP Emergency Plan regarding recovery and reentry planning and post-accident operations is acceptable and meets the requirements of 10 CFR 50.47(b)(13) because it conforms with the guidance in Evaluation Criterion M of NUREG-0654/FEMA-REP-1, and complies with the applicable portions of Appendix E to 10 CFR Part 50 as described above.

13.3C.14 Exercises and Drills

13.3C.14.1 *Regulatory Basis*

In determining whether the proposed emergency plan met the applicable regulatory requirements in 10 CFR 50.47(b)(14) for exercises and drills, the staff evaluated it against the detailed evaluation criteria in NUREG-0654/FEMA-REP-1. The staff also evaluated the proposed emergency plan against applicable regulatory guidance related to the area of "Exercises and Drills," in NSIR/DPR-ISG-01 and requirements in Appendix E to 10 CFR Part 50.

13.3C.14.2 *Emergency Preparedness Exercise Purpose and Content*

Technical Information in the Emergency Plan: [N.1.a]
Section N, "Exercises and Drills," of the LNP Emergency Plan states that Duke Energy implements a program of periodic exercises to evaluate major portions of emergency response capabilities and to develop and maintain key emergency response skills. Section N.1, "Exercises," defines an exercise as an event that tests the integrated capability and a major portion of the basic elements existing within EP plans and organizations. In RAI 13.3-53(1)(a), the staff requested the applicant clarify whether EP exercises will simulate an emergency that results in offsite radiological releases which would require response by offsite authorities, and are conducted as set forth in NRC and FEMA rules. In response, the applicant acknowledged

the need to incorporate this information into its emergency plan and proposed a revision accordingly.

Technical Evaluation: [N.1.a]
The staff finds the additional information and proposed textual revisions to the emergency plan submitted in response to RAI 13.3-53(1)(a) is acceptable because it proposes to incorporate the criteria for exercises consistent with NUREG-0654/FEMA-REP-1. The staff created Confirmatory Item 13.3-53(1)(a) to track the applicant's revision to the emergency plan consistent with this RAI response.

<u>Resolution of Confirmatory Item 13.3-53(1)(a)</u>

Confirmatory Item 13.3-53(1)(a) is an applicant commitment to update the LNP Emergency Plan. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). As a result, Confirmatory Item 13.3-53(1)(a) is now closed.

The staff finds that the LNP Emergency Plan adequately states that exercises will test the integrated capability and the major elements of the emergency plans and preparedness program. In addition, the emergency preparedness exercise will, as appropriate, simulate an emergency that results in offsite radiological releases which would require response by offsite authorities and that exercises will be conducted as set forth in NRC and FEMA rules.

Technical Information in the Emergency Plan: [N.1.b]
Section N.1.a, "Exercise Scope and Frequency," states that an exercise will be conducted every two years. Revision 1 of the LNP Emergency Plan states that the scenario will be varied to ensure all major elements of the LNP Emergency Plan are tested within a 6-year period. Major elements to be tested include: management and coordination of emergency response, accident assessment, protective action decision-making, and plant system repair and corrective action. State and local agencies will be invited to participate in off-year exercises. Section N.1.b, "Exercise Scenario and Participation," states the frequency of the State of Florida's participation in exercises with Duke Energy (formerly Progress Energy) is discussed in Chapter 14 of the State Plan. The State's participation may be either full or partial depending on the objectives of the exercise and the degree to which the State and local plans are tested. The State Division of Emergency Management is responsible for assuring that exercises are conducted as set forth in NRC and FEMA rules. Post-exercise meetings with participants and observers will be conducted to assess emergency response actions. Comments resulting from these sessions should serve as input to the critique as discussed in Section N.5, "Exercise and Drill Critiques," of the emergency plan. In RAI 13.3-53(2), the staff requested the applicant clarify whether the following provisions for the conduct of EP exercises have been made: 1) an EP exercises shall start between 6:00 p.m. and 4:00 a.m. once every six years; 2) exercises will be conducted during different seasons of the year to vary weather conditions; and 3) some exercises will be unannounced. In response, the applicant acknowledged that the provisions for exercises stated above in this RAI have been made and proposed a revision to the emergency plan incorporating this information. Revision 6 of the LNP Emergency Plan describes the use of exercises to test

major elements of the plans and preparedness organizations within an eight-year exercise cycle.

Technical Evaluation: [N.1.b]
The staff finds the additional information and proposed textual revisions to the emergency plan submitted in response to RAI 13.3-53(2) to be acceptable because they conform to the guidance in NUREG-0654/FEMA-REP-1. The staff created **Confirmatory Item 13.3-53(2)** to track the applicant's proposed revisions to the emergency plan consistent with this RAI response.

Resolution of Confirmatory Item 13.3-53(2)

Confirmatory Item 13.3-53(2) is an applicant commitment to update the LNP Emergency Plan. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). As a result, Confirmatory Item 13.3-53(2) is now closed.

The staff finds that the LNP Emergency Plan adequately states that exercises will include mobilization of State and local personnel and resources adequate to verify the capability to respond to an emergency event. In addition, the LNP Emergency Plan adequately describes provisions for a critique of the biennial exercise by Federal and State observers/evaluators. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.14.3 *Emergency Preparedness Exercises*

Technical Information in the Emergency Plan: {Appendix E, Section IV.F.2}
Section N of the LNP Emergency Plan states that Progress Energy implements a program of periodic exercises to evaluate major portions of emergency response capabilities and to develop and maintain key emergency response skills. In RAI 13.3-53(1)(b), the staff requested the applicant clarify whether the following provisions for the conduct of EP exercises have been made: 1) exercises will test the adequacy of timing and content of implementing procedures and methods; 2) exercises will test emergency equipment, communication networks, and the public notification system; and 3) exercises will ensure the members of the ERO are familiar with their duties. In response, the applicant acknowledged that the provisions for exercises stated above in this RAI have been made and proposed a revision to the emergency plan incorporating this information. In RAI 13.3-64 the staff requested the applicant clarify in the LNP Emergency Plan whether EP exercises will be designed to test the public alert and notification system. In response, the applicant proposed a revision to the emergency plan (Section N.1) as described above.

Technical Evaluation: {Appendix E, Section IV.F.2}
The staff finds the additional information and proposed textual revisions to the emergency plan submitted in response to RAIs 13.3-53(1)(b) and 13.3-64 to be acceptable because they meet the requirements in Appendix E to 10 CFR Part 50. The staff created Confirmatory Item 13.3-

53(1)(b) and 13.3-64 to track the applicant's proposed revisions to the emergency plan consistent with this RAI response.

<u>Resolution of Confirmatory Item 13.3-53(1)(b) and 13.3-64</u>

Confirmatory Items 13.3-53(1)(b) and 13.3-64 are the applicant's commitment to update the LNP Emergency Plan. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). As a result, Confirmatory Item 13.3-53(1)(b) and 13.3-64 are now closed.

Therefore, the staff finds that the LNP Emergency Plan adequately describes provisions for the conduct of emergency preparedness exercises and specifies that exercises test the adequacy of timing and content of implementing procedures and methods, test emergency equipment and communications networks, test the public notification system, and ensure that emergency organization personnel are familiar with their duties. This is acceptable because it meets the requirements in Appendix E to 10 CFR Part 50.

13.3C.14.4 *Full Participation Exercise Prior to Fuel Load*

Technical Information in the Emergency Plan: {Appendix E, Section IV.F.2.a}
Section 13.3A.3 of this SER provides discussion and evaluation on EP implementation milestones to include a full participation exercise prior to fuel load. In addition, the applicant proposed EP ITAAC 12.0 to ensure the conduct of a full participation exercise that tests major portions of emergency response capabilities, and includes participation by each State and local agency within the plume exposure pathway EPZ, and each State within the ingestion control EPZ. The exercise will be conducted within the specified time periods of 10 CFR Part 50, Appendix E.

Technical Evaluation: {Appendix E, Section IV.F.2.a}
The staff finds that the LNP Emergency Plan adequately describes provisions for the conduct of a full participation exercise at least one year before fuel load. This is acceptable because it meets the applicable requirements in Appendix E to 10 CFR Part 50. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.14.5 *Onsite Biennial Exercise*

Technical Information in the Emergency Plan: {Appendix E, Section IV.F.2.b}
Section N.1.a of the LNP Emergency Plan states that an emergency response exercise will be conducted every 2 years. Section N.1 states, in part, that at least one drill involving principal areas of onsite emergency response capabilities will be conducted during the interval between the biennial exercise. Drills will include management and coordination of emergency response, accident assessment, protective action decision-making, plant system repair, and corrective actions, which would assure that emergency organization personnel are familiar with their duties. State and local agencies will be invited to participate in off-year drills.

Technical Evaluation: {Appendix E, Section IV.F.2.b}
The staff finds that the LNP Emergency Plan adequately states that an exercise of its onsite emergency plan will be conducted every 2 years and adequately describes actions that will be taken to ensure that adequate emergency response capabilities are maintained during the interval between biennial exercises by conducting drills, including at least one drill involving a combination of some of the principal functional areas of the licensee's onsite emergency response capabilities. This is acceptable because it meets the applicable requirements in Appendix E to 10 CFR Part 50.

13.3C.14.6 *Offsite Biennial Exercise / Ingestion Pathway Exercise with State*

Technical Information in the Emergency Plan: {Appendix E, Section IV.F.2.c} {Appendix E, Section IV.F.2.d}
Section N.1.b of the LNP Emergency Plan states, in part, that the (Florida) State Division of Emergency Management is responsible for implementing Chapter 14, "Exercises and Drills," of the State Plan which specifies the frequency that the State of Florida will participate in an exercise with Duke Energy. The Division of Emergency Management will assure that exercises are conducted as set forth in NRC and FEMA rules. Duke Energy will conduct an emergency response exercise every 2 years, with intermediate drills, to test specific sections of the plans. State and local agencies will be invited to participate in these intermediate drills.

Technical Evaluation: {Appendix E, Section IV.F.2.c} {Appendix E, Section IV.F.2.d}
The staff reviewed FEMA's findings and determinations regarding the adequacy of offsite exercise participation by State and local government authorities, in addition to the REMPs of the State of Florida, and counties of Levy, Citrus, and Marion. The staff confirmed that the plans addressed the applicable requirements of Appendix E to 10 CFR Part 50. The staff finds that the LNP Emergency Plan adequately addresses the requirements for biennial exercises of authorities having a response role at the LNP site, and the States' participation in the ingestion pathway exercise. This is acceptable because it meets the applicable requirements in Appendix E to 10 CFR Part 50.

13.3C.14.7 *Enabling Local and State Participation in Drills*

Technical Information in the Emergency Plan: {Appendix E, Section IV.F.2.e}
Section N.2 of the LNP Emergency Plan states, in part, that upon request, Duke Energy allows affected State and local governments located within the plume exposure EPZ to participate in drills.

Technical Evaluation: {Appendix E, Section IV.F.2.e}
The staff finds that the LNP Emergency Plan adequately describes how the licensee will enable any State or local government located within the plume exposure pathway EPZ to participate in the licensee's drills when requested by such State or local government. This is acceptable because it meets the applicable requirements in Appendix E to 10 CFR Part 50.

13.3C.14.8 *Remedial Exercises*

Technical Information in the Emergency Plan: {Appendix E, Section IV.F.2.f}
Section N of the LNP Emergency Plan describes how exercises are conducted to evaluate emergency response capabilities. Section N.1 describes the exercise scope, frequency, scenarios, and participation. In RAI 13.3-53(3), the staff requested the applicant clarify in the LNP Emergency Plan whether remedial exercises will be conducted for unsatisfactory performance during a biennial exercise that results in the loss of NRC and FEMA reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. In response, the applicant acknowledged that the provisions for exercises stated above in this RAI have been made and proposed a revision to the emergency plan incorporating this information.

Technical Evaluation: {Appendix E, Section IV.F.2.f}
The staff finds the additional information and proposed textual revisions to the emergency plan submitted in response to RAI 13.3-53(3) to be acceptable because it meets the requirements in Appendix E to 10 CFR Part 50. The staff created Confirmatory Item 13.3-53(3) to track the applicant's proposed revisions to the emergency plan consistent with this RAI response.

Resolution of Confirmatory Item 13.3-53(3)

Confirmatory Item 13.3-53(3) is an applicant commitment to update the LNP Emergency Plan. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). As a result, Confirmatory Item 13.3-53(3) is now closed.

The staff finds that the LNP Emergency Plan adequately describes provisions for how remedial exercises will be conducted if the emergency plan is not satisfactorily tested during the biennial exercise, such that the NRC and FEMA, cannot find reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. This is acceptable because it meets the requirements in Appendix E to 10 CFR Part 50.

13.3C.14.9 *Drills*

Technical Information in the Emergency Plan: [N.2]
Section N.2, "Drills," of the LNP Emergency Plan states that Duke Energy conducts drills between biennial exercises to maintain adequate emergency response capabilities. Drills would include activities such as management and coordination of emergency response, accident assessment, protective action decision-making, plant system repair, and corrective actions. Drills are used to consider accident management strategies, provide supervised instruction, allow the operating staff to resolve problems and focus on internal training objectives. Exercises may include one or more drills. State and local governments located within the plume exposure pathway EPZ are invited to participate in the drills when requested.

By letter dated November 8, 2012, from PEF to the NRC, the applicant provided its response to address the Final Rule on Enhancements to Emergency Preparedness Regulation effective

November 23, 2011. In regard to implementation of the EP rule pertaining to the conduct of exercises and drills, the staff requested additional information from the applicant in RAI 13.3-61 to clarify in the LNP Emergency Plan whether the EOF staff for LNP and CR3 will demonstrate its ability to perform their consolidated EOF functions in at least one drill or exercise per exercise cycle thereafter. In response, the applicant proposed a revision to the LNP Emergency Plan (Section N.2) to practice its EOF integrated capability with CR3.

Technical Evaluation: [N.2]
The staff finds the additional information and textual revisions to the LNP Emergency Plan submitted in response to RAI 13.3-61 to be acceptable since it conforms to the regulatory guidance in NSIR/DPR-ISG-01. The staff verified that the LNP Emergency Plan (Revision 6) was appropriately updated (or revised). By letter dated April 18, 2013, from PEF to NRC regarding impacts from retirement of the CR3 nuclear plant, the applicant renamed the EOF to remove any reference to the Crystal River Training Center. The EOF is now referred to in the LNP Emergency Plan as the LNP EOF. In addition, the applicant added a conditional statement to address when the EOF is required for use by CR3. As discussed in Section 13.3.4, CR3 was granted exemptions from specific EP standards including the requirement to have an EOF and to conduct drills and exercises.

In consideration of the applicant's response to the new EP rule and deletion of the reference to the Crystal River Training Center with additional conditional language, the staff finds the LNP Emergency Plan adequately describes how a drill is a supervised instruction period aimed at testing, developing, and maintaining skills in a particular operation. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and NSIR/DPR-ISG-01.

13.3C.14.10 *Communications Drills*

Technical Information in the Emergency Plan: [N.2.a] {Appendix E, Section IV.E.9(b)}
Section N.2.a, "Communications Drills," of the LNP Emergency Plan states that Duke Energy tests communications with State and local governments within the plume exposure EPZ monthly. Duke Energy tests communications with Federal EROs and States within the ingestion exposure pathway EPZ monthly. Communications tests between the facility, State, and local EOCs, and field assessment teams are performed annually. Communications drills evaluate the operability of the communications systems and the ability to understand message content. Additional information related to communication systems and testing can be found in Section F.3, "Communication System Reliability," of the LNP Emergency Plan.

Technical Evaluation: [N.2.a] {Appendix E, Section IV.E.9(b)}
The staff finds the LNP Emergency Plan adequately describes communication drills and testing frequencies with Federal, State and local governments in the plume exposure and ingestion exposure pathway EPZs. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1, and the applicable requirements of Appendix E to 10 CFR Part 50.

13.3C.14.11 *Fire Drills*

Technical Information in the Emergency Plan: [N.2.b]
Section N.2.b, “Fire Drills,” of the LNP Emergency Plan states that Duke Energy conducts fire drills as discussed in Section 9.5.1.8.2.2 of the LNP COL FSAR. Section 9.5.1.8.2.2.4 of the LNP COL FSAR, “Drills,” states that fire brigade drills are conducted at least once per calendar quarter for each shift, with each member of the fire brigade participating in at least two drills annually. Drills are either announced or unannounced. At least one unannounced drill is held annually for each shift of the fire brigade. At least one drill is performed annually on a “back shift” for each shift’s fire brigade. The drills provide for offsite fire department participation at least annually. Triennially, a randomly selected, unannounced drill shall be conducted and critiqued by qualified individuals independent of the plant staff. Training objectives are established prior to each drill and reviewed by plant management. Criteria to be critiqued during the drills are also listed. Performance deficiencies identified during the drill is used as the basis for additional training and repeat drills. Unsatisfactory drill performance is followed by a repeat drill within 30 days.

Technical Evaluation: [N.2.b]
The staff finds the LNP Emergency Plan adequately describes how fire drills will be conducted in accordance with the LNP COL FSAR. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.14.12 *Medical Emergency Drills*

Technical Information in the Emergency Plan: [N.2.c]
Section N.2.c, “Medical Emergency Drills,” of the LNP Emergency Plan states that Duke Energy conducts annual medical drills that will include a simulated contaminated injury. These drills may involve participation by the local support service agencies (e.g., medical transportation and offsite medical treatment facility).

Technical Evaluation: [N.2.c]
The staff finds the LNP Emergency Plan adequately describes the scope, frequency, and participation of a medical emergency drill. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.14.13 *Radiological Monitoring Drills*

Technical Information in the Emergency Plan: [N.2.d]
Section N.2.d, “Radiological Monitoring Drills/Health Physics Drills,” of the LNP Emergency Plan states that Duke Energy conducts radiological monitoring drills, involving both onsite and offsite radiological monitoring activities, annually. These drills test procedures for collecting, analyzing samples, and recording results; collection and analysis of all sample media for which the facility is responsible; communications with monitoring teams; and record keeping. Radiological

monitoring drills may be coordinated with drills conducted by State and local government entities or conducted independently.

Technical Evaluation: [N.2.d]
The staff finds the LNP Emergency Plan adequately describes that plant environs and radiological monitoring drills (onsite and offsite) will be conducted annually; and where appropriate, local organizations participate. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.14.14 *Health Physics Drills*

Technical Information in the Emergency Plan: [N.2.e]
Section N.2.e, "Sampling Drills," of the LNP Emergency Plan states that onsite radiation protection drills are conducted at least semi-annually. Drills include: the response to, and analysis of, simulated elevated airborne and liquid activity levels; response to simulated elevated area radiation levels; and analysis of the simulated radiological situation using the appropriate procedures. State and local participation during these drills is discussed in Section 13.3C.14.13 of this SER.

Technical Evaluation: [N.2.e]
The staff finds the LNP Emergency Plan adequately describes how radiation protection drills will be conducted semi-annually and involves response to, and analysis of, simulated elevated airborne and liquid samples and direct radiation measurements in the environment. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.14.15 *Conduct of Drills and Exercises*

Technical Information in the Emergency Plan: [N.3.a-f]
Section N.3, "Conduct of Drills and Exercises," of Revision 1 of the LNP Emergency Plan states that the EP organization is responsible for the overall development and direction of the exercise. The Exercise Director (ED) is responsible for the development of an exercise plan for each exercise to include the following: 1) the objectives of the exercise and evaluation criteria; 2) the date, time, place, and participating organizations; 3) a time schedule of real and simulated events; 4) a narrative summary of the event including such items as emergency classification at various times in the simulated accident, 5) offsite assistance and details about the plant conditions; and 6) a description of the arrangement for official observers. In RAI 13.3-53(4), the staff requested the applicant clarify whether the discussion in the LNP Emergency plan is also applicable for drills. In response, the applicant stated, in part, that Section N.3 is applicable to exercises and drills, which describes exercise content that shall be included in the exercise plan. The plan content listed in Section N.3.a-e should also be used for large scale integrated drills that involve activation and participation by both onsite and offsite agencies.

Technical Evaluation: [N.3.a-f]
The staff finds that the additional information and proposed textual revisions to the LNP Emergency Plan provided in response to RAI 13.3-53(4) to be acceptable because it clarifies

that the information contained in the emergency plan (Section N.3) is applicable to emergency preparedness drills, and conforms to the guidance in NUREG-0654/FEMA-REP-1. The staff created Confirmatory Item 13.3-53(4) to track the applicant's proposed revision to the emergency plan provided in response to this RAI.

Resolution of Confirmatory Item 13.3-53(4)

Confirmatory Item 13.3-53(4) is an applicant commitment to update the LNP Emergency Plan. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). As a result, Confirmatory Item 13.3-53(4) is now closed.

The staff finds that the LNP Emergency Plan adequately describes how exercises and drills will be carried out to allow free play for decision-making and to meet the exercise objectives. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.14.16 Observing, Evaluating, and Critiquing Drills and Exercises

Technical Information in the Emergency Plan: [N.4] {Appendix E, Section IV.F.2(g)}
Section N.4, "Exercise and Drill Evaluation," of the LNP Emergency Plan states that qualified Duke Energy instructors/evaluators will supervise and evaluate drills and exercises. A qualified instructor/evaluator is an individual whose knowledge, skills, and abilities have been evaluated by the EP Manager or designee to determine whether they are qualified to observe and evaluate the planned activities against established criteria. Specific areas to be observed by the evaluators will be defined in the form of pre-printed critique sheets. Critiques will be performed as soon as practicable following each exercise. Duke Energy staff, the NRC, State, local, and other participants, and observers/evaluators, will participate in the critiques. A formal evaluation will result from the critique. In RAI 13.3-25, the staff requested that the applicant clarify whether critiques also apply to drills. In its response, the applicant committed to revise Section N.4 to clarify that critiques are for drills and exercises and that a formal evaluation is strictly for an evaluated exercise by NRC or FEMA.

Technical Evaluation: [N.4] {Appendix E, Section IV.F.2(g)}
The staff finds the clarification and textual revisions to the emergency plan provided in response to RAI 13.3-25 to be acceptable because they conform to the guidance in NUREG-0654/FEMA-REP-1, and meet the applicable requirements in Appendix E to 10 CFR Part 50. The staff confirmed that the changes referenced above were included in Revision 1 to the LNP Emergency Plan. Therefore, the staff finds that the LNP Emergency Plan adequately describes provisions for official observers from Federal, State, or local governments to observe, evaluate, and critique the required exercises. This is acceptable because it meets the applicable requirements in Appendix E to 10 CFR Part 50 and the guidance described in NUREG-0654/FEMA-REP-1.

13.3C.14.17 *Means to Correct Areas Needing Improvement*

Technical Information in the Emergency Plan: [N.5]
Section N.5, "Exercise and Drill Critiques," of the LNP Emergency Plan states that Duke Energy records the input from exercise and drill critique participants and then evaluates the needs for changes to the Plan, procedures, equipment, facilities, and other components of the EP program, and develops an action plan to address substantive issues. Duke Energy tracks identified corrective actions to completion using the site's Corrective Action Program. In RAI 13.3-25, the staff requested that the applicant clarify whether the results of critiques are factored into initial and retraining of personnel. In its response, the applicant committed to revise Section N.5 to clarify that the adequacy of the Emergency Preparedness training program is considered for improvement during exercise and drill critiques. Revision 6 of the LNP Emergency Plan states that the exercise and drill scenario package and post-exercise/drill critiques are filed as records.

Technical Evaluation: [N.5]
The staff finds the clarification and textual revisions to the emergency plan provided in response to RAI 13.3-25 to be acceptable because they conform to NUREG-065/FEMA-REP-1. The staff confirmed that the changes referenced above were incorporated into Revision 1 to the LNP Emergency Plan. Therefore, the staff finds that the LNP Emergency Plan adequately describes a means for evaluating observer and participant comments on areas needing improvement, including emergency plan procedural changes, and for assigning responsibility for implementing corrective actions. The LNP Emergency Plan also establishes management control used to ensure that corrective actions are implemented. This is acceptable because it conforms to the guidance described in NUREG-0654/FEMA-REP-1.

13.3C.14.18 *Conclusion*

The staff concludes that the information provided in the LNP Emergency Plan regarding exercises and drills is acceptable and meets the requirements of 10 CFR 50.47(b)(14) because it conforms with the guidance in Evaluation Criterion N of NUREG-0654/FEMA-REP-1 and NSIR/DPR-ISG-01 and meets the applicable requirements of Appendix E to 10 CFR Part 50 as described above.

13.3C.15 Radiological Emergency Training

13.3C.15.1 *Regulatory Basis*

In determining whether the proposed emergency plan met the applicable regulatory requirements in 10 CFR 50.47(b)(15) for radiological emergency training, the staff evaluated it against the detailed evaluation criteria in NUREG-0654/FEMA-REP-1. The staff also evaluated the proposed emergency plan against applicable regulatory requirements related to the area of "Radiological Emergency Training," in Appendix E to 10 CFR Part 50.

13.3C.15.2 *Training for Off-site Emergency Organizations*

Technical Information in the Emergency Plan: [O.1.a]
Section O.1, "General Requirements," of the LNP Emergency Plan states, in part, that Duke Energy implements a training program that provides for initial training and retraining for individuals and organizations who have been assigned emergency response duties.
Section O.1.a, "Off-site Emergency Response Training," of the LNP Emergency Plan states that Duke Energy conducts, or supports the site-specific training for offsite personnel who may be called upon to provide assistance in the event of an emergency. Duke Energy provides or supports training for affected hospital, ambulance/rescue, police and firefighting personnel, which includes their expected emergency response roles, notification procedures, and radiation protection precautions. In addition, Section O.1.a states that Duke Energy provides or supports training for offsite responders that addresses LNP access procedures and identifies (by position) the individual who will control onsite activities. Appendix 5 of the LNP Emergency Plan identifies an Administrative Procedure, "Emergency Preparedness Training," that supports and implements Section O, "Radiological Emergency Response Training," of the LNP Emergency Plan. Additional information regarding Emergency Plan Training can be found in Section 13.2.2 of this SER.

Technical Evaluation: [O.1.a]
The staff finds that the LNP Emergency Plan adequately describes the site-specific emergency response training to be provided for offsite emergency organizations that may be called upon to provide assistance in the event of an emergency. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.15.3 *Onsite Emergency Response Organization Training*

Technical Information in the Emergency Plan: [O.2]
Section O.2, "Duke Energy Emergency Response Training," of the LNP Emergency Plan states that the emergency response training program includes Duke Energy personnel who may be called upon to respond to an emergency, in which each individual completes the required training prior to being assigned to a position in the ERO. Section O.4, "Emergency Response Training and Qualification," provides a discussion regarding the categories of specialized training programs (e.g., training and retraining for directors or coordinators of the response organization) and scope of training for the onsite ERO. Section N of the LNP Emergency Plan states that Duke Energy implements a program of periodic drills and exercises to develop and maintain key emergency response skills. Section N.2 states that Duke Energy may use drills to provide supervised instruction, allow the operating staff to resolve problems, and focus on internal training objectives. Additional information regarding the retraining of onsite emergency responders is provided in Section 13.3C.15.16 of this SER. Additional information regarding Emergency Plan Training can be found in Section 13.2.2 of this SER.

Technical Evaluation: [O.2]
The staff finds that the LNP Emergency Plan adequately describes the training program for members of the onsite emergency organization. This is acceptable because it conforms to the guidance described in NUREG-0654/FEMA-REP-1.

13.3C.15.4 *First Aid and Rescue Team Training*

Technical Information in the Emergency Plan: [O.3] [O.4.f] {Appendix E, Section IV.F.1(b)(vi)}
Section O.3, "First Aid Training," states that Duke Energy provides first aid training to all individuals assigned to Medical Response teams in accordance with approved procedures. Section O.4, "Emergency Response Training and Qualification," states that the scope of associated training for first aid and rescue team responders includes emergency organizational interfaces, search and rescue procedures, and communication systems. Section L.2 of the LNP Emergency Plan states that first aid assistance is provided by medical response personnel who are onsite individuals trained in basic medical procedures and certified by the State of Florida Department of Health, Bureau of Emergency Medical Services and Community Health Resources. In addition Section L.2 states that medical response personnel are trained to handle injured personnel with or without radiological considerations in accordance with implementing procedures. Appendix 5 of the LNP Emergency Plan identifies an EPIP titled, "Medical Response." Additional information regarding the retraining of first aid and rescue team emergency responders is in Section 13.3C.15.16 of this SER

Technical Evaluation: [O.3] [O.4.f] {Appendix E, Section IV.F.1(b)(vi)}
The staff finds that the LNP Emergency Plan adequately describes specialized initial and refresher training for first aid and rescue teams. This is acceptable because it conforms to the guidance described in NUREG-0654/FEMA-REP-1 and meets the requirements of Appendix E to 10 CFR Part 50.

13.3C.15.5 *Training Program to Implement the Emergency Plan*

Technical Information in the Emergency Plan: [O.4] {Appendix E, Section IV.F.1}
Section O.4 of the LNP Emergency Plan states that Duke Energy conducts a program for instructing and qualifying all personnel who implement the LNP Emergency Plan. Personnel complete the required training prior to assignment to a position in the ERO. The training program establishes the scope, nature, and frequency of the required training and qualification measures. The program provides position-specific training for members of the ERO that is appropriate for the duties and responsibilities of the position. The positions and scope of training programs include the following: Directors, coordinators and managers in the ERO; accident assessment personnel; radiological control personnel; police security, and firefighting personnel; damage control/emergency repair teams; first aid, fire brigade, and rescue personnel; local support services/emergency service personnel; offsite medical support personnel; emergency communicators; and personnel responsible for communicating with the media and public. In addition, the emergency plan states that company personnel not assigned to the site are utilized as members of the program. Additional information regarding the

retraining of emergency responders is located in Section 13.3C.15.16 of this SER. Appendix 5 of the LNP Emergency Plan identifies an Administrative Procedure, "Emergency Preparedness Training," that supports and implements Section O, "Radiological Emergency Response Training," of the LNP Emergency Plan. Additional information regarding Emergency Plan training can be found in Section 13.2.2 of this SER.

Technical Evaluation: [O.4.] {Appendix E, Section IV.F.1}

The staff finds that the LNP Emergency Plan adequately describes the training program for instructing and qualifying personnel who will implement radiological emergency response plans. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and meets the requirements of Appendix E to 10 CFR Part 50.

13.3C.15.6 *Training for Emergency Response Organization Management*

Technical Information in the Emergency Plan: [O.4.a] {Appendix E, Section IV.F.1(b)(i)}
Section O.4.a of the LNP Emergency Plan states that Directors, coordinators, and managers in the ERO receive training that includes emergency condition assessment and classification, notification systems and procedures, organizational interfaces, site evacuation, radiation exposure controls, offsite support, and recovery. Additional information regarding the retraining of emergency response management is located in Section 13.3C.15.16 of this SER. Additional information regarding Emergency Plan training can be found in Section 13.2.2 of this SER.

Technical Evaluation: [O.4.a] {Appendix E, Section IV.F.1(b)(i)}
The staff finds that the LNP Emergency Plan adequately describes the training program for instructing and qualifying directors, managers, and coordinators who will implement radiological emergency response plans. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and meets the requirements of Appendix E to 10 CFR Part 50.

13.3C.15.7 *Training for Accident Assessment Personnel*

Technical Information in the Emergency Plan: [O.4.b] {Appendix E, Section IV.F.1(b)(ii)}
Section O.4.c of the LNP Emergency Plan states that accident assessment personnel receive training that includes emergency condition assessment and classification, notification systems and procedures, and organizational interfaces. In response to RAI 13.3-40(2), the applicant proposed a revision, in part, to Section O.4 of the LNP Emergency Plan that includes a discussion regarding CR (operations) staff, including the STA, which will receive training in emergency condition assessment and classification, offsite dose assessment, site evacuation, and recovery operations. Additional information regarding the retraining of accident assessment emergency responders is in Section 13.3C.15.16 of this SER.

Technical Evaluation: [O.4.b] {Appendix E, Section IV.F.1(b)(ii)}
The staff finds that the additional information and proposed textual revisions provided in response to RAI 13.3-40(2) are acceptable because they conform to the guidance in NUREG-0654/FEMA-REP-1, and meet the applicable requirements of Appendix E to 10 CFR Part 50.

The staff confirmed that the proposed revisions to the emergency plan provided in response to RAI 13.3-40(2) have been incorporated into Revision 2 of the LNP Emergency Plan. The staff finds that the LNP Emergency Plan adequately describes specialized initial training for personnel responsible for accident assessment, including control room shift personnel. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and meets the requirements of Appendix E to 10 CFR Part 50.

13.3C.15.8 *Training for Radiological Monitoring and Analysis Personnel*

Technical Information in the Emergency Plan: [O.4.c] {Appendix E, Section IV.F.1(b)(iii)}
Section O.4.c of the LNP Emergency Plan states that radiological control personnel receive training that includes dose assessment, emergency exposure evaluation, protective measures, protective actions, contamination control and decontamination, monitoring systems, and procedures. Additional information regarding radiological analysis training specific to CR staff including the STA is located in Section 13.3C.15.7 of this SER. In response to RAI 13.3-45(4), the applicant stated that the Radiological Monitoring Team is responsible for evaluating the radiological conditions of the site boundary and beyond. The Radiological Monitoring Team is responsible for plume tracking, monitoring, and other sampling activities. The emergency plan (Section O.4) will be revised to specify the training for this team that will include the following topics: equipment checks, plume tracking and map reading, field measurement of airborne radioactivity, radiation levels and contamination in the EPZ, environmental sample collection, recordkeeping, communications, and procedures. Additional information regarding the retraining of radiological monitoring and analysis personnel is in Sections 13.3C.15.2 and 13.3C.15.16 of this SER.

Technical Evaluation: [O.4.c] {Appendix E, Section IV.F.1(b)(iii)}
The staff finds that the additional information and proposed textual revisions to the LNP Emergency Plan provided in response to RAI 13.3-45(4) are acceptable because they clarify the training content and scope for the team assigned to perform offsite radiation monitoring during an emergency. This conforms to the guidance in NUREG-0654/FEMA-REP-1. The staff created **Confirmatory Item 13.3-45(4)** to track the applicant's revision to the emergency plan.

Resolution of Confirmatory Item 13.3-45(4)

Confirmatory Item 13.3-45(4) is an applicant commitment to update the LNP Emergency Plan. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). As a result, Confirmatory Item 13.3-45(4) is now closed.

The staff finds that the LNP Emergency Plan adequately addresses the specialized initial training describing radiological monitoring and analysis personnel. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and meets the requirements of Appendix E to 10 CFR Part 50.

13.3C.15.9 *Training for Fire Fighting Teams*

Technical Information in the Emergency Plan: [O.4.d] {Appendix E, Section IV.F.1(b)(iv)}
Section O.4.d of the LNP Emergency Plan states that firefighting personnel receive training that includes the notification of station personnel, facility activation, personnel accountability and evacuation, and access control. In addition, Section O.4.f of the LNP Emergency Plan states that firefighting personnel receive training in emergency organizational interfaces, firefighting, search and rescue procedures, and communications systems. Section O.4.e of the LNP Emergency Plan describes training for firefighting personnel. Additional information regarding site-specific training and retraining for offsite firefighting personnel can be found in Section 13.3C.15.16 of this SER and Section 9.5.1.8.2.2, "Fire Brigade Training," of the LNP COL FSAR. Section 9.5.1.8.2.2 of the LNP FSAR provides supporting discussion regarding the individuals qualified to conduct fire brigade training, the scope of course content, classroom instruction and fire fighting techniques, refresher training, practice in fire fighting, and periodic fire drills.

Technical Evaluation: [O.4.d] {Appendix E, Section IV.F.1(b)(iv)}
The staff finds that the LNP Emergency Plan adequately describes the specialized initial and refresher training for firefighting personnel. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and meets the requirements of Appendix E to 10 CFR Part 50.

13.3C.15.10 *Training for Repair and Damage Control Teams*

Technical Information in the Emergency Plan: [O.4.e] {Appendix E, Section IV.F.1(b)(v)}
Section O.4.f of the LNP Emergency Plan states that Damage Control/Emergency Repair Teams receive training that includes information on the damage control organization, communication systems, and planning and coordination of damage control tasks. Additional information regarding the retraining of repair and damage control teams is in Section 13.3C.15.16 of this SER.

Technical Evaluation: [O.4.e] {Appendix E, Section IV.F.1(b)(v)}
The staff finds that the LNP Emergency Plan adequately describes the specialized initial and refresher training for repair and damage control teams. This is acceptable because it conforms to the guidance described in NUREG-0654/FEMA-REP-1 and meets the requirements of Appendix E to 10 CFR Part 50.

13.3C.15.11 *Training for Local Emergency Management Personnel*

Technical Information in the Emergency Plan: [O.4.g] {Appendix E, Section IV.F.1}
Initial and refresher training for local support services personnel including emergency services personnel is addressed in Section O.4.h of the LNP Emergency Plan.

Technical Evaluation: [O.4.g] {Appendix E, Section IV.F.1}
The technical evaluation of specialized initial and refresher training for local support services personnel including emergency service personnel is addressed in Section 13.3C.15.2 and 13.3C.15.16 of this SER.

13.3C.15.12 *Training for Medical Support Personnel*

Technical Information in the Emergency Plan: [O.4.h] {Appendix E, Section IV.F.1(b)(vii)}
Initial and refresher training for medical support personnel is addressed in Section O.4.j of the LNP Emergency Plan.

Technical Evaluation: [O.4.h] {Appendix E, Section IV.F.1(b)(vii)}
The technical evaluation of specialized initial and refresher training for medical support personnel is addressed in Sections 13.3C.15.2, 13.3C.15.4 and 13.3C.15.16 of this SER.

13.3C.15.13 *Training for Headquarters Support Personnel*

Technical Information in the Emergency Plan: [O.4.i] {Appendix E, Section IV.F.1(b)(viii)}
Section O.4 in Revision 1 of the LNP Emergency Plan states, in part, that Progress Energy conducts a program for instructing and qualifying all personnel and company personnel not assigned to the site that implement the emergency plan. In RAI 13.3-54, the staff requested the applicant clarify the specialized initial and periodic refresher training (including the scope, nature, and frequency) for corporate support personnel. In response the applicant stated, in part, that Company personnel that are not assigned to the site, such as corporate support personnel, may be members of the LNP ERO. However, all personnel regardless of whether they are assigned to the site or not, will receive the same training for the ERO designated position they are assigned per the emergency plan. Initial training and retraining is described in Sections O.4 and O.5 of the emergency plan. Additional information regarding the retraining of corporate emergency response personnel is located in Section 13.3C.15.16 of this SER.

Technical Evaluation: [O.4.i] {Appendix E, Section IV.F.1(b)(viii)}
The staff finds that the additional information and proposed textual revisions to the LNP Emergency Plan provided in response to RAI 13.3-54 related to the training and retraining of corporate support personnel to be acceptable because it clarifies the information in the emergency plan. This conforms to the guidance in NUREG-0654/FEMA-REP-1 and meets the requirements of Appendix E to 10 CFR Part 50. The staff created **Confirmatory Item 13.3-54(a)** to track the applicant's revision to the emergency plan provided in response to this RAI.

<u>Resolution of Confirmatory Item 13.3-54(a)</u>

Confirmatory Item 13.3-54(a) is an applicant commitment to update the LNP Emergency Plan. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). As a result, Confirmatory Item 13.3-54(a) is now closed.

The staff finds that the LNP Emergency Plan adequately describes the initial training and retraining for corporate support personnel. This is acceptable because it conforms to the guidance described in NUREG-0654/FEMA-REP-1 and meets the requirements of Appendix E to 10 CFR Part 50.

13.3C.15.14 *Training Related to Transmitting Emergency Information*

Technical Information in the Emergency Plan: [O.4.j]
Section O.4.j of Revision 1 of the LNP Emergency Plan states that Company personnel responsible for communicating with the media and public are trained prior to position assignment. In RAI 13.3-54(b), the staff requested the applicant provide a general discussion regarding the specialized initial and periodic refresher training (including the scope, nature, and frequency) for ENC or corporate communications personnel responsible for communicating with the media and public during an emergency. In response, the applicant stated, in part, that the Emergency News Coordinator responsible for communicating with the media is assigned to the ERO and receives initial and annual retraining. Training for communicating with the media includes: development and issuance of news releases, coordination and conduct of media briefings, rumor control, and media monitoring and correction of misinformation. In addition, Section O.4.i of the LNP Emergency Plan states emergency communicators receive training that includes notifications, reports to offsite authorities, and communication systems. Sections O.4.k and O.4.l describe these positions in Revision 6 of the LNP Emergency Plan. Additional information regarding the retraining of emergency response personnel responsible for transmitting emergency information is in Section 13.3C.15.16 of this SER.

Technical Evaluation: [O.4.j]
The staff finds that the additional information and proposed textual revisions to the LNP Emergency Plan provided in response to RAI 13.3-54(b) related to the training and retraining of personnel responsible for the transmission of emergency information to be acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and meets the requirements of Appendix E to 10 CFR Part 50. The staff created **Confirmatory Item 13.3-54(b)** to track the applicant's revision to the emergency plan provided in response to this RAI.

<u>Resolution of Confirmatory Item 13.3-54(b)</u>

Confirmatory Item 13.3-54(b) is an applicant commitment to update the LNP Emergency Plan. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). As a result, Confirmatory Item 13.3-54(b) is now closed.

The staff finds that the LNP Emergency Plan adequately describes the initial training and retraining of personnel responsible for the transmission of emergency information and instructions. This is acceptable because it conforms to the guidance described in NUREG-0654/FEMA-REP-1 and meets the requirements of Appendix E to 10 CFR Part 50.

13.3C.15.15 *Training for Security Personnel*

Technical Information in the Emergency Plan: {Appendix E, Section IV.F.1(b)(ix)}
Section O.4.e, "Emergency Response Training and Qualification," of the LNP Emergency Plan states that security personnel receive training that includes the notification of station personnel, facility activation, personnel accountability and evacuation, and access control. Additional information regarding the retraining of Security personnel is in Section 13.3C.15.16 of this SER.

Technical Evaluation: {Appendix E, Section IV.F.1(b)(ix)}
The staff finds that the LNP Emergency Plan adequately addresses the specialized training described for security personnel. This is acceptable because it meets the requirements of Appendix E to 10 CFR Part 50.

13.3C.15.16 *Retraining of Emergency Response Personnel*

Technical Information in the Emergency Plan: [O.5] {Appendix E, Section IV.F.1}
Section O.5, "Retraining," states that Progress Energy conducts or supports annual retraining for personnel with emergency response responsibilities, in accordance with the plant training program. Personnel that have not successfully completed this training as specified in plant training program requirements will be removed from the ERO pending completion of the required training.

Technical Evaluation: [O.5] {Appendix E, Section IV.F.1}
The staff finds that the LNP Emergency Plan adequately describes the provisions for retraining of personnel with emergency response responsibilities. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and meets the requirements of Appendix E to 10 CFR Part 50.

13.3C.15.17 *Conclusion*

The staff concludes that the information provided in the LNP Emergency Plan regarding radiological emergency training is acceptable and meets the requirements of 10 CFR 50.47(b)(15) because it conforms with the guidance in Evaluation Criterion O of NUREG-0654/FEMA-REP-1 and meets the applicable requirements of Appendix E to 10 CFR Part 50 as described above.

13.3C.16 Responsibility for the Planning Effort

13.3C.16.1 *Regulatory Basis*

In determining whether the proposed emergency plan met the applicable regulatory requirements in 10 CFR 50.47(b)(16) for responsibility for the planning effort, the staff evaluated it against the detailed evaluation criteria in NUREG-0654/FEMA-REP-1. The staff also evaluated the proposed emergency plan against applicable regulatory requirements related to the area of "Responsibility for the Planning Effort," in Appendix E to 10 CFR Part 50.

13.3C.16.2 *Training for Personnel Responsible for Planning Effort*

Technical Information in the Emergency Plan: [P.1]
Section P.1, "Training," of the LNP Emergency Plan states that Progress Energy implements a process to ensure the Emergency Preparedness Supervisor and supporting staff are properly trained for the effective implementation of the EP effort consistent with regulatory requirements and guidance, license conditions, other commitments, and accepted good practices. Training is primarily through on-the-job experience related to plan preparation, periodic revisions, or drills and exercises. Other training may include formal education, professional seminars, plant-specific training, and industry meetings.

Technical Evaluation: [P.1]
The staff finds that the LNP Emergency Plan adequately describes the training that will be provided for individuals responsible for the planning effort. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.16.3 *Person Responsible for Emergency Planning*

Technical Information in the Emergency Plan: [P.2]
Section P.2, "Responsibility for Radiological Emergency Response Training," of the LNP Emergency Plan states that the Vice President, Corporate Governance and Operations Support, has the overall authority and responsibility for ensuring that an adequate level of emergency preparedness is maintained. The EP Supervisor is delegated responsibility for the radiological emergency preparedness planning effort.

Technical Evaluation: [P.2]
The staff finds that the LNP Emergency Plan adequately identifies the individual, by title, with the overall authority and responsibility for radiological emergency response planning. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.16.4 *Designation of an Emergency Response Coordinator*

Technical Information in the Emergency Plan: [P.3]
Section P.3, "Emergency Planning Coordination," of the LNP Emergency Plan states that the Emergency Preparedness Supervisor is designated as the EP Coordinator and responsible for developing and updating the LNP Emergency Plan and for the coordination of LNP Emergency Plan with other response organizations.

Technical Evaluation: [P.3]
The staff finds that the LNP Emergency Plan adequately designates an EP Coordinator with responsibility for the development and updating of emergency plans and coordination of these plans with other response organizations. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.16.5 *Update and Maintenance of the Emergency Plan*

Technical Information in the Emergency Plan: [P.4] {Appendix E, Section IV.G}
Section P.4, "Plan Reviews and Updates," of the LNP Emergency Plan states that the emergency plan will be reviewed, updated, and certified to be current on an annual basis by the EP Coordinator. Revisions to the Plan will be reviewed in accordance with 10 CFR 50.54(q). Section P.9, "Emergency Plan Audits," identifies the Emergency Plan and implementing procedures, ERFs, equipment, and supplies to be within the scope of independent periodic audits. Section N.5 of the LNP Emergency Plan states that input captured from drill and exercise critiques will be used by Progress Energy to evaluate the need for changes to the LNP emergency Plan. In RAI 13.3-57, the staff requested the applicant clarify in the LNP Emergency Plan whether written agreements and implementing procedures are maintained up-to-date. In response, the applicant confirmed that in addition to the emergency plan, written agreements and EPIPs in support of the plan will be reviewed, updated, and certified to be current on an annual basis by the EP Coordinator. Written agreements shall be certified current annually.

Technical Evaluation: [P.4] {Appendix E, Section IV.G}
The staff finds that the additional information and proposed textual revisions to the LNP Emergency Plan provided in response to RAI 13.3-57 related to the update and maintenance of written agreements and EPIPs to be acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and meets the requirements of Appendix E to 10 CFR Part 50. The staff created **Confirmatory Item 13.3-57** to track the applicant's revision to the emergency plan provided in response to this RAI.

<u>Resolution of Confirmatory Item 13.3-57</u>

Confirmatory Item 13.3-57 is an applicant commitment to update the LNP Emergency Plan. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). As a result, Confirmatory Item 13.3-57 is now closed.

The staff finds that the LNP Emergency Plan adequately describes provisions for updating, and certifying the current emergency plan, written agreements, and EPIPs on an annual basis. In addition, the updating provisions described, take into account changes identified by drills and exercises. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1 and meets the applicable requirements in Appendix E to 10 CFR Part 50.

13.3C.16.6 *Distribution of Emergency Plans*

Technical Information in the Emergency Plan: [P.5]
Section P.5, "Distribution of Revised Plans," of the LNP Emergency Plan states, in part, that the EP Coordinator will incorporate changes to the emergency plan following its annual review. Changed pages will be marked and dated to highlight each change. Following approval of the updated plan by the Site Executive, the LNP document control organization will distribute the updated plan to those individuals or organizations responsible for its implementation.

Technical Evaluation: [P.5]
The staff finds that the LNP Emergency Plan adequately describes that the emergency response plans and approved changes to the plan will be forwarded to all organizations and appropriate individuals with responsibility for implementation of the plan. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.16.7 *Supporting Plans*

Technical Information in the Emergency Plan: [P.6]
Section P.6, "Supporting Plans," of the LNP Emergency Plan includes a list of plans that support the LNP Emergency Plan. In supplemental RAI 13.3-43, the staff requested the applicant include reference to the REMPs for Levy, Citrus, and Marion counties. In response, the applicant stated that the plans for these three counties will be incorporated into Section P.6 in a future revision to the LNP Emergency Plan. The applicant also committed to adding the three plans to Appendix 2, "References." Section L.1.3, "Off-site Medical Support Plans," of the LNP Emergency Plan which states that both Seven Rivers Regional Medical Center and Citrus Memorial Hospital have plans for emergency handling of radiation accident cases from the LNP to carry out the terms of the hospital's agreement with Progress Energy. In RAI 13.3-55, the staff requested that the applicant incorporate reference to these plans in the LNP Emergency Plan. In response, the applicant proposed to revise the emergency as recommended above.

Technical Evaluation: [P.6]
The staff finds that the additional information and proposed textual revisions to the LNP Emergency Plan provided in response to RAI 13.3-55 to be acceptable because they conform to the guidance in NUREG-0654/FEMA-REP-1. The staff created **Confirmatory Item 13.3-55** to track the applicant's revision to the emergency plan provided in response to this RAI. The staff also confirmed that the additional information and proposed textual revisions to the LNP Emergency Plan provided in response to RAI 13.3-43 have been incorporated into Revision 2.

Resolution of Confirmatory Item 13.3-55

Confirmatory Item 13.3-55 is an applicant commitment to update the LNP Emergency Plan. The staff verified that the LNP Emergency Plan was appropriately updated (or revised). As a result, Confirmatory Item 13.3-55 is now closed.

The staff finds that the LNP Emergency Plan adequately describes supporting emergency response plans. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.16.8 *Emergency Plan Implementing Procedures*

Technical Information in the Emergency Plan: [P.7]
Section P.7, "Implementing Procedures," of the LNP Emergency Plan states that changes to implementing procedures are developed and approved consistent with the requirements of

10 CFR 50.54(q) and the guidance provided in NRC Regulatory Information Summary 2005-02, "Clarifying the Process for Making Emergency Plan Changes." Appendix 5, "List of Emergency Plan Supporting Procedures," provides a list of implementing and administrative procedures that support and implement applicable sections of the emergency plan. In supplemental RAI 13.3-42, the staff requested the applicant provide additional clarification regarding some procedure titles, and the potential need for implementing procedures (e.g., security's emergency response role and ERO staff roles and responsibilities) referenced in the LNP Emergency Plan. In its response, the applicant has proposed clarification to Appendix 5 of the LNP Emergency Plan, including the addition of procedures titled, "Radiological Exposure Control," and "Duties of the LNP Security Organization." The applicant proposed EP ITAAC 15.1 to ensure that detailed implementing procedures for its emergency plan are submitted no less than 180 days prior to fuel load.

Technical Evaluation: [P.7]
The staff finds the applicant's response to supplemental RAI 13.3-42 to be acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1. The staff confirmed that the additional information and proposed textual revisions to the emergency plan provided in response to RAI 13.3-42 have been incorporated into Revision 2 of the LNP Emergency Plan. The staff finds that the LNP Emergency Plan adequately includes a listing of the procedures by title that are required to implement the emergency plan. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1. The staff's evaluation of EP ITAAC is provided in Section 13.3C.19 of this SER.

13.3C.16.9 *Table of Contents and Cross-Reference Table*

Technical Information in the Emergency Plan: [P.8]
Section P.8, "Table of Contents and NUREG-0654 Cross Reference," states, in part, that the LNP Emergency Plan includes a specific table of contents, and the format for the emergency plan directly follows the format of NUREG-0654/FEMA-REP-1. Appendix 8, "NUREG-0654 Cross Reference," of the emergency plan includes a cross-reference between the guidance provided in NUREG-0654/FEMA-REP-1, including specific acceptance criteria, and the LNP Emergency Plan. A cross-reference to Appendix E to 10 CFR Part 50, as specified in RG 1.206, C.I.13.3.1, "Combined License Application and Emergency Plan Content," is also included as supplemental information to Part 5 of the COL application.

Technical Evaluation: [P.8]
The staff finds that the LNP Emergency Plan adequately provides a table of contents and a cross-reference table to facilitate the use of the LNP Emergency Plan. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.16.10 *Annual Independent Review of the Emergency Plan*

Technical Information in the Emergency Plan: [P.9]
Section P.9, "Emergency Plan Audits," of the LNP Emergency Plan states that Progress Energy's Nuclear Oversight organization will perform or oversee independent audits of the LNP

EP Program consistent with the requirements of 10 CFR 50.54(t). Progress Energy establishes and maintains the frequency of the periodic audits based on an assessment of performance as compared to performance indicators; however, the audit frequency may not be less than once every 24 months. Programs audits are also performed as soon as possible but no longer than 12 months after a change occurs in personnel, procedures, equipment, and facilities that could adversely affect the status of EP. The minimum elements of the Emergency Preparedness Program, consistent with NUREG-0654/FEMA-REP-1; Evaluation Criterion P.9, included in the audit are outlined. Progress Energy's Nuclear Oversight organization will ensure that all audit findings are subject to management controls consistent with the facility's corrective action program. Results of the audit are sent to the LNP facility, Progress Energy management, and affected governments. The audit results, including recommended improvements, answers to the recommended improvements, and a description of the corrective actions taken, are maintained by records management for 5 years.

Technical Evaluation: [P.9]
The staff finds that the LNP Emergency Plan adequately describes arrangements for and the conduct of independent reviews of the emergency preparedness program at least every 12 months. This is acceptable because it conforms to the guidance in NUREG-0654/FEMA-REP-1.

13.3C.16.11 *Quarterly Update of Emergency Telephone Numbers*

Technical Information in the Emergency Plan: [P.10]
Section P.10, "Emergency Telephone Numbers," of the LNP Emergency Plan states that the EP Coordinator reviews telephone numbers in emergency response procedures quarterly and is responsible for ensuring required revisions are completed.

Technical Evaluation: [P.10]
The staff finds that the LNP Emergency Plan adequately provides for updating telephone numbers in emergency procedures at least quarterly. This is acceptable because it conforms to the guidance provided in NUREG-0654/FEMA-REP-1.

13.3C.16.12 *Conclusion*

The staff concludes that the information provided in the LNP Emergency Plan regarding the responsibility for EP is acceptable and meets the requirements of 10 CFR 50.47(b)(16) because it conforms with the guidance in Evaluation Criterion P of NUREG-0654/FEMA-REP-1 and meets the applicable requirements of Appendix E to 10 CFR Part 50 as described above.

13.3C.17 Security-Based Event Considerations

13.3C.17.1 *Regulatory Basis*

NUREG-0800, Chapter 13.3, "Emergency Planning," specifies that applicants for a COL address the information in the Commission Orders issued February 25, 2002, as well as any subsequent NRC guidance, to determine what security-related aspects of EP and preparedness should be addressed in the emergency plan.

NUREG-0800, the Commission Orders issued February 25, 2002, and security-related enhancements identified in NRC Bulletin 2005-02, “Emergency Preparedness and Response Actions for Security-Based Events,” identify the following areas that applicants should consider in the COL application, emergency plan, or implementing procedures:

1. Security-based emergency classification levels and EALs - The emergency plan, or implementing procedures includes EALs to ensure that a site-specific, security event results in an emergency classification declaration of at least a notification of unusual event. The classification scheme should also reflect the strategy for escalation to a higher-level event classification.

2. NRC Notifications - Notification procedures allow for NRC notification of safeguards events immediately after notification of LLEAs, or within about 15 minutes of the recognition of a security-based threat.

3. Onsite Protective Measures - Consideration has been given to a range of protective measures for site workers, as appropriate, during a security-based event (e.g., evacuation of personnel from target buildings, site evacuation by opening security gates, dispersal of licensed operators, sheltering of personnel in structures away from potential site targets, and arrangements for accounting for personnel after attack).

4. ERO Augmentation - ERFs and alternative facilities have been identified to support the rapid response from ERO members to mitigate site damage from a security-based event once the site is secured. The alternative facilities could likely be located outside of the PA and should include the following characteristics: accessible even if the site is under threat or actual attack; communication links with the EOF, CR and plant security; the capability to perform offsite notifications; and the capability for engineering assessment activities, including damage control team planning and preparation. The alternative facility should also be equipped with general plant drawings and procedures, telephones, and computer links to the site.

5. Potential Vulnerabilities from Nearby Hazardous Facilities, Dams, and other Sites - 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.

6. Drills and Exercises - Emergency Preparedness drill and exercise programs maintain the key skills necessary for mitigating security-based events. The ERO demonstrates security-based emergency preparedness program activities under the schedule as committed to in its emergency plans.

7. Emergency Preparedness and Response to a Security-based Event - Onsite staffing, facilities, and procedures are adequate to accomplish actions necessary to respond to a security-based event, and the Emergency Plan and implementing procedures reflect the site-specific needs.

By letter dated November 8, 2012, from PEF to the NRC, the applicant provided its response to address the Final Rule on Enhancements to Emergency Preparedness Regulation effective November 23, 2011. The staff's review included the guidance in NSIR/DPR-ISG-01 and the requirements in Appendix E to 10 CFR Part 50.

13.3C.17.2 *Security-Based Emergency Classification and Emergency Action Levels*

Technical Information in the Emergency Plan: (NUREG-0800)
Emergency classifications and action levels for security or hostile action based events are included in the EALs addressed in Section 13.3C.4 of this SER.

Technical Evaluation: (NUREG-0800)
The staff's evaluation of the LNP emergency classification and action level scheme is included in Section 13.3C.4 of this SER.

13.3C.17.3 *NRC Notification*

Technical Information in the Emergency Plan: (NUREG-0800)
In RAI 13.3-23(C), the staff asked the applicant to describe how the LNP Emergency Plan addressed emergency preparedness for security-based events as outlined in NRC Bulletin 2005-02. The applicant's response, in part, referenced implementing procedures that provide instructions for notification to Federal authorities that includes an accelerated call to the NRC. In supplemental RAI 13.3-37(2), the staff asked the applicant to clarify in the emergency plan the notification to the NRC of hostile-action based events immediately after notification of local law enforcement agencies, or within about 15 minutes following its recognition. In response, the applicant stated, in part, that they will revise the LNP Emergency Plan to add direction to notify the NRC within about 15 minutes immediately after notification of local law enforcement in the event of a hostile-based threat against LNP. In addition, the applicant stated that specific actions to complete the NRC notification will be included in EPIPs.

Technical Evaluation: (NUREG-0800)
The staff finds the additional information and proposed textual revisions to the emergency plan provided in response to supplemental RAI 13.3-37(2), in consideration of its response to RAI 13.3-23(C), to be acceptable because it provided instructions for an accelerated call to the NRC (within 15 minutes) immediately after notification of local law enforcement in the event of a security-based or hostile action event. This is acceptable because it conforms to the guidance in NRC Bulletin 2005-02 and the specific evaluation criteria in NUREG-0800. The staff confirmed that the applicant's responses to these RAIs are incorporated into Revisions 1 and 2 of the LNP Emergency Plan. Therefore, the staff finds that the LNP Emergency Plan adequately describes provisions for an accelerated call to the NRC in the event of a

hostile-based threat against LNP. This is acceptable because it conforms to the guidance in NRC Bulletin 2005-02 and the specific evaluation criteria in NUREG-0800.

13.3C.17.4 *Onsite Protective Measures during a Security-Based Event*

Technical Information in the Emergency Plan: (NUREG-0800) {Appendix E, Section IV.I}
Section J.5, "Personnel Accountability," of the LNP Emergency Plan states that assembly and accountability may be delayed during a security event, if the EC (in consultation with Security) determines that performing accountability could be detrimental to the safety of site personnel. If accountability is delayed, then accountability should be performed immediately when conditions warrant. In RAI 13.3-23(C), the staff requested additional information from the applicant regarding onsite protective measures during a security-based event. In response, in part, the applicant provided clarification of the personnel accountability process, including a description of the decision-making process by the EC with input from Security to protect onsite personnel during a site security event. The applicant stated, in part, that the EC may direct protective measures including:

- evacuation of site personnel;
- site evacuation while continuing to defend security gates;
- dispersal of key personnel;
- onsite sheltering;
- staging of ERO personnel in alternate locations pending the restoration of safe conditions; or
- implementation of accountability measures following restoration of safe conditions.

Technical Evaluation: (NUREG-0800) {Appendix E, Section IV.G}
The staff finds the additional information and proposed textual revisions to the emergency plan provided in response to RAI 13.3-23(C) to be acceptable because they describe onsite protective measures, other than evacuation, that can be taken during a security-based event. This is acceptable because it conforms to the guidance in NRC Bulletin 2005-02, the specific evaluation criteria in NUREG-0800 and NSIR/DPR-ISG-01, and meets the requirements in Appendix E to 10 CFR Part 50. The staff confirmed that the information provided in response to this RAI was incorporated into Revision 1 of the LNP Emergency Plan.

By letter dated November 8, 2012, from PEF to the NRC, the applicant provided its response to address the Final Rule on Enhancements to Emergency Preparedness Regulation effective November 23, 2011. In regard to implementation of the EP rule pertaining to the protective actions for onsite personnel, there were no changes warranted for the LNP Emergency Plan. Therefore, the staff finds that the LNP Emergency Plan adequately describes onsite protective measures necessary to respond to a security-based event and to ensure the continued ability of the licensee to safely shut down the reactor, while performing the functions of the licensee's emergency plan. This is acceptable because it conforms to the guidance in NRC Bulletin 2005-02, the specific evaluation criteria in NUREG-0800, and NSIR/DPR-ISG-01, and meets the requirements in Appendix E to 10 CFR Part 50.

13.3C.17.5 *Emergency Response Organization Augmentation*

Technical Information in the Emergency Plan: (NUREG-0800)
Section E.1.1, "Progress Energy Emergency Response Organization," states that notifications of an emergency will be made to personnel assigned to the ERO, and if the emergency involves a security threat, alternate assembly areas may be used to protect the responding ERO members. In RAI 13.3-23(C), the staff requested additional information from the applicant regarding ERO augmentation during a security-based event. In response, in part, the applicant provided reference to an EPIP that includes additional instruction on assembly, protective actions and response to an alternate assembly area for responding ERO personnel, if required. In supplemental RAI 13.3-37(1), the staff asked the applicant to describe in the emergency plan an alternative facility to support rapid response to a hostile-action event with functionality similar to the EOF. In response, the applicant stated that the EOF/ENC is the alternate ERF. The proposed revision to the emergency plan will address the characteristics needed for an alternate facility to support the rapid response to a severe weather event, hostile-action event, or any other situation that prevents the LNP ERO from responding to normal onsite facilities. In addition, the applicant provided reference to an EPIP that will be added to Appendix 5 of the LNP Emergency Plan titled, "Activation and Operation of the Alternate Emergency Response Facility," which will provide specific setup criteria for this facility. Additional information regarding ERO augmentation at an alternate facility is in 13.3C.8.39 of this SER.

Technical Evaluation: (NUREG-0800)
The staff finds the additional information and proposed textual revisions to the emergency plan provided in response to supplemental RAI 13.3-37(1), in consideration of its response to RAI 13.3-23(C), to be acceptable because it describes an alternate facility and functionality to support the augmentation of ERO personnel and rapid response to a security-based or hostile action event. The staff confirmed that Revision 6 of the LNP Emergency Plan contained the proposed text revisions. This is acceptable because it conforms to the guidance in NRC Bulletin 2005-02 and the specific evaluation criteria in NUREG-0800. The staff confirmed that the information provided in response to these RAIs have been incorporated into Revisions 1 and 2 of the LNP Emergency Plan. Additional information regarding ERO augmentation at an alternate facility is in 13.3C.8.39 of this SER. Therefore, the staff finds that the LNP Emergency Plan adequately describes provisions for use of an alternate facility to support augmentation of ERO personnel and the rapid response to a security-based or hostile action event. This is acceptable because it conforms to the guidance in NRC Bulletin 2005-02 , the specific evaluation criteria in NUREG-0800, and NSIR/DPR-ISG-01, and meets the requirements in Appendix E to 10 CFR Part 50.

13.3C.17.6 *Potential Vulnerabilities from Nearby Hazardous Facilities, Dams, and Other Sites*

Technical Information in the Emergency Plan: (NUREG-0800)
Part 2, "FSAR," of the LNP COL application, Section 2.2, "Nearby Industrial, Military, and Transportation Facilities," provides information regarding the potential effect on the plant from damage to nearby hazardous facilities, dams, and other nearby sites. Section J.10, "Protective

Measures Implementation," of the LNP Emergency Plan states that evacuation routes are illustrated in Figure A.6-2, "Levy Evacuation Routes and Shelters." Appendix 5 provides reference to an EPIP for evacuation and accountability of personnel. In supplemental RAI 13.3-37(3), the staff asked the applicant to clarify whether the potential to onsite staffing with augmentation and evacuation strategies, in consideration of a security event from damage to nearby hazardous facilities, dams, and other nearby sites, have been considered in the LNP Emergency Plan. In response, the applicant stated, in part, that the LNP Emergency Plan adequately addresses the ability to classify, notify, and augment staff during emergencies regardless whether the initiating condition originates onsite or offsite. The applicant provided reference to the LNP emergency classification and action level scheme as the means to be used for classifying such an emergency. In addition, the applicant stated that when an emergency classification is deemed necessary that requires activation of the LNP ERO the emergency facilities would be staffed accordingly:

1. When ERO personnel are onsite as is the case during a normal work day, the onsite facilities would be staffed as normal. An event at a nearby site is unlikely to cause an immediate health concern or nuclear safety concern preventing personnel from commuting to onsite facilities such as the TSC or OSC. Ventilation systems and other onsite protective measures protect the staff upon arrival.

2. When ERO personnel are offsite as is typical during night time and weekends, notification is made to personnel to respond to the onsite facilities as normal. In the event access to the site is deemed hazardous, the ERO is notified to respond to the alternate ERF.

Notification and mobilization of the ERO is discussed in Section E of the LNP Emergency Plan. In addition, the applicant provided reference to Section J, "Protective Response," within the LNP Emergency Plan, which provides additional direction to evacuate, relocate, stage, disperse, or shelter personnel onsite based on the hazard present regardless of the origination source.

Technical Evaluation: (NUREG-0800)
The staff's evaluation of the potential effect on the physical plant resulting from damage to offsite hazardous facilities, dams, and other nearby sites is located in Chapter 2 of this SER. Section 13.3C.4 of this SER includes the staff's evaluation regarding the applicant's means and methodology for classifying an emergency that initiates offsite. As described in Section 13.3C.17.5 in this SER, the applicant identifies an alternate facility (EOF/ENC) that will serve as a location for ERO members to assemble and activate in the event that access to the plant's onsite ERF locations are not accessible due to a severe weather event, hostile-action or any other reason. In response to RAI 13.3-37(3), the applicant provided additional clarification regarding the use of an alternate facility (EOF) for the protection of ERO personnel responding to an emergency at LNP. The staff finds the additional clarification provided in response to RAI 13.3-37(3) acceptable because it conforms to the guidance provided in NRC Bulletin 2005-02 and acceptance criteria in NUREG-0800. In addition, the applicant provides reference to, and the LNP Emergency Plan includes, protective strategies described in Section 13.3C.17.4 of this SER for the protection of onsite personnel and responding ERO

members. In response to RAI 13.3-37(3), the applicant did not propose any textual revisions to the emergency plan. The staff finds this acceptable. Therefore, the staff finds that the information LNP Emergency Plan adequately describes the assessment of other nearby hazards that could potentially affect the safety of the LNP facility. This is acceptable because it conforms to the guidance in NRC Bulletin 2005-02 and specific evaluation criteria in NUREG-0800.

13.3C.17.7 *Security-Based Drills and Exercises*

Technical Information in the Emergency Plan: (NUREG-0800)
Section N.1.a, "Exercise Scope and Frequency," states, in part, that provisions for drills and exercises using terrorist based events are part of the Drill and Exercise Program.

Technical Evaluation: (NUREG-0800)
The staff finds that the LNP Emergency Plan adequately describes the consideration for terrorist-based events in the LNP Drill and Exercise Program. This is acceptable because it conforms to the guidance in NRC Bulletin 2005-02 and specific evaluation criteria in NUREG-0800.

13.3C.17.8 *Emergency Preparedness and Response to a Security-Based Event*

Technical Information in the Emergency Plan: (NUREG-0800)
Sections 13.3C.2 and 13.3C.8 of this SER provides reference to information regarding the onsite and offsite EROs described in the LNP Emergency Plan, including the identification of minimum on shift and augmented staffing levels which would support activation of the ERO and associated ERFs in the event of a declared security-based event at the LNP site.

In addition, Sections 13.3C.17.2 through 13.3C.17.7 of this SER provides additional information regarding the applicant's ability to classify an emergency based on a security-related event; make an accelerated notification to the NRC; provide for protection of onsite ERO responders; assemble the augmented ERO staff at an alternate facility in support of rapid response should unsafe site conditions exist; and practice the ERO's response to a security-related event.

Appendix 5 of the LNP Emergency Plan includes a listing of EPIPs that encompass the spectrum of response activities associated with EP and security (non-safeguards) at the LNP site.

Technical Evaluation: (NUREG-0800)
The staff finds that the LNP Emergency Plan adequately describes emergency planning and response to a security-based or hostile action event at LNP. This is acceptable because it conforms to the guidance in NRC Bulletin 2005-02 and specific evaluation criteria in NUREG-0800.

13.3C.17.9 *Conclusion*

The staff concludes that the LNP Emergency Plan adequately addresses the preparation and response to security-based or hostile action events. This is acceptable because it conforms to the guidance in NRC Bulletin 2005-02, specific evaluation criteria in NUREG-0800, and NSIR/DPR-ISG-01, and meets the applicable requirements of Appendix E to 10 CFR Part 50 as described above.

13.3C.18 Evacuation Time Estimate (ETE) Analysis

The LNP Emergency Plan includes an analysis of the time required to evacuate the plume exposure pathway EPZ. The report titled "Levy Nuclear Plant Development of Evacuation Time Estimates," Revision 5, dated February 2011, (ETE Report) was provided as a separate document in the COL application as Appendix 5, "Evacuation Time Estimate Study." The Pacific Northwest National Laboratory and the Sandia National Laboratory assisted the staff in performing a technical review of the ETE Report. The ETE Report includes analyses and responses to RAIs and provides the basis for the staff's conclusions as to the adequacy of its content and conformity with Appendix 4 to NUREG-0654/FEMA-REP-1. The staff notes that the Crystal River Nuclear Plant (CRNP), located within the Levy plume exposure pathway EPZ has permanently ceased operations, initiated decommissioning, and has been exempted from specific EP standards as discussed in Section 13.3.4, including the requirement in 10 CFR Part 50, Appendix E, Section IV.5 to have an EPZ and to update the ETEs. Therefore, the historic RAIs in this section related to CRNP are no longer relevant.

13.3C.18.1 *Regulatory Basis*

The staff considered the following regulatory requirements and guidance in the review of the ETE analysis:

- 10 CFR 52.79(a)(21) refers to Appendix E to 10 CFR 50, Section IV, "Content of Emergency Plans," of which requires that the nuclear power reactor operating license applicant provide an analysis of the time required to evacuate and for taking other protective actions for various sectors and distances within the plume exposure pathway EPZ for transient and permanent populations.

The staff evaluated the ETE Report against Appendix 4, "Evacuation Time Estimates within the Plume Exposure Pathway Emergency Planning Zone," to NUREG-0654/FEMA-REP-1. Appendix 4 includes detailed guidance that the staff considered in determining whether the ETE analysis meets the applicable regulatory requirements in Appendix E to 10 CFR Part 50.

13.3C.18.2 *Introductory Materials Related to the ETE Report*

Technical Information in the ETE Report: [Section I of Appendix 4]
Section 1, "Introduction," of the ETE Report provides a basic description of the process used to determine the ETEs for the proposed LNP site. A description of the LNP site location, including

a map (Figure 1-1, "Levy Nuclear Plant Site Location"), illustrating the plume exposure pathway EPZ and surrounding area is provided. In RAI 13.3-2(A), the staff requested additional information regarding the lack of elevations, surrounding communities, and political boundaries identified on the map. In response, the applicant revised the map in Figures 1-1 and 3-1 to include counties within the plume exposure pathway EPZ and their boundaries. The applicant also revised the text in Section 1.2, “The Levy Nuclear Plant Location,” for clarification of these figures.

Section 2, “Study Estimates and Assumptions,” provides the basis for the population data estimates used in the ETE. Population estimates are based on the 2000 census data using the ArcGIS Software and the block centroid method. Estimates of employee and special facility populations are based on data provided by county emergency management officials. Vehicle occupancy factors are based on a statistical analysis of data acquired from a telephone survey. Additional assumptions regarding the development of population estimates, including pass-through populations and regional employees, are provided in Section 3, “Demand Estimation,” and Appendix E, “Special Facility Data.” Assumptions about transit-dependent and special populations are provided in Section 8, “Transit-Dependent and Special Facility Evacuation Time Estimates,” and Appendix E of the ETE Report. Development of trip generation times from survey responses is described in Section 5, “Estimation of Trip Generation Times.”

Eleven study assumptions used as the basis for the calculation of the ETEs are provided in Section 2.3, “Study Assumptions,” of the ETE Report. This study assumes that everyone will evacuate according to assigned evacuation routes. Schools will be notified in advance of the general population and given priority for use of transportation resources. Buses that are not being used for school evacuation will be used to transport those without access to private vehicles. Additional information regarding bus capacity assumptions was requested in RAI 13.3-3(F). In response, the applicant stated that the ETE Report assumes that there are 22 to 24 seats in most school buses in which 8 seats could accommodate 15 patients, leaving 14 to 16 seats for stacking of wheelchairs and patients’ personal items.

Traffic control points (TCPs) and access control points (ACPs) will be established to aid the flow of traffic out of the plume exposure pathway EPZ. Additional information was requested in RAI 13.3-3(C) to determine what effect traffic control will have on evacuation times. In response, the applicant stated that that the ETE Report assumes that the capacity estimates presented in Appendix K are not enhanced nor compromised by the establishment of a TCP at an intersection. The establishment of TCPs is recommended to provide guidance and reassurance to evacuees of the appropriate actions to take and route information, in addition to providing fixed point surveillance of evacuation activities. The applicant stated that there would be no effect on the ETE if TCPs were not established.

Voluntary and shadow evacuations are considered potential impediments to the overall evacuation effort. In RAIs 13.3-8(A) and 13.3-9(B), the staff requested clarification regarding why Lake Rousseau was part of the shadow region and not included in one of the protective action zones (PAZs). In response, the applicant stated that Lake Rousseau was not included as part of the Shadow Evacuation Region. The applicant revised the ETE Report to reflect Lake Rousseau is within PAZs C3, C4, L5, L6, and M9; and the PAZ boundaries now follow the county boundaries. In addition, the applicant stated that the transients visiting Lake Rousseau have been accounted for as part of the EPZ population and no changes to the analysis are needed.

In RAI 13.3-8(C), the staff requested information on how voluntary evacuees were addressed in Table 6-3, "Percent of Population Groups for Various Scenarios." In its response, the applicant stated that the numbers presented in Tables 6-3 and 6-4 are for a 100 percent evacuation of the full EPZ (Region R03). The applicant added a footnote to Table 6-4 for clarification and included a new Table H-1, "Percent of PAZ Population Evacuating for each Region," in the ETE Report that identifies the voluntary evacuation percentages for each PAZ for each regional configuration. In addition, the applicant revised the text of page H-1 of the ETE Report to include a discussion of Table H-1. The applicant further stated that a review of the input streams to interactive dynamic evacuation (IDYNEV) indicated that the voluntary evacuation percentages were not properly specified for any region, except Region R03. PAZs C1 and C3 were originally included in the 5-mile evacuation. Based on comments received during the review process, PAZ C1 and C3 were removed from the 5-mile evacuation. Tables 6-1, 7-2 and J-2, as well as the figures in Appendix H were revised; however, the input stream was not modified accordingly. The applicant corrected these percentages to show the values in Table H-1 and recomputed the ETE. The ETE values presented in the executive summary in Tables 7-1A through 7-1D and Tables J-1A through J-1D were updated based on these changes. I-DYNEV was modified to allow for the input of specific bus routes speed. This new feature of I-DYNEV was used to compute the average speed during evacuation on each of the school and transit-dependent bus routes servicing the EPZ. The average speeds discussed in Section 8.4 of the ETE Report were updated accordingly. Tables 8-5A and 8-5B, and Tables 8-7A and 8-7B, were also updated accordingly. Pages ES-11 and ES-12 in the Executive Summary were revised to reflect the new information.

In RAI 13.3-9(C), the staff requested clarification on assumptions regarding the "shadow" population that is expected to evacuate and the numbers of vehicles that were proposed to be used. In response, the applicant revised the text in Section 7.1, "Voluntary Evacuation and Shadow Evacuation," to identify the population within the Shadow Region and the methodology used to compute that estimate.

In RAI 13.3-14(F), the staff requested clarification on how the data in Figure F-11, "Time to Prepare Home for Evacuation," was used in development of the ETE. In its response, the applicant stated this distribution was "truncated" to avoid the bias of those few stragglers who take significantly longer to mobilize. In "truncating" these distributions, the mobilization of the stragglers is advanced. Therefore, the stragglers are not eliminated from the ETE. Additional information was provided in response to RAI 13.3-3(B).

In RAI 13.3-9(A), the staff requested clarification on whether a densely populated area, Dunnellon and Citrus Springs, was bisected by this boundary, and if so, to provide a resolution for the boundary of these zones. In response, the applicant stated the boundaries were developed in conjunction with the offsite authorities (State of Florida and EPZ counties) along well-defined features that would be easily identifiable to area residents and that would conform to an EPZ radius of about 10 miles. The PAZ boundaries, as defined, adhere to NRC guidelines and will be maintained.

An outline of the approach to estimating the ETE is presented with a link-node map [Figure 1-2, "Levy Nuclear Plant Link-Node Analysis Network"] of the highway network developed through the use of GIS mapping software and field observations. Details of the link-node map are presented in Appendix K, "Evacuation Roadway Network Characteristics." The IDYNEV System was used to analyze the highway network to determine routes used for evacuation and estimate evacuation times. A description of the IDYNEV System and associated sub-models is provided in Section 1.3, "Preliminary Activities," of the ETE Report. The IDYNEV system consists of several sub-models - a macroscopic traffic simulation model, an intersection capacity model, and a dynamic, node-centric routing model that adjusts the "base" routing in the event of an imbalance in the levels of congestion on the outbound links. Another model of the IDYNEV System is the traffic assignment and distribution model, which integrates an equilibrium assignment model with a trip distribution algorithm to compute origin-destination volumes and paths of travel designed to minimize travel time. A discussion of algorithms used is provided in detail in Section 4, "Estimation of Highway Capacity." Additional information on algorithms used in the estimations was requested in RAIs 13.3-4(A)(B)(C)(D)(E).

In RAI 13.3-4(A), the staff requested a general description of other important algorithms used in the traffic simulation model. In response, the applicant stated that Appendices B through D of the ETE Report provide additional detail on the IDYNEV system and its use in computing ETEs. The applicant revised page 1-6 of the ETE Report to include references to other documents that can be accessed for additional information. In RAI 13.3-4(B), the staff requested a discussion on how certain intersections will be controlled by traffic control personnel and how this may affect the variable in the equation, and/or intersection capacity, and the traffic simulation model. In response, the applicant stated the ETE calculations do not rely upon any of the traffic control measures in Appendix G of the ETE Report. The estimates of capacity used by the IDYNEV model are based on the factors described in Section 4, "Estimation of Highway Capacity," of the ETE Report and observations made during the road survey. It is assumed that these capacity estimates are not enhanced nor compromised by the establishment of a TCP at an intersection. The values of the variables in the intersection algorithm in Section 4 were derived by applying the IDYNEV system as an analysis tool rather than as a single "pass-through" calculation of an ETE. The applicant revised Item 7 in Section 2.3; and the text in Section 9 and page G-1 to clarify the use of ACPs and TCPs. In RAI 13.3-4(C), the staff requested values, or a range of possible values, for the parameters in the equation, where applicable, including "Mean Duration of Green Time," and "Mean Queue Discharge;" clarification on whether these values are estimated or field verified; and a discussion on how this equation is applied to staffed intersections where traffic control is in place. In response, the applicant provided additional information related to the parameters used in the equations in Section 4. Clarification on how these equations were applied to staffed intersections was also provided. The applicant included a new section, "Simulation and Capacity Estimation," at the end of Section 4 of the ETE Report for further clarification. In RAI 13.3-4(D), the staff requested a description of how the values for each variable in Section 4 were derived. For example, on page 4-2, the variables F1 and F2 are only defined as the various known factors that influence the turn-movement-specific mean discharge headway h_m. In response, the applicant provided additional information related to the variables F1 and F2. The applicant stated that this level of detail is not appropriate for inclusion in an ETE Report. The applicant revised the text on page 4-3 to include reference to Chapters 16 and 17 of the Highway Capacity Manual (HCM) where additional information can be found.

Further details on the use of traffic models is provided in Appendix C, "Traffic Simulation Model: PC-DYNEV," and Appendix D, "Detailed Description of Study Procedure" of the applicant's ETE Report.

Technical Evaluation: [Section I of Appendix 4]
The staff finds the additional information and proposed textual revisions provided by the applicant in response to RAIs 13.3-2(A), 13.3-3(C), 13.3-3(F), 13.3-4(A), 13.3-4 (B), 13.3-4(C), 13.3-4(D), 13.3-4(E), 13.3-8(A), 13.3-8(B), 13.3-8(C), 13.3-9(A), 13.3-9(B), 13.3-9(C), and 13.3-14(F), to be acceptable because they meet the requirements of Appendix E, Section IV to 10 CFR Part 50 and conform to the guidance in Appendix 4 to NUREG-0654/FEMA-REP-1. The staff confirmed that the changes proposed in the RAIs above have been incorporated into Revision 4 of the LNP ETE Report. The staff finds that the LNP ETE Report includes a map showing the proposed site and plume exposure pathway EPZ, as well as transportation networks, topographical features, and political boundaries. Also, the boundaries of the plume exposure pathway EPZ, in addition to the evacuation subareas within the plume exposure pathway EPZ, are based on factors such as current and projected demography, topography, land characteristics, access routes, and jurisdictional boundaries. The ETE Report also describes the method of analyzing the evacuation times. A general description of the evacuation model was provided including the assumptions used in the ETE analysis. Therefore, the information provided in the introductory materials of the LNP ETE Report meets the requirements of Appendix E, Section IV to 10 CFR Part 50 and conforms to the guidance in Appendix 4 to NUREG-0654/FEMA-REP-1.

13.3C.18.3 *Demand Estimation*

Technical Information in the ETE Report: [Section II of Appendix 4]
Section 3, "Demand Estimation," provides an estimate of demand expressed in terms of people and vehicles. The permanent resident population was projected out to 2007 by comparing the 2005 census data with the 2000 census data to obtain growth rates for each county. Based on information obtained in a telephone survey, the permanent resident average household size is estimated at 2.25 persons per household. In RAI 13.3-5(A), the staff requested additional information on the correct value to use for the average number of vehicles per household and whether Table 3-2 and Figure 3-3 would require updating if the number of vehicles per household is changed. In response, the applicant stated that both the 1.32 value used for vehicles per household in Table 1-1 and the 1.37 value in Figure F-8 were incorrect. The applicant stated the correct value is 1.39 as shown in Table 1. Using the correct value (1.39) results in a 5.3 percent increase in permanent resident vehicles, which should not significantly affect evacuation estimates. The applicant revised the number of evacuating vehicles per

household from 1.32 and 1.37 to 1.39 in the ETE Report; re-computed the number of evacuating vehicles for permanent residents and the Shadow Region; re-ran all the ETE scenarios using the updated vehicle estimates; and updated various tables and figures in the ETE Report to reflect the revised results.

Estimates of the permanent resident population and their vehicles are presented for each PAZ in Table 3-2, "Permanent Resident Population and Vehicles by PAZ," and by polar coordinate representation in Figures 3-2, "Permanent Resident by Sector," and Figure 3-3, "Permanent Resident Vehicles by Sector."

In RAI 13.3-5(D)(1), the staff requested clarification as to whether Table 6-4 of the ETE Report represented an evacuation of Region R03 (entire EPZ). In response, the applicant added a footnote to Table 6-4 stating, "The values presented are for an evacuation of the full EPZ (Region R03)." In RAI 13.3-5(D)(2), the staff requested a discussion of the county-specific growth rates used to obtain the permanent resident population and shadow population expanded to the year 2017 for Scenario 11. In response, the applicant provided the growth rates for Citrus, Levy, and Marion Counties obtained from the County Planning Departments. In RAI 13.3-5(D)(3), the staff requested clarification on how the values for residents with commuters, residents without commuters, and shadow were developed for Scenario 11 in Table 6-4. In response, the applicant stated that the values for residents with commuters, residents without commuters and shadow presented for Scenario 11 in Table 6-4 are overstated. Also, the peak construction date has shifted outward to year 2019. The simulations were re-run for all construction cases to correct the projection error and to update the peak construction year to 2019: Table 6-4, Tables 7-1A through 7-1D, Tables J-1A through J-1D, Figure J-11, and Tables 7-1C and 7-1D in the Executive Summary were revised to reflect the new simulation results. The applicant stated that the external traffic values shown in Table 6-4 are hourly volumes and will be expressed as total vehicles over the 90 minutes following the advisory to evacuate. The applicant revised the text on page 3-13 to reflect this change. In RAI 13.3-5(D)(4), the staff requested an explanation on why no additional transit buses or external traffic would be anticipated if a 60 percent growth increase is expected. In response, the applicant stated that the construction projections and 60 percent population growth were overstated in the ETE Report, and that the vehicles should be extrapolated to the peak construction year of 2019. Changes were incorporated into the IDYNEV input stream and all Scenario 11 cases were re-run. Tables 7-1A through 7-1D (Tables 7-1C and 7-1D also appear in the Executive Summary), Tables J-1 A through J-1D and Figure J-11, were updated based on this change. The discussion on page 3-2 of the ETE Report and the footnote to Table 6-4 pertaining to construction were also revised to reflect this change.

It is estimated that 1,416 people makeup the transient population. Individual activity vehicle occupancy factors were used to estimate average vehicle occupancy of 1.63 transient per vehicle. In RAI 13.3-6(A), the staff requested verification that the correct value for the transient population (1,416 versus 1,417) was used. In response, the applicant stated Figure 3-4 indicates 89 people visiting the Inglis Dam Recreation Area where the table on Page E-6 states 90 transients. Figure 3-4 was updated to agree with the table on page E-6. Also, the text on page 3-7 was updated to read "1,417 people." In RAI 13.3-6(B), the staff requested clarification of the logistics for evacuation of the lake and gulf coast areas. In response, the applicant stated that the warnings and evacuation of waterways will be conducted by various State and local organizations (e.g., Fish and Wildlife Conservation Commission and Florida Department of Environmental Protection, Division of Law Enforcement). Once the transients return to the mainland, they will evacuate using private vehicles on the evacuation routes identified in Section 10 of the ETE Report. In RAI 13.3-6(C), the staff requested an explanation regarding whether consideration was given for the possibility of transients returning to a location to gather their belongings. In response, the applicant stated that some transients will evacuate immediately while others may return to the lodging facility to gather up belongings and then evacuate. Figure 5-1 and the text in Section 5 were revised to include the possibility that transients may return to lodging facilities or campsites prior to beginning their evacuation trip.

Estimates of the transient population and their vehicles are presented by polar coordinate representation in Figures 3-4, “Transient Population by Sector,” and 3-5, “Transient Vehicles by Sector,” of the ETE Report.

Employees who commute to jobs within the plume exposure pathway EPZ are assumed to evacuate along with the permanent resident and transient populations. Four major employers, LNP, Crystal River Nuclear Plant (CRNP), Sweetbay Supermarket, and Super Wal-Mart, are within the plume exposure pathway EPZ. In RAI 13.3-6(G)(1), the staff requested clarification on whether the Seven Rivers Regional Medical Center should be considered a major employer since it employs 190 people. In response, the applicant stated that the omission of Seven Rivers Regional Medical Center from the major employers listing was an oversight and was corrected in the table on page E-4 and Figure E-2 of the ETE Report. In RAI 13.3-6(G)(2), the staff requested a discussion regarding the effect on the ETE from the additional vehicle demand due to the employees of the Seven Rivers Regional Medical Center. In response, the applicant stated the ETE Report will assume that 75 percent of the workforce commutes into the EPZ to work at Seven Rivers Regional Medical Center. Based on this assumption and the average vehicle occupancy factor, 138 vehicles will have to be added into the simulation. Various changes were made to the ETE Report based on the addition of Seven Rivers Regional Medical Center as a major employer: a discussion was added of the facility as item “4” on page 3-10; Figures 3-6, 3-7, E-4, E-2, and Table 6-4 were updated based on this information.

Vehicle occupancy of 1.03 is used for the employee population. Estimates of the employee vehicles are presented by polar coordinate representation in Figures 3-7, “Employee Vehicles by Sector.” In RAIs 13.3-6(D), 13.3-6(E), and 13.3-6(F), the staff requested information on whether employees are expected to need transit service; whether LNP employees were considered in the calculations; and whether less than 100 percent of CRNP might be expected to evacuate. In RAI 13.3-6(D), the staff requested a discussion on whether employees and transients have been factored into this need for transit service. In response, the applicant stated since there is no mass transit servicing the area, therefore it is assumed that all transients and employees will have private vehicles available for evacuation. The text on page 8-1 was corrected to reflect this assumption in the revised ETE Report. In RAI 13.3-6(E), the staff requested a discussion on the ETE as to whether LNP employees were included in the calculation. In response, the applicant stated the ETE Report will be updated to include LNP as a major employer when the first unit is complete. The employment data for the CRNP is also misstated in the table on page E-4 and does not agree with the data presented on page 3-10. The table on page E-4 was revised accordingly. Also, the tables in Appendix E were labeled Tables E-1 through E-7. The discussion of construction on page 3-2 and the footnote to Table 6-4 were revised as discussed in response to RAI 13.3-5(D)(4). The input streams to IDYNEV were updated to project to a construction year of 2019 and all ETEs were re-computed. Tables 7-1A through 7-1D; Tables 7-1C and 7-1D in the Executive Summary; Tables J-1A through J-1D; and Figure J-11 were revised to reflect this change as discussed in response to RAI 13.3-05(D)(4). In RAI 13.3-6(F), the staff requested clarification on the actual percentage of CRNP employees that might be expected to evacuate. In response, the applicant included additional text to indicate that it is conservatively assumed in this study that 100 percent of CRNP employees would evacuate.

One special event scenario, Scenario 11, is included in the ETE Report. Scenario 11 represents the peak construction period during a typical winter, weekend, midday, under good weather conditions. Progress Energy estimates there will be two units constructed with Unit 1 being operational in February 2018 and Unit 2 operational in February 2019. Population estimates for permanent residents, transients, and shadow population were extrapolated out to 2019. An estimated 3600 workers and their vehicles were also included in Scenario 12. In RAI 13.3-8(B), the staff requested clarification on why a scenario, such as Scenario 7, was not chosen to be midweek with rain and new plant construction to provide a worst-case estimate. In its response, the applicant stated the specific details of construction scheduling were not determined when the ETE study was conducted. It was uncertain how inclement weather would impact the construction workforce, therefore, Scenario 8 conditions were chosen for the construction scenario, assuming that the full construction workforce would be present under good weather conditions and that this would be a “worst-case” scenario.

Permanent residents, transients, and employees make up the general population. Vehicles traveling through the plume exposure pathway EPZ (external-external trips) are assumed to continue to enter during the first 60 minutes following an accident. Subsequently, none enter the EPZ and those remaining will evacuate with the general population. Population Estimates for special facilities and people without personal vehicles are provided in Section 8, “Transit-Dependent and Special Facility Evacuation Time Estimates.” There are two elementary schools, one Middle School, and two schools/academies with K-12 grades within the plume exposure pathway EPZ. There is one youth correctional facility and 5 daycare facilities located inside the plume exposure pathway EPZ. In RAI 13.3-7(C), the staff requested the applicant clarify whether pre-school children and the youth in the correctional facilities were included in the ETE Study. In response, the applicant stated that it was assumed that children at daycare centers are picked up by their parents and that this activity is accounted for in the mobilization times for residents presented in Section 5. The daycare centers identified on page E-2 of the ETE Report have been added to Tables 8-2, 8-3, 8-5A and 8-5B. The titles of these tables were revised to include daycare centers. Section 8.2 was revised to include discussion of daycare centers. Page 8-8 was revised to include a discussion of the evacuation of the Forestry Youth Camp Incarceration Center to be performed by cooperating law enforcement transporting inmates to a facility outside of the 10-mile EPZ in Tallahassee, Florida.

There are two special care facilities and one regional medical center within 10 miles of the LNP site. In RAI 13.3-7(A), the staff requested the applicant provide the basis for the assumption of loading non-ambulatory individuals in 1.5 minutes. In its response, the applicant revised the text on page 8-8 of the ETE Report to specify a loading time of 30 minutes per ambulance. In RAI 13.3-7(D), the staff requested clarification regarding whether a transit-dependent special needs population exists. If so, discuss whether it was considered in the ETE study. In its response, the applicant stated that recent communication with county emergency management agencies yielded data on a registered special needs population. Section 8.5 entitled, “Evacuation of Homebound Special Needs Population,” was incorporated into the ETE Report. A separate map is provided identifying recreational areas in Appendix E, “Special Facility Data.” In RAI 13.3-7(B), the staff requested a clarification of the LNP plume exposure pathway EPZ lodging table in Appendix E since it appears twice. In its response, the applicant stated that the repeated table was a PDF conversion error and that the table titled, “Levy EPZ: Lodging (As of July 2007),” on page E-2 has been replaced with the table titled, “Table E-1, Levy EPZ Schools (As of July 2007).” All tables in Appendix E were renumbered in the ETE Report.

Telephone survey results (reported in Appendix F, “Telephone Survey”) are used to estimate the portion of the population requiring transit service. The transit-dependent population includes persons in households without vehicles and persons in households whose vehicles are unavailable at the time of evacuation do to commuter use. In RAI 13.3-3(A)(1), the staff requested the actual number of completed survey forms and the sampling error used throughout the telephone survey. In its response, the applicant stated that the total of the required sample column was shown as 550 when it should have been 553. The applicant stated that this was a “rounding-off” error. Table F-1 was revised in the response to RAI 13.3-5(B) and has been incorporated into the ETE Report. The ETE Report now includes additional confidence bound estimates for Figures 5-3, F-1, F-5, F-6, F-7, and F-8; and Table 5-1. In RAI 13.3-3(A)(2), the staff requested clarification on whether completed survey forms received from the public included populations within the associated zip codes, outside of the plume exposure pathway EPZ. In its response, the applicant stated that it is assumed that the demographics are uniform across a zip code. Therefore calls made within the zip codes identified in Table F-1 will produce valid results, even if the person may live just outside the EPZs of the two plants. In RAI 13.3-3(A)(3), the staff requested clarification on what population size was used as a basis for the telephone sampling plan and whether or not the population size used had an effect on the ETEs, if different from the 22,758 population size found on page 3-4. In its response, the applicant stated that due to the close proximity of LNP and CRNP, a combined telephone survey of residents living within the zip codes identified in Table F-1 of the ETE Report was deemed appropriate. The population size used as a basis for the telephone survey sampling plan is 34,880. The computation of this population size is discussed in the response to RAI 13.3-5(B)(1). This population size differs from the EPZ population of 22,758 shown in Table 3-2 of the ETE Report, and this difference is explained in the response to RAI 13.3-5(B)(2).

In RAI 13.3-3(B), the staff requested clarification on the inconsistency in the ETE report regarding the time it takes to evacuate 100 percent of the general population. In its response, the applicant provided a discussion on the process of “truncating” the ETEs used to avoid biasing values. The applicant provided revised text for page 5-11 of the ETE Report as well as a new Appendix M, “Procedure for Estimating Mobilization Time Based upon Survey Data.” In RAI 13.3-5(B)(1), the staff requested clarification on how the population values per zip code were determined for Table F-1. In its response, the applicant provided a discussion of the use of zip code area shape files to obtain population values. The values presented in the second column of Table F-1 of the ETE Report are the Year 2004 population estimates that were mistakenly labeled as Year 2000 population. Table F-1 has been revised to provide Year 2000 population and household data. Table F-1 has been re-titled, “Combined Levy and Crystal River Nuclear Plants Telephone Survey Sampling Plan.” The text on page F-2 has been revised to indicate that a combined survey was performed. In RAI 13.3-5(B)(2), the staff requested the population for each listed zip code in Table F-1. In its response, the applicant provided a discussion on the EPZ population for each listed zip code. The applicant stated, in part, that a combined LNP and CRNP telephone survey was used. The applicant stated that its survey sampling plan, as documented in the new Table F-1, is valid and is being maintained. In supplemental RAIs 13.3-34 and 13.3-36, the staff requested that the applicant include the information provided in response to RAI 13.3-5(B)(2) (Table-2 and text) in the next revision to the ETE Report. In its response, the applicant provided additional clarification regarding the use of a combined telephone survey since the LNP EPZ boundaries had not been finalized at the time the initial ETE Report was developed. The applicant stated, in part, that the EPZ boundaries for LNP have since been defined and the information presented in response to RAI 13.3-5(B)(2) will be incorporated into in a future revision of the ETE Report.

The transit-dependent population is discussed in Section 8.4 of the ETE Report. In RAI 13.3-07(E)(1), the staff requested clarification regarding whether there are enough bus drivers and resources to support a single evacuation wave. In its response, the applicant stated that the ETE Report (Section 8.4) assumes that there are sufficient drivers for all buses available to the EPZ counties. This assumption has been added to Section 2.3 of the ETE Report. The applicant stated that there are sufficient resources of each type available to each county for a single wave evacuation with the exception of buses in Levy County. This issue can be addressed either through a mutual aid agreement with Marion and Citrus Counties, or by using the surplus wheelchair vans within Levy County to evacuate the homebound special needs population. The capacities provided in the discussion of “medical facilities” on page 3-13 of the ETE Report are incorrect and do not reflect the capacities used in this study. This section has been revised to match the capacities provided in Section 8.3. Table 4 illustrates the available and required resources for each county within the LNP EPZ. Table 4 has been added to Section 8 as Table 8-11 of the ETE Report.

In RAI 13.3-7(E)(2), the staff requested clarification regarding the impact on transit services if CRNP had an evacuation at the same time as LNP. In its response, the applicant stated, in part, that there is considerable overlap of the EPZs for the CRNP and the LNP. However, the only PAZ within the CRNP EPZ that is not within the LNP EPZ is PAZ C2 in Citrus County. Therefore, only the resources for Citrus County would be affected by simultaneous evacuation of both EPZs. The applicant stated that by comparing the available resources in Citrus County with the resources needed, a shortage of ambulances and wheelchair vans is identified. The shortage of wheelchair vans can be addressed using the surplus of wheelchair buses in the county. The shortage of ambulances can be resolved by establishing a mutual aid agreement with Marion County, who has excess ambulance resources. As noted in the response to RAI 13.3-11(E)(2), a discussion of a simultaneous evacuation of CRNP and LNP EPZs has been added to Appendix I of the ETE Report. Also, a recommendation has been added to Section 13 of the ETE Report indicating that a mutual aid agreement is needed between Marion and Citrus Counties for ambulance resource support in the rare event that a simultaneous evacuation is advised.

In RAI 13.3-7(F), the staff requested clarification regarding whether the bus travel time estimate takes into consideration the necessary time to traverse traffic control points. In its response, the applicant stated that the inbound bus speed of 45 mph will be unaffected as buses traverse traffic control points. The applicant added the statement, “All transit trips and other responders entering the EPZ to support the evacuation are assumed to be unhindered by personnel manning TCP,” of Section 9 of the ETE Report.

The total number of people expected to evacuate for each scenario and vehicles to be used is discussed in Section 6, “Demand Estimation for Evacuation Scenarios,” of the ETE Report. The LNP plume exposure pathway EPZ contains 8 PAZs with boundaries along major roads or rivers. The boundary definitions are provided in Appendix L, “Protective Action Zone Boundaries,” of the ETE Report. Evacuation will be performed by regions that include multiple PAZs. A description of the evacuation regions and their associated PAZs can be found in Table 6-1,”Description of Evacuation Regions.” A description of the evacuation scenarios used for this study can be found in Table 6-2, “Evacuation Scenario Definitions.” The percentage of population groups expected to evacuate for each scenario is described in Table 6-3, “Percentage of Population Groups for Various Scenarios.” In RAI 13.3-05(C)(1), the staff requested clarification on the values used in Column 2 (Residents with Commuters in Household) of Table 6-3. In its response, the applicant stated that it is conservatively assumed that all households with at least one commuter will await the return of the commuter before beginning their evacuation trip. Assumption 3 in Section 2.3 of the ETE Report has been revised to reflect this information. The data provided on page F-7 (59 percent of households await return of the commuter) was not used in this study. In RAI 13.3-5(C)(2), the staff requested a discussion on how the percentages in Table 6-3 were developed. In its response, the applicant provided an in-depth discussion of the evacuation percentages for each population group as shown in Table 6-3. However the applicant stated that the employment percentages for the weekend scenarios (Scenarios 3, 4, 8, 9 and 11) are overstated at 75 percent. All weekend and evening scenario employee percentages have been changed to 15 percent (conservatively rounded up from the estimated 12.5 percent). Tables 6-3 and 6-4 have been revised accordingly. This change was incorporated into the IDYNEV input stream and all simulations were re-run. The ETE has been recomputed. Tables 7-1A through 7-1D (7-1C and 7-1D also appear in the Executive Summary); Tables J-1A through J-1D; Figures 7-3 through 7-7; and Figures J-1 through J-11 were also updated accordingly.

Technical Evaluation: [Section II of Appendix 4]
The staff finds the additional information, clarifications, and textual revisions provided in response to RAIs 13.3-3(A)(1)-(A)(3), 13.3-3(B), 13.3-5(A), 13.3-5 (B)(1), 13.3-5(B)(2), 13.3-5(C)(1), 13.3-5(C)(2), 13.3-5(D)(1)-(D)(4), 13.3-6(A), 13.3-6(B), 13.3-6(C), 13.3-6(D), 13.3-6(E), 13.3-6(F), 13.3-6(G)(1), 13.3-6(G)(2), 13.3-7(A), 13.3-7(B), 13.3-7(C), 13.3-7(D), 13.3-7(E)(1), 13.3-7(E)(2), 13.3-7(F), 13.3-8(A), 13.3-8(B), supplemental RAIs 13.3-34 and 13.3-36 to be acceptable because they meet the requirements of Appendix E.IV to 10 CFR Part 50 and conforms to the guidance in Appendix 4 to NUREG-0654/FEMA-REP-1. The staff confirmed that the changes proposed in the above RAIs have been incorporated into Revision 4 of the LNP ETE Report, with one exception. The staff created Confirmatory Item 13.3-36 to track the inclusion of the table and textual revisions provided in response to supplemental RAI 13.3-36 into the next revision of the LNP ETE Report.

Resolution of Confirmatory Item 13.3-36

Confirmatory Item 13.3-36 is an applicant commitment to update the LNP ETE Report. The staff verified that the LNP ETE Report was appropriately updated (or revised). As a result, Confirmatory Item 13.3-36 is now closed.

The staff finds that the ETE Report provides an estimate of the number of people who may need to be evacuated. This is acceptable because it conforms to the guidance in Section II of Appendix 4 to NUREG-0654/FEMA-REP-1. Three population segments are considered: permanent residents, transients, and persons in special facilities. The permanent population is adjusted for growth, and the population data is translated into two groups: those using automobiles and those without automobiles. The number of vehicles used by permanent residents is estimated using an appropriate automobile occupancy factor. In addition, ETEs for evacuation of the entire plume exposure pathway EPZ were determined. Estimates of transient populations were developed using local data, including peak tourist volumes and employment data. Estimates for special facility populations are also provided. The subareas, for which ETEs were determined, encompass the entire area within the plume exposure pathway EPZ. The maps are generally adequate for the purpose, and the level of detail is approximately the same as United States Geological Survey (USGS) quadrant maps. The assumptions on evacuation are based on simultaneous evacuation of inner and outer sectors.

13.3C.18.4 *Traffic Capacity*

Technical Information in the ETE Report: [Section III of Appendix 4]
Section 4, “Estimation of Highway Capacity,” describes the process used to determine vehicle capacities for roadways in the transportation network. The methods used are generally taken from the HCM published by the Transportation Research Board of the National Research Council. Appendix K, “Evacuation Roadway Network Characteristics,” identifies all evacuation route segments and their characteristics, including capacity. A map of the transportation network is provided in Figure 1-2, “Levy Nuclear Plant Link-Node Analysis Network.” Additional information describing the road network used for evacuation routes was requested in RAI 13.3-10(A). In its response, the applicant stated that Figures 10-2 and 10-3 in the ETE

Report have been updated to include highway numbers and road names for the major evacuation routes. Figures 10-1, 10-2 and 10-3 were also modified to include the names of the reception centers. Section 10 of the ETE Report was reviewed for consistency. Table 10-1 has been revised to only show one entry for Bronson High School and indicate that it is a primary shelter and a daycare shelter. Also, the text of Section 10 has been revised to indicate that it is assumed the shelters/reception centers to be used for the LNP EPZ are the same as those identified for the CRNP.

In RAIs 13.3-11(A)(1) and 13.3-11(A)(2), the staff requested information related to lane width. In its response, the applicant stated that in Appendix K, the term "full lanes" is used to identify the number of lanes that extend over the entire length of the roadway segment or link; it does not pertain to lane width. A discussion regarding the use of geometric features in modeling was also provided. Additional text has been added to Section 1.3, and Appendix K to further describe the road survey and to clarify what is meant by "Full Lanes." In RAIs 13.3-11(B)(1) to (B)(3), the staff requested information related to unusual road characteristics. In its response, the applicant stated that the number of bridges, sharp curves, narrow shoulders and other capacity-reducing features on the evacuation network were observed and considered in estimating capacity. The capacity drops to 1714 vehicles per hour per lane across the U.S. Route 19 Bridge crossing the Cross Florida Barge Canal and the number of lanes decreases to 1 as shown for link (117, 63) in Appendix K. The properties of all links representing bridges are recorded in Appendix K (with all other links), but are not otherwise delineated. For further clarification see response to RAI 13.3-11(A).

In RAIs 13.3-11(C)(1) to (C)(3), the staff requested information related to ideal conditions and roadway capacity. In its response, the applicant stated that the capacity and free flow speed data input to IDYNEV and documented in Appendix K are based upon observations made during the road survey. Where the base conditions are not realized, downward adjustments to the capacity estimate of 1700 pc/hr were made. The link capacities presented in Appendix K are accurate; therefore the ETE are unaffected.

In RAI 13.3-3(G), the staff requested the applicant explain the significance of the identified roadway unusual characteristics, including how they impact the proposed LNP site. In its response, the applicant stated that the responses to RAIs 13.3-11(A), (B), and (C), include a detailed discussion of the road survey. In addition, a large-scale (4 ft by 3 ft) version of Figure 1-2 is provided with node numbers annotated so that links can be cross referenced with Appendix K information. In supplemental RAI 13.3-33, the staff requested that the applicant clarify in the ETE analysis whether any physical characteristics unique to the proposed LNP site exist, which could pose a significant impediment to the development of the LNP Emergency Plan. In its response, the applicant stated that conversations held between KLD, Progress Energy, Emergency Management personnel from the State of Florida and the counties of Citrus, Levy and Marion, revealed "…no physical characteristics unique to the proposed LNP site that could pose a significant impediment to protecting the public under normal conditions at the time the ETE Report was conducted."

In RAIs 13.3-11(D)(1) and 13.3-11(D)(2), the staff requested additional information, in part, on whether the 0.85 reduction factor was applied to all roadways, including freeways. In its response, the applicant provided a reference to the origin of the reduction factor and a description for how it was applied within the 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. Appendix G, "Traffic Management," provides a description of TCPs and ACPs and provides maps of their location within the plume exposure pathway EPZ.

Section 10, "Evacuation Routes," illustrates the emergency evacuation routes for the three counties surrounding the LNP site. Evacuation routes provide for evacuation first to the plume exposure pathway EPZ boundary and then to reception centers. The TRAD model was used to determine routes that would minimize exposure to risk by balancing traffic demand relative to road capacity. Evacuation routes were also developed to minimize travel outside the plume exposure pathway EPZ and relate traffic volume to reception center capacity. Section 7.2, "Patterns of Traffic Congestion During Evacuation," identifies areas of traffic congestion that arise for the case when the entire plume exposure pathway EPZ (Region R3) is advised to evacuate during the summer, weekend, and midday period under good weather conditions. This is illustrated in Figures 7-3, "Congestion Patterns at 1 hour After the Advisory to Evacuate (Scenario 8)," 7-4, "Congestion Patterns at 1 hour, 30 minutes after the Advisory to Evacuate (Scenario 8)," 7-5, "Congestion Patterns at 2 hours After the Advisory to Evacuate (Scenario 8)," and Figure 7-6, "Congestion Patterns at 2 hours, 30 minutes After the Advisory to Evacuate (Scenario 8)." Additional information regarding travel times and delay durations during evacuation was requested in RAIs 13.3-14(A) to (F) and 13.3-15(A)-(F).

In RAI 13.3-14(A), the staff requested that a map be provided which identifies where these zonal centroids were located in the model. In its response, the applicant provided a larger scale version of Figure 1. In RAIs 13.3-14(B)(1) and 13.3-14(B)(2), the staff requested clarification on how traffic control affects the modeling parameters and any assumptions on traffic speed, service flow, capacity, and queue discharge through a staffed intersection. In its response, the applicant stated that the traffic control points are modeled as traffic signals with a reasonable allocation of effective green time to each of the competing traffic streams. In RAI 13.3-14(B)(3), the staff requested clarification on the impact on traffic timing and traffic loading if CRNP had an evacuation at the same time as LNP. The applicant stated this information is provided in the response to RAI 13.3-13(F)(3). In RAI 13.3-14(C), the staff requested clarification on whether the evacuation activity, "Depart Place of Work," (Step 3) should also be included in the last row of the event sequence in the table on page 5-3 of the ETE Report. In its response, the applicant provided a discussion of evacuation activities. Based on this discussion the applicant concluded that there is not a need to add Step 3 to the evacuation sequence, "Prepare to leave for evacuation trip." As stated in the second paragraph on page 5-4, event number 5 depends on the time distributions of all activities preceding that event. The table on page 5-3 is intended to provide the definition of each individual activity; for simplicity, all preceding dependent events (excluding event 2) have not been included in this table. Figure 5-1 and the text in Section 5 have been revised as discussed in response to RAI 13.3-6(C). In RAI 13.3-14(D), the staff requested trip generation time elements for the transient population. In its response, the applicant stated that as shown in Table 5-1 of the ETE Report, transient mobilization time (Distribution A) extends over a period of 2 hours, with 78 percent of transients mobilizing in the first hour and the remaining 22 percent in the last hour. The applicant stated that it is reasonable to expect that 2 hours will be sufficient time for those who are boating or diving in the area to return to the shore and begin their evacuation trip. Additional information related to notification of boaters and divers was provided in the response to RAI 13.3-6(B). In RAI 13.3-14(E), the staff requested the basis for the statement that 85 percent of the population within the plume exposure pathway EPZ will become aware of the accident within 30 minutes. In its response, the applicant stated that the notification distribution is assumed based on the presence of the siren alert system. This assumption has been added to Section 2 of the ETE Report and the discussion on notification of the public on page 5-4 has also been revised accordingly.

In RAI 13.3-15(A)(1), the staff requested discussion on how the 100 minute value was derived when Appendix F, "Telephone Survey," states on page F-8 that this activity is completed in approximately 120 minutes and shows a curve extending to 150 minutes. In RAI 13.3-15(A)(2), the staff requested clarification on how the 120 minutes was derived when Figure F-10, “Work to Home Travel,” indicates that less than 100 percent have traveled home in 120 minutes, and the curve for this figure projects to 150 minutes. In its responses to these RAIs, the applicant stated that the distribution was "truncated" to 100 minutes on page F-8 and to 120 minutes on Page F-9 to avoid the bias of stragglers. “Truncating” the distributions advances the mobilization of the stragglers. Therefore, the stragglers are not eliminated from the ETE. See the response to RAI 13.3-3(B) for additional detail on the truncation procedure.

In RAI 13.3-15(A)(3)(a), the staff requested a discussion on the difference in data between Appendix F and Section 5. In its response, the applicant stated that the response to RAI 13.3-3(B) discusses that Appendix F presents the raw telephone survey data. Section 5 of the ETE Report presents the trip generation for the EPZ population, which includes some truncation of the distributions presented in Appendix F. A new Appendix M has been added to the ETE report which describes this truncation procedure as stated in the response to RAI 13.3-3(B). In RAI 13.3-15(A)(3)(b), the staff requested a clarification of the statement under Distribution #4 (Page 5-8), “These data are provided directly from the survey.” In its response, the applicant stated that as noted in the response to RAI 13.3-15(A)(3)(a) the distributions provided in Section 5 of the ETE Report are truncated from the raw distributions presented in Appendix F. The statement on pages 5-7 and 5-8 has been revised accordingly. In RAI 13.3-15(A)(4), the staff requested a reconciliation of Figure 5-2, "Evacuation Mobilization Activities," and Figure 5-3, "Comparison of Trip Generation Distributions," with the comments on use of telephone survey data. In its response, the applicant stated that no changes are needed to Figures 5-2 and 5-3. Appendix M has been added and is referenced in Section 5 of the ETE Report to explain the differences between the raw distributions presented in Appendix F and the final distributions presented in Section 5.

In RAI 13.3-15(B), the staff requested clarification as to why Figure 7-7, "Evacuation Time Estimates Winter, Weekend, Midday, Good Weather (Scenario 8)," was not projected to include 100 percent of the population. In its response, the applicant stated that the ETE is defined as the elapsed time after the advisory to evacuate (ATE) when the last person exits the EPZ. Based on this definition, Figure 7-7, which plots evacuating vehicles versus elapsed time after the ATE, ends at the 100th percentile when the last vehicle has exited the EPZ. Figures J-1 through J-11 are presented in the same fashion; the endpoint of each curve is the 100th percentile ETE.

In RAI 13.3-15(C), the staff requested clarification on how a value of 45 percent was derived in Table 8-1. In its response, the applicant stated that Figure F-6 indicates that 55 percent of the households surveyed have 0 commuters. Therefore, 45 percent of households have at least 1 commuter. In RAI 13.3-15(D), the staff requested the queuing locations and estimated delay times on the maps in Figures 7-3, “Congestion Patterns at 1 hour after the Order to Evacuate (Scenario 8),” through Figure 7-6, “Congestion Patterns at 2 hours 30 minutes after the Order to Evacuate (Scenario 8).” In its response, the applicant stated that Figures 7-3 through 7-6 have been revised to include the major roads and to identify congestion points. Table 7-3 provides a description of each congestion point and the link from Figure 1-2 corresponding to that area of congestion. In RAI 13.3-15(E), the staff requested clarification on how a 50 percent increase in demand for buses given in Section 8-1 of the ETE Report could still be accommodated if buses are assumed to be at 68 percent capacity. In its response, the applicant stated that a 50 percent increase in demand is equivalent to applying a factor of 1.5 to the estimated demand. An equation has been added before the final paragraph on page 8-2 of the ETE Report that demonstrates how this factor is used.

Technical Evaluation: [Section III of Appendix 4]
The staff finds the additional information and proposed textual revisions submitted in response to RAIs 13.3-10(A), 13.3-11(B)(1)-(B)(3), 13.3-11(A)(1), 13.3-11(A)(2), 13.3-11(C)(1)-(C)(3), 13.3-11(D)(1,), 13.3-11(D)(2), 13.3-14(A), 13.3-14(E), 13.3-14(B)(1)-(B)(3), 13.3-14(C), 13.3-14(D), 13.3-15(A)(1), 13.3-15(A)(2), 13.3-15(A)(3a), 13.3-15(A)(3b), 13.3-15(A)(4), 13.3-15(B), 13.3-15(C), 13.3-15(D), 13.3-15(E), and supplemental RAI 13.3-33 to be acceptable because they meet the requirements of Appendix E.IV to 10 CFR Part 50 and conform to the guidance in Appendix 4 to NUREG-0654/FEMA-REP-1. The staff confirmed that the changes proposed in the above RAIs have been incorporated into Revision 4 of the LNP ETE Report. The staff finds that the LNP ETE Report provides a complete review of the evacuation road network. Analyses are made of travel times and potential locations for congestion. The ETEs are not dependent on the establishment of traffic control points and access control points. Therefore, manpower and equipment shortages have no effect on the ETE calculations. In addition, 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. Therefore, the information provided in the LNP ETE Report with regard to traffic capacity meets the requirements of Appendix E, Section IV to 10 CFR Part 50 and conforms to the guidance in Appendix 4 to NUREG-0654/FEMA-REP-1.

13.3C.18.5 *Analysis of Evacuation Times*

Technical Information in the ETE Report: [Section IV of Appendix 4]
Sections 4, 5, and 6 of the ETE Report describe the methods used to estimate the evacuation times. Section 4, "Estimation of Highway Capacity," describes how data collected during field surveys of the transportation network were combined with methods suggested in the 2000 HCM. Section 5, "Estimation of Trip Generation Time," provides estimates of the four different distributions of elapsed times associated with mobilization activities undertaken by the public to prepare for the evacuation trip. The elapsed time associated with each activity is represented as a statistical distribution reflecting differences between members of the public. In RAI 13.3-7(I), the staff requested clarification whether stopping and dwell time were considered in the estimation of the average route time proposed for transit services. In its response, the applicant stated that stopping and dwell time was considered as the "pickup time." This discussion of pickup time for transit-dependent persons provided in the response has been added to the discussion of, "Activity: Board Passengers (C--D)," on page 8-5 of the ETE Report. Also, the IDYNEV model has recently been improved to include a bus route feature. The applicant provided revisions to the ETE Report based on this new feature as discussed in RAI 13.3-8(C). In RAI 13.3-07(J), the staff requested information on the "experience" used to establish the mobilization time of 90 minutes for buses. In its response, the applicant stated that the mobilization time for transit vehicles is based on discussions with local emergency management personnel at this site and several others, and was approved by the counties as indicated by the signed certification letters submitted with the COL. In RAI 13.3-7(L), the staff requested information on why the relocation center locations are not identified on the map in Figure 8-2 of the ETE Report. In its response, the applicant stated that Figure E-1 has been revised to include the locations of the relocation schools. The following footnote has been added to Table 8-3, "Figure E-1 in Appendix E identifies the location of all EPZ schools and the relocation schools they are evacuated to."

Additional information was requested in RAI 13.3-11(E) regarding the average roadway speeds at various times of the evacuation and whether these speed values would change if CRNP had an evacuation at the same time as LNP. In RAI 13.3-11(E)(1), the staff requested an explanation of how, in Section 8.4 of the ETE Report, the average speed can exceed 50 miles per hour (mph) when more than 70 percent of the roadway segments in Appendix K have free flow speeds between 30 and 50 mph. In its response, the applicant provided a discussion of free flow speeds which is in good agreement with the speeds presented on pages 8-6 through 8-8 of the ETE Report. The applicant also added that the IDYNEV model has recently been improved to include a bus route feature which will provide more accurate route-specific speeds than using the average network-wide speed output by IDYNEV. The ETE has been updated as discussed in response to RAI 13.3-8(C). In RAI 13.3-11(E)(2), the staff requested a discussion on the impact of the average evacuation travel speeds if an evacuation occurred at the same time at CRNP. In its response, the applicant stated that a sensitivity study was conducted to measure the effects of a simultaneous evacuation of the EPZs for both the LNP and the CRNP during Scenario 6 conditions. The combined EPZ differs from the LNP EPZ with the addition of PAZ C2 within the CRNP EPZ, as shown in Figure 1. The increased congestion in the combined EPZ results in lower average speeds. A discussion of the simultaneous evacuation of the CRNP and LNP EPZs has been added as a sensitivity study in Appendix I as discussed in response to RAI 13.3-7(E)(2).

Section 6, “Demand Estimation for Evacuation Scenarios,” defines the various evacuation cases for which time estimates were made; a case is a combination of a scenario and a region. A scenario is 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 as a grouping of contiguous evacuation PAZs, which forms either a "keyhole" sector-based area, or a circular area within the plume exposure pathway EPZ, that must be evacuated in response to a radiological emergency. Reception centers are shown on maps in Section 10, “Evacuation Routes.” The assumptions on evacuation are based on simultaneous evacuation of inner and outer sectors.

A summary of the ETE is provided in Section 7, “General Population Evacuation Time Estimates (ETE).” These results cover 13 regions within the LNP plume exposure pathway EPZ and the 11 evacuation scenarios discussed in Section 6. The evacuation times are presented for 13 evacuation regions and 11 scenarios in Appendix J, “Evacuation Time Estimates for All Evacuation Regions and Scenarios and Evacuation Time Graphs for Region R03, for all Scenarios.” Results are presented for 50 percent, 90 percent, 95 percent, and 100 percent of vehicles. In RAI 13.3-13(A), the staff requested a discussion of any assumptions related to how rail traffic may affect the ETE. In its response, the applicant stated there is no commuter rail or Amtrak service in the area. There is a rail line running to the Crystal River Energy Complex, which is primarily used for coal. Trains can be stopped from entering the EPZ in the event of an incident at either the CRNP or LNP. In RAIs 13.3-13(B) and 13.3-13(F)(2), the staff requested the assumptions with regard to shadow evacuation trip generation times and loading of the transportation network. In its response, the applicant stated that shadow vehicles shown in Table 6-4 are loaded on the link-node analysis network (Figure 1-2) using the same trip generation times as EPZ residents with Commuters - Distribution C in Table 5-1. This statement has been added to Section 7.1 for clarification. In RAI 13.3-13(C), the staff requested a clarification regarding how the evacuation time of 5 hours 10 minutes for R03 for Scenario 11 which has 41,898 vehicles in Table 7-1D, can be the same for all other scenarios, some of which can have as few as 23,834 vehicles. In its response, the applicant provided a detailed discussion on how the ETE for the 100th percentile of the evacuating population mimics the trip generation time and lesser percentiles that may be affected by congestions, such as the case with Scenario 11. In RAI 13.3-13(D), the staff requested a discussion on why the time to clear 100 percent of the indicated area for the 5-mile ring, is the same as the time listed for the entire plume exposure pathway EPZ. In its response, the applicant stated that as indicated in the response to RAI 13.3-8(C), PAZ C1 and C3 were mistakenly included in the 5-mile region. As shown in Figure 3-1, PAZ C1 and C3 extend all the way to the EPZ boundary. Therefore, the distance traveled to exit the 10-mile region is similar to that of exiting the 5-mile region. PAZ C1 and C3 have been removed from the 5-mile region and all ETE simulations have been re-run. Tables 7-1A through 7-1D and Tables J-1A through J-1D have been updated as discussed in the response to RAI 13.3-8(C).

In RAI 13.3-13(E), the staff requested a discussion on the note for Distribution No. 2 and No. 3 in Section 5, including the process used to normalize the data. In its response, the applicant stated that to address the occasional “don't know” responses from a large sample, the “don't know” responses are essentially ignored and the distributions are based upon the positive data that is acquired. In RAI 13.3-13(F)(1), the staff requested an explanation of which value in Section 6 is being used for shadow resident vehicles. In its response, the applicant provided a discussion on the calculation of shadow vehicles based on a ratio of employee vehicles to resident vehicles. In RAI 13.3-13(F)(3), the staff requested a clarification of the impact on traffic timing and traffic loading if CRNP had an evacuation at the same time as LNP. In its response, applicant provided a discussion related to simultaneous evacuation more specifically for residents residing in Yankeetown and Inglis. Suggested traffic control points were provided in Figure G-2 of the ETE Report.

Results are provided for good and adverse conditions. In RAI 13.3-3(D), the staff requested clarification regarding why there is an effect to mobilization time for schools and special facilities, but not for the general public. In its response, the applicant stated that the “No Effect” identified in the table on page 2-5 refers to the mobilization time for the general population. The applicant stated that the only portion of this mobilization that involves driving is the time to return home. The mobilization times discussed in Section 8 are for that portion of the population which is dependent on transit resources - schoolchildren, special facility populations and those people who do not have access to a private vehicle. The majority of this mobilization time for the bus driver is spent driving; as a result, the reductions of 10 percent in capacity and in speed for rain are assumed to add a total of 10 minutes to the mobilization time.

The methodology for the general population uses distribution functions. Figures describing the time distribution of evacuating vehicles follow the format of NUREG-0654, Appendix 4, Figure 4. In RAI 13.3-12(A), the staff requested an explanation of why only Region 03 is affected by rain when evacuating 90 percent, 95 percent, or 100 percent of the population. In its response, the applicant stated that the presence of rain reduces capacity and free speed on all network links by 10 percent (page 2-5). When evacuating the entire EPZ (Region 03), this reduction in speed and capacity led to a modest increase (10 minutes or less) in ETE at both the 90^{th} percentile and 95^{th} percentile level of evacuation (compare Scenarios 8 and 9 in Table 7-1 B and Scenarios 1 and 2 in Table 7-1C). As shown in Figure 7-5, all congestion within the EPZ has dissipated by 2 hours after the ATE. Rain does not affect the ETE for the 100^{th} percentile population because capacity is no longer a factor after 2 hours following the ATE. A change in ETE of 10 minutes would not likely change the protective action decision making process.

Section 8, “Transit-Dependent and Special Facility Evacuation Time Estimates,” discusses evacuation plans for schools, residents without vehicles, and special care facilities. These groups are expected to merge with general evacuation traffic following notification and mobilization. In RAI 13.3-7(G), the staff requested an explanation on how transit dependent individuals are expected to get from their residences to the bus routes, and whether this time was factored into the ETE. In its response, the applicant stated that evacuees without access to private transportation are expected to walk to the bus routes. Those who are unable to walk to the route should register with the county as special needs persons. See the response to RAI 13.3-7(D) for discussion of this group.

Separate estimates of population size and necessary transportation were made for schools, special facilities and the transit-dependent populations. Schools are given advanced notification, if possible, in order to determine transportation needs. The estimated students and their transportation needs, based on student to bus ratios, are provided in Table 8-2, "School Population Demand Estimates." In RAI 13.3-3(E), the staff requested additional detail regarding the assumptions used to support boarding 1100 students in five minutes. In its response, the applicant provided two satellite pictures above the Dunnellon Middle School in support of the student loading time of 5 minutes. In RAI 13.3-7(K), the staff requested a discussion on the assumptions related to the estimated time to load buses for evacuation. In response, the applicant cited information provided in response to RAI 13.3-3(E). In RAI 13.3-8(D), the staff requested clarification on the apparent inconsistency of whether school is in session for Tables 6-3 and 6-4, and a discussion on whether school bus usage accounts for summer school. In response, the applicant stated for Scenarios 1 and 2, the buses are evacuating summer school students. It is assumed that summer school enrollment is approximately 10 percent of enrollment for the regular school year. This assumption has been added to Section 2 of the ETE Report and to the "School and Transit Buses," footnote to Table 6-3. The references to, "school not in session," for the summer season in Section 7.4 and Section J.A has been removed to avoid confusion.

Transportation resources should be adequate to evacuate schools in one wave, but additional resources can be requested from nearby cities if necessary. Mobilization of drivers and students was factored into the total evacuation times. The estimated time to evacuate schools within the plume exposure pathway EPZ is provided in Table 8-5A, "School Evacuation Time Estimates-Good Weather," and Table 8-5B, "School Evacuation Time Estimates-Rain," of the ETE Report. Evacuation of other special facilities is given the same consideration as schools with the exception of increased loading time. The estimated population and necessary transportation resources can be found in Table 8-4, "Special Facility Transit Demand."

Remaining transportation resources and those that become available following the evacuation of schools will be used to evacuate the portion of the population without vehicles. The estimated time to evacuate transit-dependent people within the plume exposure pathway EPZ is provided in Table 8-6A, "Transit Dependent Evacuation Time Estimates-Good Weather," and Table 8-5B, "Transit Dependent Evacuation Time Estimates-Rain." In RAIs 13.3-7(H)(1) and 13.3-7(H)(2), the staff requested the applicant clarify whether buses will make random stops or if the stops are predetermined. In addition, if the stops are predetermined, provide maps that show the bus stop locations, including a discussion on the effect to ETE calculations. In its response, the applicant stated that it is assumed that transit-dependent persons will walk to the nearest route and "flag" down a bus traversing the route. Thus, there are no pre-established pickup points for transit-dependent persons. This assumption has been added to the discussion of, "Activity: Board Passengers (C→D)," on page 8-5 of the ETE Report.

A series of sensitivity tests are documented in Appendix I, "Evacuation Sensitivity Studies," regarding the sensitivity of the results to trip generation time (directly related to time-dependent traffic loading) and to the amount of shadow evacuation.

Technical Evaluation: [Section IV of Appendix 4]
The staff finds the additional information and proposed textual revisions submitted in response to RAIs 13.3-3(D)(E), 13.3-7(E)(2), 13.3-7(G), 13.3-7(J), 13.3-7(K), 13.3-8(D), 13.3-11(E)(1), 13.3-11(E)(2), 13.3-12(A), 13.3-13(A), 13.3-13(B), 13.3-13(C), 13.3-13(D), 13.3-13(E), 13.3-13(F)(1)-(F)(3), 13.3-13(H)(1), 13.3-13(H)(2), 13.3-13(I), and 13.3-13(L) to be acceptable because they meet the requirements of Appendix E.IV to 10 CFR Part 50 and conforms to the guidance in Appendix 4 to NUREG-0654/FEMA-REP-1. The staff confirmed that the changes proposed in the RAIs above have been incorporated into Revision 4 of the LNP ETE Report. The staff finds that the LNP ETE Report provides a total of 252 ETEs computed for the evacuation of the general public. Each ETE quantifies the aggregate evacuation time estimated for the population within one of the 21 evacuation regions to completely evacuate from that region, under the circumstances defined for one of 12 evacuation scenarios (13 x 11 = 143). Separate ETEs are calculated for transit-dependent evacuees, including school children.

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.

There are separate distributions for auto-owning households, school population, and transit-dependent populations.

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 upon public transportation is developed, in a manner similar to that used for the auto-owning population. Therefore, the information provided in the LNP ETE Report with regard to evacuation times meets the requirements of Appendix E, Section IV to 10 CFR Part 50 and conforms to the guidance in Appendix 4 to NUREG-0654/FEMA-REP-1.

13.3C.18.6 *Other Requirements*

Technical Information in the ETE Report: [Section V of Appendix 4]
Section 12, "Confirmation Time," of the ETE Report suggests a possible alternative procedure to confirm that the evacuation process is effective in the sense that the public is complying with the Advisory to Evacuate. The suggested procedure employs a stratified random sample and a telephone survey. Based on calculations, it would be necessary to make 300 random phone calls to confirm that 20 percent of the population has not yet evacuated. This process could be

completed within 75 minutes if five people are assigned to the task. Since confirmation begins three hours after the advisory, confirmation should be made when the evacuation area is clear. If more than 20 percent of the population is determined to have not yet evacuated, the telephone survey will be repeated after an hour interval until evacuation is complete.

The development of the ETE Report was coordinated with emergency planners from the State of Florida and Levy, Marion, and Citrus County who are involved in emergency response for the site. In RAI 13.3-16, information was requested regarding the review of the ETE Report by State and local organizations involved with emergency response and whether their comments had been included in the ETE Report. In its response, the applicant provided a description of the approval process. In addition, it was stated that the signed certification letters from each EPZ county and from the State of Florida, included in the COL, verify that the offsite agencies approved the ETE document, including the traffic management plan as provided in Section 9 and Appendix G, and the telephone survey instrument as provided in Appendix F of the ETE Report.

Technical Evaluation: [Section V of Appendix 4]
The staff finds the additional information provided in response to RAI 13.3-16 to be acceptable because it meets the requirements of Appendix E, Section IV to 10 CFR Part 50 and the guidance in Appendix 4 to NUREG-0654/FEMA-REP-1. In addition, the development of the ETE Report was coordinated with emergency planners from the State of Florida and Levy, Marion, and Citrus County who are involved in emergency response for the site. The staff finds that the LNP 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 Section V of Appendix 4 to NUREG-0654/FEMA-REP-1.

13.3C.18.7 *Conclusion*

On the basis of its review of Revision 4 of the LNP ETE Report as described above, the staff concludes that the information provided in the ETE Report is consistent with those portions of Section 13.3 of NUREG-0800 related to the evacuation time estimate analysis and is consistent with the guidance in Appendix 4 to NUREG-0654/FEMA-REP-1. Therefore, the ETE Report is acceptable and meets the applicable requirements of 10 CFR Part 50, Appendix E.IV.

13.3C.19 Inspection, Test, Analysis, and Acceptance Criteria (EP ITAAC)

13.3C.19.1 *Regulatory Basis*

The staff considered the following regulatory requirement and guidance in the evaluation of the information in the COL application related to EP ITAAC:

10 CFR 52.80(a) requires that a COL application include the proposed inspections, tests, and analyses, including those applicable to EP, 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 be operated in conformity with the COL, the provisions of the Atomic Energy Act, and the Commission's rules and regulations.

Table 14.3.10-1, "Emergency Planning Generic Inspection, Tests, Analyses, and Acceptance Criteria," of NUREG-0800.

13.3C.19.2 *Proposed EP ITAAC*

Technical Information Related to the Emergency Plan: (52.80(a))
The applicant addresses EP ITAAC in Table 3.8-1, "Emergency Plan Inspections, Tests, Analyses, and Acceptance Criteria," of Part 10 of the COL application. The LNP COL application also incorporates by reference Tier 1 Table 3.1-1, "Inspections, Tests, Analyses, and Acceptance Criteria," from the AP1000 DCD. The results of the staff's evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements. As noted in Section 13.3.4 of this SER, the staff will include the following license condition for LNP Units 1 and 2:

The licensee shall perform and satisfy the ITAAC defined in SER Table 13.3-1, "Emergency Plan ITAAC."

SER Table 13.3-1 consists of the EP ITAAC identified in Table 3.8-1 of Appendix B to Part 10 of the LNP COL application, as modified by letters dated June 3, 2009, December 18, 2009, March 26, 2010, and March 15, 2011, and January 10, 2014.

In its review of Table 3.8-1 of Appendix B to Part 10 of the COL application, the staff used as review guidance, the generic EP ITAAC in Table 14.3.10-1 to Section 14.3.10 of NUREG-0800. Table 14.3.10-1 identifies a generic set of acceptable EP ITAAC. Since these EP ITAAC were established on a generic basis, they are not associated with any particular site or design. As such, several of the generic EP ITAAC requires the COL applicant to provide more specific acceptance criteria that reflect the plant-specific design and site–specific emergency response plans and facilities.

Based on this comparison, the staff requested additional information in RAIs 14.3.10-1(A) to 14.3.10-1(O), supplemental RAIs 13.3-32(1) to 13.3-32(6), and supplemental RAIs 13.3-44(1) to 13.3-44(5), to address inconsistencies identified between EP ITAAC Table 3.8-1 of the COL application and Table 14.3.10-1 in NUREG-0800. The staff reviewed the applicant's RAI responses, proposed revisions to Table 3.8-1, including Revisions 1 and 2 to Part 10 of the COL application, and found that the applicant adequately addressed most of the inconsistencies. The staff identified the following inconsistencies and issued supplemental RAIs 13.3-49(4)(b), 13.3-51, and 13.3-58 as described below:

- In supplemental RAI 13.3-49(4)(b), the staff requested that the applicant describe the capability of the TSC and EOF equipment and data displays to clearly identify and reflect the affected unit during a declared emergency, or propose EP ITAAC to demonstrate this capability. In response, the applicant stated, in part, that the TSC and EOF equipment and data displays will clearly identify this capability and proposed EP ITAAC 12.1.1.D.2.d to validate this capability exists.

- In RAI 13.3-51, the staff requested, in part, that the applicant to discuss why EP ITAAC 8.4 does not include reference to the EOF as being able to display meteorological parameters consistent with the description of meteorological capabilities provided in Section I.5 of the emergency plan. In response, the applicant proposed to revise EP ITAAC 8.4 to align with Section I.5 of the emergency plan incorporating this capability in the EOF.

- In RAI 13.3-58, the staff requested that the applicant revise EP ITAAC 12.1.1.E.3 and 8.1.B.3 to align with ERO staffing augmentation times as identified in Table B-1 of the emergency plan. In response, the applicant proposed the following revisions, in part, to EP ITAAC:

    - EP ITAAC 8.1.B.3: Demonstrate the ability to activate one radiological monitoring team consisting of two personnel within 30 minutes of event declaration, and a second radiological monitoring team consisting of two personnel within 60 minutes of event declaration.

    - EP ITAAC 12.1.1.E.3.a: One radiological monitoring team consisting of two personnel is ready to be deployed no later than 30 minutes from the declaration of an alert or higher emergency, and a second radiological monitoring team two personnel is ready to be deployed no later than 60 minutes from the declaration of an alert or higher emergency.

    - EP ITAAC 5.1, 7.5, 8.6, 8.7, 8.8, 8.9, 12.1.1.E.4, and 12.1.1.E.4.b will be revised to change its reference from field monitoring teams to radiological monitoring teams consistent with proposed changes to Table B-1 and Section I.4.1 regarding on-site dose assessment in the emergency plan.

    - EP ITAAC 12.1.1.C.1.a will be revised to reflect a demonstration of command and control capabilities by the EC and EOF Director in the TSC and EOF within 60 minutes of ERO notification.

In a letter dated March 15, 2011, the applicant proposed the following revision to EP ITAAC 12.1.3:

- The exercise was completed within the specified time periods of Appendix E to 10 CFR Part 50, offsite objectives were met, and there were no uncorrected offsite deficiencies, or a license condition requires offsite deficiencies to be corrected prior to operation above 5% of rated power as described in 10 CFR 50.54(gg).

The staff created Confirmatory Item 13.3-61 to track the applicant’s revision to Table 3.8-1 in Part 10 of the COL application.

<u>Resolution of Confirmatory Item 13.3-61</u>

Confirmatory Item 13.3-61 is an applicant commitment to update EP ITAAC in Part 10 of the COL application. The staff verified that Part 10 of the COL application was appropriately updated (or revised). As a result, Confirmatory Item 13.3-61 is now closed.

Technical Evaluation: (52.80(a))
The responses to the RAIs referenced above conform to NUREG-0800 EP ITAAC guidance and meet the requirements of 10 CFR 52.80(a). Therefore, the staff finds the RAI responses to be acceptable. The staff has incorporated the proposed markup to Table 3.8-1 into SER Table 13.3-1.

By letter dated April 18, 2013, from PEF to NRC, the applicant proposed a revision to the LNP Emergency Plan to address the future state of CR3 as it relates to decommissioning activities and the anticipated relaxation of offsite EP responsibilities for CR3. In consideration of these circumstances, the applicant anticipates the EOF will no longer be required for response to an emergency event at CR3. In LNP Emergency Plan, Revision 6, the EOF has been renamed the LNP EOF and is expected to support the future needs of LNP only. The staff anticipates a lapse in time for which the readiness capabilities of the EOF will no longer be required. By letter dated January 10, 2014, from DEF to the NRC, the applicant proposed EP ITAAC 7.2.3 through 7.2.5 to address regulatory guidance criteria in NUREG-0696 and Supplement 1 to NUREG-0737 that are not addressed in the LNP Emergency Plan. In addition, the applicant proposed a revision to the Inspection, Tests, and Analyses (ITA) text for item 7.2 to clarify that an inspection of the as-built EOF will be performed and the facility will meet the regulatory guidance criteria. Prior to fuel load, these EP ITAAC will provide staff assurance confirm that the EOF continues to comply with the uniform building code; the EOF is environmentally controlled to provide room air temperature, humidity, and cleanliness appropriate for personnel and equipment; and the EOF is provided with industrial security when it is activated to exclude unauthorized personnel and when it is idle to maintain its readiness. Given that the EOF may not be required to maintain its functionality for some time prior to LNP operations, the staff found these ITAAC necessary to ensure that the EOF is constructed as designed, as required by 10 CFR 52.80. Therefore, the staff finds the applicant’s proposed EP ITAAC 7.2.3 through 7.2.5 and text revision to ITAAC 7.2 acceptable since they conform to the guidance in NUREG-0696 and Supplement 1 to NUREG-0737 and meet the requirements in 10 CFR 52.80.

The staff reviewed the EP ITAAC provided in Table 3.8-1 of Appendix B to Part 10 of the LNP COL application, as modified by the applicant’s letters dated June 3, 2009, December 18, 2009, March 26, 2010, and March 15, 2011, and January 10, 2014, and confirmed that each of the generic EP ITAAC in NUREG-0800, Table 14.3.10-1, that provides an acceptable set of generic EP ITAAC were included in Table 3.8-1. The staff further confirmed that the proposed EP ITAAC have been tailored to the specific reactor design and emergency planning program requirements of the LNP site. The complete set of EP ITAAC are provided in SER Table 13.3-1, which is based on Table 3.8-1 of Appendix B to Part 10 of the LNP COL application, as modified by the applicant’s letters as described above in this section of the SER. Therefore, the staff finds that the LNP COL application adequately provides EP ITAAC as required by 10 CFR 52.80(a).

13.3C.19.3 *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 EP ITAAC, and there is no outstanding information expected to be addressed in the LNP COL application related to this section. The results of the staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

As required by 10 CFR 52.80(a), the staff concludes that the EP ITAAC in SER Table 13.3-1 include the proposed EP ITAAC that the licensee shall perform, and that are necessary and sufficient to provide reasonable assurance that, if the ITAAC are performed and met, the facility has been constructed and will operate in conformity with the license, the provisions of the Atomic Energy Act, and the NRC's rules and regulations.

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," dated 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 Staff Requirements Memorandum (SRM), dated February 22, 2006. 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

Section 13.4 of the LNP COL FSAR, Revision 9, incorporates by reference Section 13.4 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Section 13.4 and in Part 10 of the LNP COL application, "Proposed License Conditions and ITAAC [inspections, tests, analyses, and acceptance criteria])," 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 Appendix F of NUREG-1793 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 SER 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 Section 13.4 of the LNP 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 operational programs. The results of the staff's evaluation of the information incorporated by reference in the LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 13.4.4 of the VEGP SER:

> *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 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 are 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 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 operational programs, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The staff concludes that the relevant information presented in the LNP 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 SER 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 LNP COL FSAR Table 13.4-201 has been found acceptable by the NRC staff in other sections of this SER, 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. The applicant, in its plant procedures, 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

Section 13.5 of the LNP COL FSAR, Revision 9, incorporates by reference Section 13.5 of the AP1000 DCD, Revision 19.

In addition, in LNP 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.

- LNP COL 13.5-1

The applicant provided additional information in LNP COL 13.5-1 related to procedures to control radionuclide inventories and personnel doses in the Radwaste Building. This information, as well as related additional FSAR information in LNP COL 11.2-1 and proposed License Condition 13 in Part 10 of the COL application, is reviewed in Section 11.2 of the SER.

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 the Commission regulations for plant procedures are given in Sections 13.5.1.1 and 13.5.2.1 of NUREG-0800.

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

13.5.4 Technical Evaluation

The NRC staff reviewed Section 13.5 of the LNP 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. The NRC 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 NRC staff's evaluation of the information incorporated by reference in the LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 13.5.4 of the VEGP SER:

> *<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 procedures] *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 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.*

The staff finds this change acceptable as it is only a change of position title and meets the guidance of NUREG-0800, Section 13.5.1.1. This is Tier 2 information and NRC approval is not required.

13.5.5 Post Combined License Activities

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

13.5.6 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 plant procedures, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP 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 LNP COL FSAR Table 1.9-202, the applicant identified two exceptions to the criteria of NUREG-0800, Section 13.5, related to providing 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 to be acceptable because the applicant's procedure classification is consistent with the guidance in NUREG-0800, Section 13.5.

13.6 Physical Security

13.6.1 Introduction

The COL application for the LNP Units 1 and 2 describes the COL applicant's physical protection program, which is intended to meet NRC regulations for protection against the design basis threat (DBT) of radiological sabotage as stated in 10 CFR 73.1, "Purpose and Scope," and provide a high assurance that activities involving special nuclear material are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety.

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 COL application consists of the LNP Units 1 and 2 Physical Security Plan (PSP), Training and Qualification Plan (T&QP), and Safeguards Contingency Plan (SCP). Section 13.6 of the LNP COL FSAR 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." Those persons with the correct access authorization and need-to-know may view the safeguards information version of the LNP COL application Section 13.6 SER, which is located in the NRC's Secure Local Area Network, document number ES100017759.

13.6.2 Summary of Application

Section 13.6 of the LNP COL FSAR, Revision 9, incorporates by reference Section 13.6 of the AP1000 DCD, Revision 19.

Part 8 – Safeguards/Security Plans

In a letter dated July 2, 2008, PEF submitted a PSP to the NRC as part of the COL application for proposed LNP Units 1 and 2. In a letter dated July 7, 2009, PEF submitted Revision 1 to the

PSP. In a letter dated September 9, 2009, PEF submitted Revision 2 to its PSP. In a letter dated April 19, 2011, PEF submitted Revision 3 to its PSP. In a letter dated June 3, 2011, PEF submitted Revision 4 to its PSP.

In addition, in LNP 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 SER.

License Conditions

- Part 10, License Condition 3, Items D.3 and G.9

The applicant proposed a license condition in Part 10 of the LNP COL application, which provides the milestones for implementing applicable portions of the Security Program.

- Part 10, License Condition 5

The applicant proposed a license condition in Part 10 of the LNP COL application, 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 the NRC's 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.

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 (v) require that the application include an SCP in accordance with the criteria set forth in Appendix C, "Nuclear Power Plant Safeguards Contingency Plans" to 10 CFR Part 73, and a T&QP in accordance with Appendix B, "General Criteria for Security Personnel" of 10 CFR Part 73, that the applicant provide a description of the implementation of the SCP and the T&QP and that the applicant protect the PSP, SCP and T&QP 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 it will meet the regulatory requirements of 10 CFR Part 73 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 it will meet provisions of physical protection system requirements in the NRC regulations and guidance provided in NUREG-0800 and RGs listed below. However, the NRC RGs 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 the NRC regulations.

NUREG-0800 Section 13.6.1, Revision 1, June 15, 2010 was used by the NRC 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 letter dated April 9, 2009, 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 guidance 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 NRC staff reviewed Section 13.6 of the LNP 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 physical security. The results of the NRC staff’s evaluation of the information

incorporated by reference in the LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

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

- The staff compared the VEGP PSP, T&QP, and SCP to the corresponding LNP programs. The staff has determined that these plans are sufficiently similar to warrant standard content treatment.

- The staff confirmed that all applicant 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 LNP COL application, with the exception discussed in the following paragraph. This standard content material is identified in this SER by use of italicized, double-indented formatting. One clarification to the standard content material presented below is that the NRC staff's detailed evaluation of the physical protection program, which is site-specific, is provided in the safeguards information version of the LNP COL application Section 13.6 SER, which is located in the NRC's Secure Local Area Network.

There were site-specific RAIs issued to the LNP 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 LNP 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 LNP 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 Section 13.6.4 of the VEGP SER:

<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 a 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 a 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*[security plan] *(security prior to receipt of fuel onsite (protected area)) appropriate and in accordance with the requirement in 10 CFR 73.55, "Requirements for physical protection of* *licensed activities in nuclear power reactors against radiological sabotage." 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):

(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 Appendix C to 10 CFR Part 73. 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 Appendix B to 10 CFR Part 73;

(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; and

(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 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 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. Section 1 of the PSP describes the applicant’s commitment to satisfying 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.

Section 1.1 of the LNP PSP 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, Section 1.2 of the LNP COL application provides general plant descriptions that include details of the 10 to 50 mile radius of the geographical area of the LNP Units 1 and 2 site, a site area map, and general plant and site descriptions. LNP COL FSAR, Chapter 1, references the AP1000 DCD for the principal design and operating characteristics for the design and construction of the LNP Units 1 and 2. Part 1, General Information, of the LNP COL application describes the name of the applicant and principal business locations.

The NRC staff has reviewed the facility physical layout provided in Section 1.1 of the PSP and as supplemented by LNP COL FSAR. The NRC staff determined 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 NRC staff concludes that the applicant's security plans have met the requirements for content of a PSP as stated above. Therefore, the NRC staff finds the "Facility Layout" described in the PSP and the LNP COL FSAR is adequate.

The following portion of this technical evaluation section is reproduced from Section 13.6.4.1 of the VEGP SER:

> *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

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.

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, 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.

Section 4.0 of the PSP describes how the applicant meets the requirements of 10 CFR 73.55(d)(1).

Security Organization Management

Section 4.1 of the PSP describes the organization's management structure. The PSP establishes that the security organization is a critical component of the physical protection program and is responsible for the effective application of engineered systems, technologies, programs, equipment, procedures, and personnel necessary to detect, assess, interdict, and neutralize threats up to and including the DBT of radiological sabotage. The security organization may be proprietary, 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. Section 4.1 of the PSP also describes the roles and responsibilities of the Security Organization. The PSP provides that at least one full-time, dedicated Security Shift 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 a proprietary force, contractor, or other qualified personnel. The training qualification requirements are described in the T&QP.

The NRC 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 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 following portion of this technical evaluation section is reproduced from Section 13.6.4.1 of the VEGP SER:

> *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

Consistent with requirements in 10 CFR 73.55(d)(3),10 CFR 73.56, "Personnel access authorization requirements for nuclear power plants"; 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 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 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, 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.

Section 9 of the PSP 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 Commission-approved security plans, applicant's protective strategy, and implementing procedures.

The NRC staff has reviewed the applicant's description in PSP Section 9 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because 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.

The following portion of this technical evaluation section is reproduced from Section 13.6.4.1 of the VEGP SER:

> *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, "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."

Section 11 of the PSP provides a general description of how the applicant has implemented its program for physical barriers, and that this implementation is in accordance with the performance objectives and requirements of 10 CFR 73.55(b).

Owner Controlled Area (OCA) Barriers

Section 11.1 of the PSP describes LNP 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) are included in the facility procedures.

In **RAI 13.6-3**, the NRC staff requested that the applicant provide an additional description of natural terrain features that make-up portions of the outer VBS and provide reference to the criteria used to determine its acceptability and stand-off distances. If applicable, this additional information should be incorporated in the Facility Physical Layout Drawing.

In PEF Response Letter No. 066, dated October 22, 2009, the applicant indicated that the design of the VBS has not been finalized; however, the conceptual design shall consist of both active and passive barriers. Each engineered feature utilized to form a contiguous barrier will be designed and located in accordance with guidance from RG 5.76 and/or NUREG/CR 6190, as appropriate, in order to provide a standoff distance beyond the minimum distance required for protection of all current DBT criteria.

On the basis of its review, the NRC staff finds the response to RAI 13.6-3 to be acceptable because the proposed changes follow the guidance from RG 5.76 and NUREG/CR 6190. The staff considers this RAI closed.

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."

Section 11.2.3 of the PSP describes that a site-specific analysis for a water-borne DBT has been conducted and documented. However, there is no waterborne access to LNP, 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: (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 (PA) barrier are provided in the PSP Section 11.3. 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).

Section 11.3 of the PSP 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 PA barrier is not separated from a vital area barrier.

Section 11.3 of the PSP describes isolation zones. As required in 10 CFR 73.55(e)(7), the isolation zone is maintained in outdoor areas adjacent to the 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 Commission and identified in the security plans." In addition, 10 CFR 73.55(e)(5) requires that certain vital areas shall be bullet resisting.

Section 11.4 of the PSP describes that vital areas are restricted access areas surrounded by physical barriers with the capability to restrict access to only authorized individuals. All vital areas are constructed in accordance with established regulatory requirements. Section 11.4 also 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 13.6-18, the NRC staff asked for clarification regarding functionality in certain vital areas. The PEF Response Letter No. 066, dated October 22, 2009, confirmed that the response provided in R-COLA RAI 13.6-13 (VEGP eRAI 3394) is also applicable to Levy Nuclear Plant.

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 § 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."

Section 11.5 of the PSP 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 to be acceptable for the DC review of the AP1000 physical protection program. Westinghouse stated, in TR-94, 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 PA).

The NRC staff has reviewed the applicant's description in Sections 11.5 and 14.5 of the PSP, Section 7 of the SCP 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 Commission regulations and NUREG-0800 acceptance criteria. Because the applicant's description in Sections 11.5 and 14.5 of the PSP, Section 7 of the SCP 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 Sections 11.5 and 14.5 of the PSP and Section 7 of the SCP 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, which were not subject to an NRC staff review as part of this COL application, and are, therefore, 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 LNP, Units 1 and 2.

Section 11.6 of the PSP includes a description of the use of Delay Barriers to meet requirement of 10 CFR 73.55(e).

The NRC staff has 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 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), and is, therefore, acceptable.

The following portion of this technical evaluation section is reproduced from Section 13.6.4.1 of the VEGP SER:

> *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*
>
> *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

<u>Access Authorization and Fitness for Duty</u>

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.

Section 14.1 of the PSP describes that the access authorization program implements regulatory requirements utilizing the provisions in RG 5.66. "Nuclear Power Plant Access Authorization Program," Revision 1, dated July 2009. The NRC staff finds that RG 5.66, is an acceptable method for meeting the requirements of 10 CFR 73.55(b)(7).

The NRC staff has 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. 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)(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.

Section 14.2 of the PSP 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, Section 14.2 describes the integration of the programs mentioned above to form a cohesive and effective insider mitigation program. The applicant addresses the observations for the detection of tampering. The NRC staff finds that this approach is an acceptable method for meeting the requirements 10 CFR 73.55(b)(9).

The NRC staff has reviewed the applicant's description in PSP Section 14.2 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because 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.

Section 14.3 of the PSP 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 NRC staff has reviewed the applicant's description in PSP Section 14.3 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because 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, which 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.

Section 14.4 of the PSP 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 the applicant to search vehicles at the owner controlled area and 10 CFR 73.55(h)(3) provides requirements for searches of personnel, vehicles and materials prior to entering the protected area.

Section 14.4.1 of the PSP 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 states that it has developed specific implementing procedures to address vehicle and materials searches at these locations.

PA Packages and Materials Search

Section 14.4.2 of the PSP describes the process for conducting searches of packages and materials for firearms, explosives, incendiary devices, or other items, which 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 LNP, Units 1 and 2 protected area. Detailed requirements 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.

PA Vehicle Search

Section 14.4.3 of the PSP describes the process for the search of vehicles for firearms, explosives, incendiary devices, or other items, which 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 requirements 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.

PA Personnel Searches

Section 14.4.4 of the PSP 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, which 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 LLEA 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.

PA Access Controls

Section 14.4.5 of the PSP describes the process for controlling access at all points where personnel or vehicles could gain access into 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 PA after positive ID verification, access authorization verification, and a search is performed per Section 14.4 of the PSP. 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 of this chapter. 10 CFR 73.55(g)(8) discusses escort requirements. Applicants are required to implement procedures for processing, escorting and controlling 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.

Section 14.4.6 of the PSP describes the process for control of visitors. The PSP affirms that procedures address the identification, processing, and escorting of visitors and the maintenance of a visitor control register. Training requirements for escorting visitors includes responsibilities, communications and escort ratios. All escorts are trained to perform escort duties in accordance with site requirements. All visitors wear a badge that clearly indicates that an escort is required.

The NRC staff has reviewed 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 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(h)(2), (h)(3), (g)(7) and (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 in order to perform their duties and responsibilities. The list must be approved by a cognizant applicant manager or supervisor who is responsible for directing the work activities of the individual who is granted unescorted access to each vital area, and updated and re-approved no less frequently than every 31 days.

Section 14.5 of the PSP describes vital areas and states that 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 which is reassessed and reapproved at least once every 31 days. Additional access control measures are described in the facility procedures.

In **RAI 13.6-9**, the NRC staff asked the applicant to 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 PGN Response Letter No. 066, dated October 22, 2009, the applicant stated that the R-COLA RAI 13.6-19 response from Vogtle Electric Generating Plant (VEGP), dated October 16, 2009, is applicable to Levy Units 1 and 2. In a letter dated May 4, 2011, the applicant provided a description which clearly identifies the minimum vital areas. On the basis of its review, the NRC staff finds the revised description in the PSP Revision 4, dated June 3, 2011, to be 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 NRC staff has reviewed the applicant's description in PSP Section 14.5 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because 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.

The following portion of this technical evaluation section is reproduced from Section 13.6.4.1 of the VEGP SER:

> *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.*
>
> *<u>Illumination</u>*
>
> *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.*
>
> *<u>Surveillance Systems</u>*
>
> *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.

<u>Protected and Vital Area Patrols</u>

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 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 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.*

In **RAI 13.06.01.**, the NRC staff requested clarification pertaining to how the 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 FSAR Table 13.4-201 to ensure that the security plan continues to meet 10 CFR 73.55(b)(4). This RAI also requested the applicant to 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 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 NRC 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 NRC 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.

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.

Section 18 of the PSP describes an armed response team, responsibilities, training, and equipment, and requires an adequate number of armed response force personnel immediately available at all times to implement each site's protective strategy. The applicant ensures that training is conducted 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. Procedures are in place to manage the threat warning system.

In RAI 13.6-27 the NRC staff requested that the licensee clarify PSP, Section 18, which details the minimum number of armed responders continuously in the protected area. The staff requested the applicant explain how this number correlates with the expected number detailed in Westinghouse Technical Report (TR) 94, AP1000 Safeguards Assessment Report.

In a letter dated May 4, 2011, the applicant provided an explanation of how they determined the minimum numbers of Armed Responders needed for the LNP 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 LNP Site Units 1 and 2.

On the basis of its review, the NRC staff finds the response to RAI 13.6-27 to be 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 NRC staff has reviewed the applicant's description in PSP Section 18 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because 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.

The following portion of this technical evaluation section is reproduced from Section 13.6.4.1 of the VEGP SER:

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 or 10 CFR Part 52, 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.

<u>Refueling/Major Maintenance</u>

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).

<u>Construction and Maintenance</u>

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

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 Sections 20 and 20.1 through 20.6 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because 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 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 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, "Records," 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

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 of 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 10 CFR 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."

Suspension of Security Measures in Accordance with 10 CFR 50.54(x) and (y)

Section 24.1 of the PSP addresses suspension of security measures in accordance with 10 CFR 50.54(x) and 10 CFR 50.54(y). Specifically, the plan 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, "Reporting of safeguards events." 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

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

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

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 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.*

<u>*Physical Examination*</u>

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.

<u>*Medical Examinations and Physical Fitness Qualifications*</u>

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.

<u>Documentation</u>

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.

<u>Physical Requalification</u>

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. These areas of training include performing assigned duties and responsibilities in accordance with the requirements of the T&QP and the PSP, and be trained and qualified in the use of all equipment or devices required to effectively perform all assigned duties and responsibilities.

Section 3.1 of the T&QP details the requirements that individuals assigned duties must be trained in their duties, meet minimum qualifications, and be trained and qualified in all equipment or devices required to perform their duties.

The NRC staff has reviewed 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 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.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 must include individual demonstration of the knowledge, skills and abilities provided during the training process. Individuals 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 NRC staff has reviewed the applicant's description in T&QP Section 3.2 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because 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 Commission-approved security plans, licensee protective strategy, and implementing procedures shall, before assignment, demonstrate proficiencies in implementing the knowledge, skills and abilities to perform assigned duties.

The T&QP includes a critical task matrix as Table 1 of the T&QP. 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 NRC staff has reviewed the applicant's description in T&QP Section 3.3 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because 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.

Section 3.4 of the T&QP 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 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) 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, to 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.

Sections 3.4.1 and 3.4.2 of the T&QP describe the measures that are implemented by the applicant that meet the requirements stated above.

The NRC staff has 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 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.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 Commission-approved T&QP. The results of requalification must be documented by a qualified training instructor and attested by a security supervisor.

Section 3.5 of the T&QP 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 three (3) months before or three (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 NRC staff has reviewed the applicant's description in T&QP Section 3.5 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because 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 Commission-approved T&QP to demonstrate an acceptable understanding of assigned duties and responsibilities.

Section 3.5.1 of the T&QP provides that each individual will be tested, in part, with an annual written exam that, at a minimum, covers: the role of security personnel; use of deadly force; the requirements in 10 CFR 73.21; authority of private security personnel; power of arrest; search and seizure; offsite law enforcement response; tactics and tactical deployment and engagement.

The NRC staff has 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 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.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, B.2(c)(2), B.3(a), B.4(b)(1), B.4(b)(3), B.5(a), C.2(a), C.2(b), C.3(a), C.3(b) C.3(d), D.1(a), D.1(b)(1), D.1(b)(2), D.1(b)(3), and D.1(c)) state, in part, that an individual must demonstrate required knowledge, skills and abilities, to carry out assigned duties and responsibilities.

Section 3.5.2 of the T&QP provides that all knowledge, skills and abilities will be demonstrated in accordance with a systematic approach to training (SAT) program as described in RG 5.75.

The NRC staff has 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 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.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 Commission-approved T&QP. Training must be conducted by certified firearms instructors who shall be recertified at least every three (3) years. Applicants shall conduct annual firearms familiarization, and armed members of the security organization must participate in weapons range activities on a nominal four (4) month periodicity.

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 four (4) month periodicity.

The NRC staff has 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 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.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 Commission requirements and the Commission-approved T&QP for assigned weapons. The results of weapons qualification and requalification must be documented and retained as a record.

Section 3.6.2 of the T&QP provides that all armed personnel are qualified and re-qualified with assigned weapons. All weapons qualification and re-qualification will be documented and retained as a record.

The NRC staff has 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 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.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 Commission-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.

Section 3.6.3 of the T&QP provides that a tactical qualification course of fire is used to assess armed security force personnel in tactical situations to ensure they are able to demonstrate required tactical knowledge, skills and abilities remain proficient.

The NRC staff has 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 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.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.

Section 3.6.4 of the T&QP provides a description of the firearms qualification courses used to ensure armed members of the security organization are properly trained and qualified. Courses of fire are used individually for handguns, semiautomatic rifles, and enhanced weapons.

The NRC staff has reviewed the applicant's description in T&QP Section 3.6.4 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because 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 Commission requirements and the Commission-approved T&QP, and the results documented and retained as a record. Firearms requalification must be conducted using the courses of fire outlined in 10 CFR Part 73, Appendix B, Sections VI.F.2, VI.F.3, and VI.F.4.

Section 3.6.5 of the T&QP describes that armed members of the security organization re-qualify at least annually with each weapon assigned, using the courses of fire provided in the T&QP.

The NRC staff has 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 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.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 the maintenance of weapons and personal equipment. These requirements provide that the applicant shall provide armed personnel with weapons that are capable of performing the function stated in the Commission-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 Commission-approved security plans, applicant protective strategy, and implementing procedures.

Section 3.7 of the T&QP 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 Section 9.0 of the PSP, and maintenance is performed as described in Section 20.0 of the PSP.

The NRC staff has reviewed the applicant's description in T&QP Section 3.7 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because 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. The staff's review of Sections 9.0 and 20.0 of the PSP is in Section 13.6.4.1.9 and 13.6.4.1.20 of this SER.

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 three (3) years after termination of the individual's employment and shall retain each re-qualification record for three (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 three (3) years from the date of obtaining and recording these results.

Section 3.8 of the T&QP provides that records are retained in accordance with Section 22 of the PSP.

The NRC staff has reviewed the applicant's description in T&QP Section 3.8 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because 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.

The following portion of this technical evaluation section is reproduced from Section 13.6.4.2 of the VEGP SER:

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 <u>Appendix C Safeguards Contingency Plan</u>

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.

<u>Perceived Danger</u>

The provisions of 10 CFR Part 73, Appendix C, Section II.B.1.a 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.

<u>Definitions</u>

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.3 and are, therefore, acceptable.

13.6.4.3.2 Generic Planning Base

As required in 10 CFR Part 73, Appendix C, Section II.B.2, this section of the plan defines the criteria for initiation and termination of responses to security events, 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 for 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.

Section 3 of the SCP includes the responsibility matrix. The responsibility matrix integrates the response capabilities of the security organization (described in Section 4 of the SCP) with the background information relating to decision/actions and organizational structure (described in Section 1 of the SCP). The responsibility matrix provides an overall description of the response actions and their interrelationships. Responsibilities and actions have been predetermined to the maximum extent possible and assigned to specific entities to preclude conflicts that would interfere with or prevent the implementation of the SCP or the ability to protect against the DBT of radiological sabotage.

The NRC staff has reviewed the applicant's description in SCP Section 3 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because 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

Section 4.1.1 of the SCP details the duties and communications protocols of each member of the security organization responsible for implementing any portion of the applicant's protective strategy.

Security Chain of Command/Delegation of Authority

Section 4.1.2 of the SCP details 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.

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.

Section 4.2 of the SCP references Section 1.1 of the PSP for layouts of the OCA, PA, vital areas, site maps, and descriptions of site features.

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.

Section 4.3 of the PSP describes that safeguards systems are described in PSP Sections 9, 11, 12, 13, 15 and 16, and in facility implementing procedures/documents. Section 8 of the SCP describes how physical security systems will be used to respond to a threat at the site.

Law Enforcement Assistance

The 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.

Section 4.4 of the SCP details the role of LLEA in the site protective strategy. Additional details regarding LLEA are included in Section 8 of the PSP and Section 5.6 of the SCP.

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.

Section 4.5 of the SCP details the site security policies, including the use of deadly force and authority to request offsite assistance.

Administrative and Logistical Considerations

The 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.

Section 4.6 of the SCP outlines administrative duties of the Security Manager, Security Shift Team Leader, facility procedures and administrative forms.

The NRC staff has 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 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.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. The applicant details how they protect against the DBT with onsite and offsite organizations, consistent with the regulation 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. 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 Appendix E to 10 CFR Part 50 and 10 CFR 52.17.

Response to Threats

Section 5.1 of the SCP 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

Section 5.2 of the SCP 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

Section 5.3 of the SCP details 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

Section 5.4 of the SCP details the role of operations personnel in the site protective strategy, including responsibilities, strategies, and conditions for operator actions as discussed in 10 CFR 50.54(hh).

Emergency Plan Response

Section 5.5 of the SCP notes the integration of the 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)

Section 5.6 of the SCP meets the requirements of 10 CFR 73.55(k)(9) and 10 CFR Part 73, Appendix C, Section II.B.3.d and lists the LLEAs that will respond to the site as a part of the protective strategy. Details on the response of the LLEA are located in Section 8 of the PSP.

State Response Agencies

Section 5.7 of the SCP meets the requirements of 10 CFR 73.55(k)(9) and 10 CFR Part 73, Appendix C, Section II.B.3.d and lists the State response agencies that will respond to the site as a part of the protective strategy.

Federal Response Agencies

Section 5.8 of the SCP meets the requirements of 10 CFR 73.55(k)(9) and 10 CFR Part 73, Appendix C, Section II.B.3.d and lists the Federal response agencies that will respond to the site as a part of the protective strategy.

Response to ISFSI Events

Section 5.9 of the SCP meets the requirements of 10 CFR 73.55(k)(9) and 10 CFR Part 73, Appendix C, Section II.B.3.d describes the Response Requirements for ISFSI as a part of the protective strategy.

The NRC staff has 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 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 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 Appendix E to 10 CFR Part 50 and 10 CFR 52.17.

The following portion of this technical evaluation section is reproduced from Section 13.6.4.3 of the VEGP SER:

> *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*
>
> *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.

Section 8 of the SCP describes the site protective strategy.

The NRC staff has reviewed the applicant's description in SCP Section 8 for the implementation of the site-specific physical protection program in accordance with Commission regulations and NUREG-0800 acceptance criteria. Because 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.

The following portion of this technical evaluation section is reproduced from Section 13.6.4.3 of the VEGP SER:

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 (13-8) – 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 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 physical security, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The staff concludes that the relevant information presented in the LNP COL FSAR is acceptable based on the applicable regulations specified in Section 13.6.4 of this SER. 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 NRC staff's review of the LNP Units 1 and 2 PSP, T&QP, and SCP has 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 NRC staff has 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 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). As required by Section 3 of the applicant's PSP, a performance evaluation program will be implemented that periodically tests and evaluates the effectiveness of the overall protective strategy. This program requires that deficiencies be corrected. In addition, NRC inspectors will conduct periodic force-on-force exercises that will test the effectiveness of the applicant's protective strategy. Based on the results of the applicant's own testing and evaluation, the NRC's baseline inspections and force-on-force exercises, enhancements to the applicant's PSP, T&QP, and SCP may be required to ensure the overall protective strategy can be effectively implemented. As such, 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 to 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

13.6.A.1 Introduction

Part 10, "Proposed License Conditions and ITAAC," Appendix B, "Inspections, Tests, Analysis, and Acceptance Criteria" of the LNP COL application 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 Tier 1 Section 2.6.9 of the AP1000 DCD, 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 LNP PSP, which includes the site PSP, T&QP and the SCP.

13.6.A.2 Summary of Application

Section 14.3 of the LNP COL FSAR, Revision 9, incorporates by reference Section 14.3 of the AP1000 DCD, Revision 19. Part 10 of the LNP COL application incorporates by reference 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 Table 2.6.9-2 of Appendix B to Part 10 of the LNP COL application.

In addition, in LNP COL FSAR Section 14.3, the applicant provided the following:

Supplemental Information

- STD SUP 14.3-1

The applicant provided supplemental (SUP) information related to physical security in STD SUP 14.3-1 in LNP COL FSAR Section 14.3.2.3.2.

License Condition

- Part 10, License Condition 1

The applicant provided a license condition in Part 10 of the LNP COL application which will incorporate the ITAAC identified in the tables in Appendix B. The staff evaluates this license condition in Chapter 1 of this SER.

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 the Commission 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, and the NRC's regulations.

The LNP 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 Commission 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 Section 14.3.12 of NUREG-0800.

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 NRC staff reviewed Section 14.3 of the LNP 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 ITAAC for physical security. The results of the NRC staff’s evaluation of the information incorporated by reference in the LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP COL application. This standard content material is identified in this SER by use of italicized, double-indented formatting. The staff confirmed that the PEF letter dated September 23, 2010, 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 Section 13.6.A.4 of the VEGP SER:

<u>Supplemental Information</u>

- *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 NRC staff finds that since the applicant revised LNP COL FSAR Part 10 to incorporate the requirements for PS-ITAAC, the response to RAI 14.03.12- 1, 2 & 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 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 PS-ITAAC, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The staff concludes that the relevant information presented in LNP COL FSAR and the additional information received in the PEF letter dated September 23, 2010, is acceptable based on the applicable regulations specified in Section 13.6.A.4 of this SER. 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 NRC 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 LNP COL FSAR Chapter 13 and referenced TRs. The applicant has provided adequate descriptions of objectives, prerequisites, test methods, data required, and acceptance criteria for security related ITAAC for the approval of the LNP COL.

Table 13.6A-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.

Table 13.6A-1 – Site-Specific Physical Security Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, and Analyses	Acceptance Criteria
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.

Table 13.6A-1 – Site-Specific Physical Security Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, and Analyses	Acceptance Criteria
4. The intrusion detection and assessment equipment at the protected area perimeter:	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 both the central alarm station and secondary alarm station, and		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.		b) remains operable from an uninterruptible power supply in the event of the loss of normal power.

Table 13.6A-1 – Site-Specific Physical Security Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, and Analyses	Acceptance Criteria
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.

Table 13.6A-1 – Site-Specific Physical Security Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, and Analyses	Acceptance Criteria
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 FFD program required by 10 CFR Part 26 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

LNP COL FSAR Section 13.7 is a new section added after Section 13.6 of the AP1000 DCD. The references that are currently in AP1000 DCD Section 13.7 have been redistributed to other LNP COL FSAR sections. There is no information associated with the FFD program incorporated by reference from the AP1000 DCD.

In addition, in LNP COL FSAR Section 13.7, the applicant provided the following:

Supplemental Information

- STD SUP 13.7-1

The applicant provided standard supplemental information in LNP 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 structures, systems, and components (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 the NRC's inspection of operational programs included in the LNP 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
- 10 CFR 52.79(a)(44)

Regulatory guidance for FFD programs is included in RG 1.206.

13.7.4 Technical Evaluation

The NRC staff reviewed Section 13.7 of the LNP COL FSAR to ensure that the COL application represents the complete scope of information relating to this review topic. The NRC staff review confirmed that the information in the application addresses the required information relating to the FFD program.

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 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP COL FSAR (and other parts of the COL application, as applicable) resulting from 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 LNP COL application. This standard content material is identified in this SER 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 LNP applicant (which is a typical step when comparing the two applications), the NRC staff provides its evaluation of similar RAIs issued to LNP, 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 Section 13.7.4 of the VEGP SER:

> <u>*Supplemental Information*</u>
>
> - *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:

		Program Source	*FSAR*	*Implementation*	
Item	*Program Title*	*(required by)*	*Section*	*Milestone*	*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	Implementation 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 LNP RAIs

The NRC staff issued RAIs to the LNP applicant, the first three of which mirrored the RAIs issued to the VEGP applicant. Specifically, RAIs 13.06.01-1, 13.06.01-2, and 13.06.01-3 issued to the LNP applicant correspond to RAIs 13.6-33, 13.6-34, and 13.6-35, respectively, issued to the VEGP applicant. In addition, the NRC staff issued RAI 13.06.01-4 to LNP.

The NRC staff evaluation of the responses provided by the LNP applicant to the four questions related to the FFD program is discussed below. The LNP applicant responded to these four RAIs in a letter dated March 26, 2010.

In response to RAI 13.06.01-1, the LNP applicant stated that it currently commits to NEI 06-06, Revision 4, and will change its application in a future revision to commit to NEI 06-06, Revision 5. The LNP applicant stated that it 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 13.06.01-1 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 LNP 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 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 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.

In response to RAI 13.06.01-2, the LNP applicant stated that the response to RAI 13.06.01-1 provides the changes to the COL application that will describe the FFD program required by 10 CFR Part 26. The 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 response to RAI 13.06.01-3 provides the information on modifications to LNP COL FSAR Table 13.4-201, Item 20 to address the guidance provided in the NRC's December 2, 2009, letter to NEI. That RAI response includes changes to License Condition 3, Items A, C, and D in Part 10 of the COL application to align with the changes to LNP COL FSAR Table 13.4-201. The NRC staff verified that the proposed changes to LNP COL FSAR Table 13.4-201, Item 20 are identical to the proposed changes to the corresponding VEGP COL FSAR Table 13.4-201, which is provided in the standard content evaluation material above.

In RAI 13.06.01-3, the staff asked the applicant if proposed License Condition in, 3, A, D and G 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.

In response to RAI 13.06.01-3 the LNP applicant stated the response to R-COLA RAI 13.06-35 (VEGP eRAI 4216) is also applicable to LNP, and it does not require additional review.

The staff further evaluated the need for License Condition 3, A, D and G, for the LNP 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 LNP, and removed the license conditions with the issuance of COL FSAR Revision 2.

In RAI 13.06.01-4 the staff asked the applicant to explain the word “onsite,” which is contained in the COL application, Part 2, FSAR, Table 13.4-201 (Sheet 7 of 7), item number 20, for FFD Programs for Construction – Mgt & Oversight Personnel, in the milestone description. This was in contrast to the item for FFD Programs for Construction - Workers & First Line Supervisors, which is the same, but does not include “onsite” in its wording. Although construction is defined in 10 CFR 50.10 and 10 CFR 26.5, these definitions do not include the additional word “onsite.”

In response to RAI 13.06-4, LNP stated that RAI response 13.06.01-3, FSAR Table 13.4-201 will be modified to address the guidance in the NRC’s letter to the Nuclear Energy Institute 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,’” which will make all implementation milestones consistent.

The NRC staff compared the responses to the first three RAIs provided by the LNP applicant to the responses provided by the VEGP applicant, and concluded that the responses are essentially identical, after accounting for the differences of an Early Site Permit having been issued for the VEGP site for this issue. Therefore, the conclusions reached by the NRC staff regarding the FFD program at VEGP are applicable to the FFD program at LNP.
RAI 13.06.01-4, the LNP item not included in the VEGP RAIs, was a minor, one-word clarification. The inclusion of the information provided in the RAI responses in a future revision of the LNP COL FSAR is part of **Confirmatory Item 13.7-1** that is discussed in the standard content portion of this safety evaluation above.

Resolution of Levy Site-Specific Confirmatory Item 13.7-1

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 LNP COL FSAR was appropriately revised. As a result, Confirmatory Item 13.7-1 is now closed.

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 (13-9) – 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 NRC staff review confirmed that the applicant addressed the required information relating to the FFD program and there is no outstanding information to be addressed in the LNP COL FSAR related to this section.

The staff concludes that the information presented in the LNP COL FSAR is acceptable because it meets the regulatory requirements in 10 CFR Part 26 and 10 CFR 52.79(a)(44). The staff based its conclusion on the following:

- STD SUP 13.7-1, relating to the FFD program, is acceptable because it meets 10 CFR Part 26 and 10 CFR 52.79(a)(44).

13.8 <u>Cyber Security</u>

13.8.1 Introduction

In a letter to the NRC, dated July 23, 2010, PEF submitted Revision 1 of the CSP for LNP Units 1 and 2. The CSP applies to all critical digital assets (CDAs) required for LNP operation. In the submittal, the applicant describes how the requirements of 10 CFR 73.54 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. 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 Section 13.6 of the LNP COL FSAR. Section 13.6 of the LNP COL FSAR, Revision 9, incorporates by reference Section 13.6 of the AP1000 DCD, Revision 19. The applicant's CSP includes deviations from RG 5.71, "Cyber Security Programs for Nuclear Facilities." The staff has evaluated these deviations.

In addition, in LNP COL FSAR Section 13.6, the applicant provides the following:

<u>AP1000 COL Information Item</u>

- 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 2, COL Item 13.6-5 and License Condition 3, Item G.10

The applicant proposed a license condition in Part 10 of the LNP COL application requiring 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 Part 10 of the LNP COL application to provide a schedule to support the NRC's inspection of operational programs included in LNP 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, 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 NRC staff reviewed Section 13.6 of the LNP 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 cyber security. The results of the NRC staff's evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The staff's review of the LNP CSP has focused on ensuring that 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 reviewed the LNP CSP to assure 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 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

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

- The staff confirmed that the July 23, 2010, LNP 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, the names of the applicants 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 has completed its review and found the evaluation performed for the standard content to be directly applicable to the LNP COL application. This finding included verifying that the difference in the position charged with oversight of the program (the Vice President of Nuclear Engineering at LNP 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 Section 13.8.4 of the VEGP SER:

> _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 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 <u>Incorporating the Cyber Security Program into the Physical Protection Program (Section A.3.2 of Appendix A to RG 5.71)</u>

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 <u>Effectiveness Analysis (Section A.4.1.2 of Appendix A to RG 5.71)</u>

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 <u>Vulnerability Assessments and Scans (Section A.4.1.3 of Appendix A to RG 5.71)</u>

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 <u>Change Control (Section A.4.2 of Appendix A to RG 5.71)</u>

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 <u>Configuration Management (Section A.4.2.1 of Appendix A to RG 5.71)</u>

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 <u>Security Impact Analysis of Changes and Environment (Section A.4.2.2 of Appendix A to RG 5.71)</u>

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 <u>Security Reassessment and Authorization (Section A.4.2.3 of Appendix A to RG 5.71)</u>

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 <u>Updating Cyber Security Practices (Section A.4.2.4 of Appendix A to RG 5.71)</u>

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 <u>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)</u>

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 <u>Cyber Security Program Review (Section A.4.3 of Appendix A to RG 5.71)</u>

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.

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:

1) applying controls;

2) applying an alternative control that does not provide less protection than the corresponding control; or

3) 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, 13.8.24.53, 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. 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 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 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.

<u>License Conditions</u>

- *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 acceptable:

- License Condition (13-10) – 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 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 cyber security, and there is no outstanding information expected to be addressed in the LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

The NRC staff has reviewed the CSP for format and content using the NRC CSP template in RG 5.71, and found it to include all features considered essential to such a program. In particular the staff has found it to comply with applicable commission 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), 10 CFR 73.58, and 10 CFR Part 73, Appendix G.

14.0 INITIAL TEST PROGRAMS

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 and general plans for accomplishing it 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 Levy Nuclear Plant (LNP) COL Final Safety Analysis Report (FSAR), Revision 9, 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 LNP COL application are documented in NUREG-1793, "Final Safety Evaluation Report [FSER] 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 <u>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")</u>

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 LNP COL FSAR, Revision 9, 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 LNP COL FSAR Section 14.2.1, the applicant provided the following:

<u>*AP1000 COL Information Item*</u>

- 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 [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants."

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 [ITP] 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 LNP 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 LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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:

> *<u>AP1000 COL Information Item</u>*
>
> - *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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the information presented in the LNP 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 LNP COL FSAR, Revision 9, incorporates by reference Section 14.2 of the AP1000 DCD, Revision 19. Section 14.2 of the DCD includes Section 14.2.2.

LNP 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 LNP 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 LNP 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 LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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 [Plant Test & Operations (PT&O) Manager], 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 [Plant Test & Operations (PT&O) Support Manager], 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 [Startup Manager], 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 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.

In its letter dated March 7, 2011, LNP endorsed the VEGP letter dated Nov 11, 2010 that provided additional information on the training and qualification requirements for non-supervisory test engineers participating in initial test program activities.

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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that, the information presented in the LNP 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 LNP COL FSAR, Revision 9, 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 LNP 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.

In its letter dated May 4, 2011, the applicant proposed to add left margin annotations (LMA) for the following information items: STD COL 14.4-5, LNP COL 14.4-4, LNP COL 14.4-5, STD SUP 14.2-5, STD SUP 14.2-6, STD SUP 14.2-7, STD SUP 14.2-8, and LNP SUP 14.3-4 in a future revision to the LNP COL FSAR. These changes will be tracked as **LNP Confirmatory Item 14.2-1**. The staff confirmed that the LMA's were added at Revision 3 of the LNP FSAR. LNP Confirmatory Item 14.2-1 is resolved.

In FSAR Subsection 14.2.3.2.1, "Review and Approval Responsibilities," (STD COL 14.4-4), the third paragraph had been revised to add a LMA for LNP COL 14.4-4 to reflect replacement of the generic title of plant manager with the Levy site specific title of VP – Levy Nuclear Plant in the last sentence of the paragraph. In Revision 6, FSAR Subsection 14.2.3.2.1, the applicant removed the LNP COL 14.4-4 text and replaced it with comparable text from STD COL 14.4-4, changing the second paragraph to state, "The plant manager approves fuel loading."

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 LNP 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. In a letter dated March 7, 2011, the applicant endorsed the VEGP letter dated October 15, 2010, that proposed revisions to Items 14.4-3 and 14.4-4.

- 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. In a letter dated March 7, 2011, the applicant endorsed the VEGP letter dated October 15, 2010, that proposed revisions to Items 14.4-3 and 14.4-4.

- Part 10, License Condition 8

The proposed license condition will require the licensee to report any changes to the initial test program 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 LNP 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 LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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.*

Resolution of Standard Content Confirmatory Item 14.2-3

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.

Resolution of Standard Content Open Item 14.2-1

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, 2009, 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.

Supplemental Information

- *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 to 10 CFR Part 52 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:

<u>AP1000 COL Information Item</u>

- *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 8, 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.

Resolution of Standard Content Confirmatory Items 14.2-5 and 14.2-6

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.

Evaluation of Revised License Condition 2, Items 14.4-3 and 14.4-4

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 1 month of change, any changes to the Initial Startup Test Program described in Chapter 14 of the LNP 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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the information presented in the LNP 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: 1) the administrative controls for the review, evaluation, and approval of test results by qualified personnel; and 2) the resolution of test exceptions or tests that do not meet the acceptance criteria during each major phase of the initial test program. In addition, this standard COL action item 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 LNP COL FSAR, Revision 9, 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 LNP 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 LNP COL FSAR, Revision 9, 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 LNP 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

- Part 10, License Condition 2, Item 14.4-6

The proposed license condition addresses first-plant-only and three-plant-only tests. In a letter dated March 7, 2011, the applicant endorsed the VEGP letter dated October 15, 2010, that proposed a revision to License Condition Item 14.4-6.

- Part 10, License Condition 7

The proposed license condition will require the licensee to provide notification when first-plant-only and three-plant-only tests are completed. In a letter dated March 7, 2011, the applicant endorsed the VEGP letter dated October 15, 2010, that proposed a revision to proposed License Condition 7.

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 LNP 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 LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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.*

<u>*Resolution of Standard Content Confirmatory Item 14.2-7*</u>

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.

<u>*License Conditions*</u>

- *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, which 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*

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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the information presented in the LNP 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 LNP COL FSAR, Revision 9, 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 LNP COL FSAR Section 14.2.6, the applicant provided the following:

<u>*AP1000 COL Information Item*</u>

- 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 LNP 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 LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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.*

<u>*Resolution of Standard Content Confirmatory Item 14.2-8*</u>

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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the information presented in the LNP 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 LNP COL FSAR, Revision 9, 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 LNP 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 LNP COL FSAR, Revision 9, 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 LNP 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.

- LNP SUP 14.2-1

The applicant provided supplemental information as to the content and conditions that should be included in site-specific initial test procedures.

In addition, in Part 10 of the LNP COL application, the applicant provided the following:

License Conditions

- 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 9

The proposed license condition addresses review and evaluation of test results, as well as notification to the NRC of completion of the test phases, during power-ascension. In a letter dated March 7, 2011, the applicant endorsed the VEGP letter dated October 15, 2010, that proposed a revision to License Condition 9.

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 LNP 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 LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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:

> *<u>Supplemental Information</u>*
>
> - *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:*
>
> *<u>Test Program Schedule</u>*
>
> *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.

License Conditions

- *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*

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 testing (<5% 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 low power testing.*

---

[2] The first letter dated June 18, 2010, provided proposed License Condition 7, which is evaluated in Section 14.2.5 of this SER.

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.

LNP Site-Specific Supplemental Information

- LNP SUP 14.2-1

The applicant provided supplemental information as to the content of site-specific test procedures. The applicant stated that test procedures should provide guidance as to the expected plant response and instructions concerning what conditions warrant aborting a test, a means to control changes, and provide specific criteria and steps for safe and orderly test termination. The staff finds the site specific supplemental information provided is consistent with the guidance in NUREG-0800 Section 14.2 and RG 1.68, and therefore acceptable.

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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the information presented in the LNP 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 and LNP SUP 14.2-1 are acceptable because they provide (1) an adequate description of the administrative measures for the development of a site-specific initial test program schedule and (2) direction for the development of site-specific initial test program procedures that meet 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 LNP COL FSAR, Revision 9, 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 LNP COL FSAR, the applicant provided the following:

Departures

- LNP DEP 6.4-2

The applicant provided additional information in Section 14.2.9 of the LNP COL FSAR about LNP 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 LNP 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.

LNP COL Information Item

- LNP COL 14.4-5

The applicant provided additional information in LNP COL 14.4-5 to address site specific interface requirements for integrated testing of the raw water system.

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 LNP 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 LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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:

> *AP1000 COL Information Items*
>
> - *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.

LNP COL Information Item

- LNP COL 14.4-5

In FSAR Subsection 14.2.9.4.24, Raw Water System (STD COL 14.4-5), the COL applicant revised this section to add a site-specific LMA of LNP COL 14.4-5. The LMA is necessary to specifically address this portion of the LNP FSAR dealing with General Test Methods and Acceptance Criteria. LNP does not include heat tracing on the raw water cooling system piping, thus the subparagraph c. in the R-COLA subsection regarding component and integrated system tests was deleted. The staff found this acceptable since heat tracing is not required for the non-safety related raw water cooling system and since heat tracing is not needed for the Levy minimum normal temperature conditions.

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 [operability] *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 [sic] [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 14.3, 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. 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****.*

LNP Resolution of Standard Content VEGP Confirmatory Item 14.2-2

The staff verified that the LNP applicant has independently incorporated changes into its FSAR involving initial testing of pressurizer surge line piping. On this basis, VEGP Confirmatory Item 14.2-2 is resolved.

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.

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 [sic] [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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the information presented in the LNP 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:

- LNP 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 and LNP COL 14.4-5 are acceptable because they provide an adequate description of testing of structures and systems that are outside the scope of the DC, but within the scope of the COL.

- 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 LNP COL FSAR, Revision 9, 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 LNP 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 related to cooling towers. 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 LNP 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 LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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:

> *<u>Supplemental Information</u>*
>
> - *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.*

<u>Resolution of Standard Content Confirmatory Item 14.2-12</u>

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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the information presented in the LNP 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 <u>Certified Design Material (Related to RG 1.206, Section C.III.1, Chapter 14, C.I.14.3, "Inspections, Tests, Analyses, and Acceptance Criteria")</u>

14.3.1 Introduction

This section addresses the selection criteria and processes used to develop the LNP 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 LNP COL FSAR, Revision 9, incorporates by reference Section 14.3 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Section 14.3, the applicant provided the following:

Departures

- LNP DEP 3.2-1

The applicant revised DCD Table 14.3-2, "Design Basis Accident Analysis," Sheets 7 and 8 of 17, as new LNP COL FSAR Table 14.3-202, Sheets 1 and 2, providing additional information about LNP 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 LNP DEP 3.2-1 information appearing in other chapters of the LNP COL FSAR, is reviewed in Section 21.1 of the SER.

- LNP DEP 6.4-1

The applicant provided additional information in Section 14.3 of the LNP COL FSAR about LNP 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 LNP DEP 6.4-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.2 of this SER.

- LNP DEP 6.4-2

The applicant provided additional information in Section 14.3 of the LNP COL FSAR about LNP 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 LNP DEP 6.4-2 information appearing in other chapters of the FSAR, is reviewed in Section 21.3 of this SER.

- LNP DEP 7.3-1

The applicant provided additional information in Section 14.3 of the LNP COL FSAR about LNP 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 LNP 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.

- STD COL 13.6-1

The applicant provided additional information in STD COL 13.6-1 to state that the generic physical-security inspection, test, analysis, and acceptance criteria (PS-ITAAC) have been developed in a coordinated effort between the NRC and the Nuclear Energy Institute (NEI).

Supplemental Information

- STD SUP 14.3-1

The applicant provided supplemental information in STD SUP 14.3-1 in LNP 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.

- LNP SUP 14.3-2

The applicant provided supplemental information in LNP SUP 14.3-2 in LNP COL FSAR Section 14.3.2.3.3, “Other Site–Specific Systems,” discussing the ITAAC screening summary for site-specific systems.

- LNP SUP 14.3-3

The applicant provided supplemental information in LNP SUP 14.3-3 in LNP COL FSAR Section 14.3.3.1, “Roller Compacted Concrete Bridging Mat ITAAC (RCC-ITAAC).” The staff’s review of this ITAAC is documented in Section 3.8 of this SER.

- LNP SUP 2.5-17

The applicant provided supplemental information in LNP SUP 2.5-17 in LNP COL FSAR Section 14.3.3.2, "Waterproof Membrane ITAAC." This section describes the design of the waterproof membrane beneath the nuclear island basemat in AP1000 DCD Section 3.4.1.1.1.1. The staff's review of this ITAAC is documented in Section 3.8 of this SER.

- LNP SUP 14.3-4

The applicant provided supplemental information in LNP SUP 14.3-4 in LNP COL FSAR Section 14.3.3.3, "Turbine Building, Radwaste Building, and Annex Building Drilled Shafts ITAAC." The staff's review of this ITAAC is documented in Section 3.8 of this SER.

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 LNP 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 LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5 to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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.*

Supplemental Information

- STD SUP 14.3-1, addressing SS-ITAAC

- LNP 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 LNP 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) [STD COL 13.6-1] 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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the information presented in the LNP 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:

- LNP 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.

- LNP 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.

- LNP 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.

- LNP 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.

- LNP SUP 14.3 3, related to the RCC bridging mat ITAAC, is reviewed by the staff in Section 3.8 of this SER.

- LNP SUP 2.5 17, related to the ITAAC for the waterproof membrane beneath the nuclear island basemat, is reviewed by the staff in Section 3.8 of this SER.

- LNP SUP 14.3 4, related to the Turbine Building, Radwaste Building, and Annex Building drilled shafts ITAAC, is reviewed by the staff in Section 3.8 of this SER.

- STD COL 3.6-1, STD COL 3.9-7, STD COL 13.6-1, STD SUP 14.3-1, and LNP SUP 14.3-2, 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 Levy Nuclear Plant (LNP) COL FSAR Chapter 15 follows the format in LNP 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 by analyzing 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 LNP COL FSAR, Revision 9, incorporates by reference Section 15.0 of the AP1000 Design Control Document (DCD), Revision 19.

Departures

- LNP DEP 3.2-1

The applicant provided additional information about LNP 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 LNP DEP 3.2-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.1 of this report.

- LNP DEP 6.4-1

The applicant provided additional information in Section 15.0.11 of the LNP COL FSAR about LNP 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 LNP 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 September 23, 2010, and March 7, 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 LNP 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 a letter dated September 23, 2010, the applicant endorsed the VEGP letter dated May 21, 2010, that proposed adding Item 15.0-1 to License Condition 2, which would confirm that the plant-operating instrumentation installed for feedwater flow measurement is a Caldon/Cameron Leading Edge Flow Meter (LEFM) CheckPlus™ system. In its letter dated March 7, 2011, the applicant endorsed the VEGP letter dated October 29, 2010, letter that revised Item 15.0-1 to state that the documentation of plant calorimetric uncertainty methodology would be addressed as a plant-specific inspections, tests, analyses and acceptance criteria (ITAAC) item in lieu of License Condition 2.

- License Condition 6

In its letter dated March 7, 2011, the applicant endorsed the VEGP letter dated October 29, 2010, that proposed adding new line items to proposed License Condition 6, associated with the power calorimetric uncertainty instrumentation.

Inspections, Tests, Analyses and Acceptance Criteria

In its letter dated March 7, 2011, the applicant endorsed the VEGP letter dated October 29, 2010, that proposed ITAAC associated with the plant calorimetric uncertainty methodology.

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.

The need to address the calorimetric power uncertainty is found in Section 15.0 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants." Specifically, NUREG-0800 Section 15.0, 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." For the LOCA analysis, 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 may be used provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error.

15.0.4 Technical Evaluation

The NRC staff reviewed Section 15.0 of the LNP 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 LNP 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

---

[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.

application (VEGP Units 3 and 4) were equally applicable to the LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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:

> <u>*AP1000 COL Information Item*</u>
>
> - *STD COL 15.0-1*
>
> *In a letter dated May 21, 2010, as revised by letters dated October 29, 2010, and February 8, 2011, 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.*
>
> <u>*Compliance with Caldon/Cameron Topical Report ER-80P*</u>
>
> *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.*

> *<u>Criterion 1</u>*
>
> *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.

> *<u>Criterion 2</u>*
>
> *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.

<u>Compliance with Caldon/Cameron Topical Report ER-157P, Revision 8</u>

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.

<u>Condition 1</u>

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.

<u>Condition 2</u>

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.*

> *<u>Condition 3</u>*
>
> *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.

> *<u>Condition 4</u>*
>
> *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.

> *<u>Condition 5</u>*
>
> *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.

<u>License Conditions</u>

- *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, October 29, 2010, and February 8, 2011, 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 March 7, 2011, the applicant endorsed a letter from the Vogtle applicant dated February 8, 2011, submitted in response to a January 24, 2011, letter from the ACRS. The letter 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™ 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™ system will be tested in accordance with the LEFM CheckPlus™ 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 is tracking incorporation of the proposed changes to the LNP COL FSAR as **LNP Confirmatory Item 15.0-1**.

Resolution of LNP Confirmatory Item 15.0-1

LNP 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. LNP Confirmatory Item 15.0-1 is a duplicate of Standard Content Confirmatory Item 15.0-1. The staff verified that the LNP COL FSAR was appropriately revised. As a result, Confirmatory Item 15.0-1 is now closed.

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 proposed by the applicant 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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP COL application is acceptable and meets the NRC regulations. The staff based its conclusion on the following:

- LNP 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.

- LNP 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 LNP 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 LNP COL FSAR, Revision 9, 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 LNP COL FSAR, the applicant provided the following:

Departures

- LNP DEP 6.4-1

The applicant provided additional information in Section 15.1.5 of the LNP COL FSAR about LNP 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 LNP 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 LNP 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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP 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 LNP COL FSAR, Revision 9, 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 LNP COL FSAR, the applicant provided the following:

Departures

- LNP DEP 6.3-1

The applicant provided additional information about LNP 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 LNP DEP 6.3-1 information appearing in other chapters of the FSAR, is reviewed in Section 21.1 of this report.

- LNP DEP 3.2-1

The applicant provided additional information about LNP 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 LNP 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 LNP 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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP 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 LNP COL FSAR, Revision 9, incorporates by reference Section 15.3, "Decrease in Reactor Coolant System Flow Rate," of Revision 19 of the AP1000 DCD. In addition, in the LNP COL FSAR, the applicant provided the following:

Departures

- LNP DEP 6.4-1

The applicant provided additional information in Section 15.3.3 of the LNP COL FSAR about LNP 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 LNP 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 LNP 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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

15.4 Reactivity and Power Distribution Anomalies

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 LNP COL FSAR, Revision 9, incorporates by reference Section 15.4 of the AP1000 DCD, Revision 19. In addition, in the LNP COL FSAR, the applicant provided the following:

Departures

- LNP DEP 6.4-1

The applicant provided additional information in Sections 15.4.8 and 15.4.10 of the LNP COL FSAR about LNP 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 LNP 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 Table 1.9-204 of the FSAR, as part of STD COL 1.9-2 to address Bulletins and GLs, the applicant identified Generic Letter (GL) 85-05, "Inadvertent Boron Dilution Events."

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 LNP 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 LNP 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 LNP Units 1 and 2 COL application, the staff undertook the following reviews:

- The staff compared the VEGP COL FSAR, Revision 5, to the LNP COL FSAR. In performing this comparison, the staff considered changes made to the LNP 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 LNP 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:

> *<u>Generic Letter 85-05</u>*
>
> *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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP 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. LNP 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 LNP COL FSAR, Revision 9, 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 LNP 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 LNP COL FSAR, Revision 9, incorporates by reference Section 15.6, "Decrease in Reactor Coolant Inventory," of Revision 19 of the AP1000 DCD. In addition, in the LNP COL FSAR, the applicant provided the following:

Departures

- LNP 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 LNP COL FSAR about LNP 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 LNP 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

- LNP COL 2.3-4

The applicant provided additional information in LNP COL 2.3-4 related to site-specific atmospheric dispersion factor (χ/Q) values. The effect of LNP COL 2.3-4 on the DBA radiological consequences analyses is addressed in Section 15A of this SER.

The NRC staff reviewed Section 15.6 of the LNP 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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

15.7 Radioactive Release From a Subsystem or Component

15.7.1 Introduction

The group of events considered includes 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

15.7.2 Summary of Application

Section 15.7 of the LNP COL FSAR, Revision 9, incorporates by reference Section 15.7 of the AP1000 DCD, Revision 19.

In addition, in LNP COL FSAR Section 15.7, the applicant provided the following:

Departures

- LNP DEP 6.4-1

The applicant provided additional information in Section 15.7.4 of the LNP COL FSAR about LNP 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 LNP 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

- LNP COL 15.7-1

The applicant provided additional information in LNP 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 LNP 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, including Branch Technical Position (BTP) 11-6, and Section 2.4.13 of NUREG-0800, Acceptance Criterion Number 5.

15.7.4 Technical Evaluation

The NRC staff reviewed Section 15.7 of the LNP 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 LNP COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the LNP COL FSAR:

AP1000 COL Information Item

- LNP 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 LNP 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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP 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:

- LNP 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.

- LNP COL 15.7-1 is acceptable based on the evaluations in Sections 2.4.13 and 11.2 of this SER.

15.8 Anticipated Transients Without Scram

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 LNP COL FSAR, Revision 9, 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 LNP 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 LNP COL FSAR, Revision 9, Chapter 15, "Accident Analyses," the applicant incorporated by reference Appendix 15A to Chapter 15, "Accident Analysis," of the AP1000 DCD, Revision 19.

In addition, the applicant provided the following:

Departures

- LNP DEP 6.4-1

The applicant provided additional information in Appendix 15A of the LNP COL FSAR about LNP 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. LNP DEP 6.4-1 revises the analysis of the radiological consequences described in this section of the SER. This information, as well as related LNP 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

- LNP COL 2.3-4

In LNP COL FSAR Sections 15.6 and 15A, the applicant provided additional information in LNP COL 2.3-4 on site-specific χ/Q values to partially resolve COL Information Item 2.3-4. The applicant provided additional information in LNP 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 for the application for a COL are given in 10 CFR 52.79, "Contents of applications; technical information in final safety analysis report."

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).

(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 LNP 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.

Paragraph IV.E.8 of Appendix E to 10 CFR Part 50 requires that an onsite technical support center (TSC) be provided from which effective direction can be given and effective control can be exercised during an emergency. The associated guidance in NUREG-0696 Section 2.6 states that the TSC shall have the same radiological habitability as the control room under accident conditions, and TSC personnel shall be protected from radiological hazards to the same degree as control room personnel (see also, Section 8.2.1.f of NUREG-0737, Supplement 1). The radiation protection design of the TSC is acceptable if the total calculated radiological consequences for the postulated fission product release fall within the exposure acceptance criterion specified for the control room of 0.05 Sv (5 rem) TEDE for the duration of the accident.

15A.4 Technical Evaluation

The NRC staff reviewed Appendix 15A to Chapter 15 of the LNP 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 LNP COL application are documented in NUREG-1793 and its supplements.

The staff reviewed the information in the LNP COL FSAR:

Departures

- LNP DEP 6.4-1

The applicant provided additional information in Appendix 15A of the LNP COL FSAR about LNP 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.

LNP 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

- LNP COL 2.3-4

In LNP COL FSAR Sections 15.6 and 15A, the applicant stated that it provided additional information in LNP 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, Revision 0, 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).

LNP 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 LNP 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-specific χ/Q values given in response to LNP 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 used design reference values for the accident atmospheric dispersion factors in place of site-specific values. The

χ/Q values are the only input to the DBA radiological consequences analyses that are affected by the site characteristics. To resolve LNP COL 2.3-4, the applicant discussed the LNP site-specific short-term (accident) χ/Q values in LNP COL FSAR Section 2.3.4. The LNP site-specific EAB and LPZ χ/Q values for DBAs are given in LNP COL FSAR Table 2.0-201, and the control room χ/Q values for DBAs are given in LNP COL FSAR Table 2.0-202. In Section 2.3.4 of this SER, the NRC staff discusses its review of the LNP site-specific χ/Q values and resolution to LNP COL 2.3-4.

The AP1000 design includes a dedicated location for the technical support center (TSC) in the control support area (CSA). The CSA is served by the non-safety related nuclear island nonradioactive ventilation system (VBS), which also serves the main control room. The VBS non-safety accident responds to detection of a high gaseous radioactivity concentration in the supply air duct by initiating pressurization, filtered intake, and filtered recirculation of the air within the main control room and CSA. Staff review of the VBS is documented in Section 9.4 of NUREG-1793 and its supplements. Staff review of the accident radiological consequences in the main control room and CSA (which includes the TSC) when the VBS is operating is documented in Section 15.3 of NUREG-1793, and its supplements. The control room χ/Q values for DBAs are applicable to the operation of the VBS, and are used in the TSC dose analysis. As discussed above, the LNP control room χ/Q values are bounded by the referenced DCD values, and therefore the applicant incorporated by reference the AP1000 DCD evaluation of the TSC dose. Section 13.3 of this SER discusses the staff's review of the emergency response facilities, including the TSC.

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 DCD. All other inputs and assumptions in the radiological consequences analyses remain the same as in the 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 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 by Westinghouse in the AP1000 DCD radiological consequences analyses. Since 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 DCD design reference χ/Q values for all time periods and all accidents, the LNP site-specific estimated total dose for each DBA is, therefore, less than the AP1000 DCD estimated total dose for each DBA.

Since the AP1000 DCD Chapter 15 DBA radiological consequences analyses show that the offsite radiological consequences meet the regulatory dose requirements of 10 CFR 52.47(a)(2) and the control room consequences meet the regulatory dose requirements of GDC 19, and since, by the logic above, the LNP site-specific DBA radiological consequences are estimated to be less than those calculated in AP1000 DCD, the applicant has sufficiently shown that 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.

Since the AP1000 DCD Chapter 15 DBA radiological consequences analyses show that the radiological consequences in the TSC fall within the acceptance criterion of 0.05 Sv (5 rem) TEDE for the duration of the accident, and since, by the logic above, the LNP site-specific DBA radiological consequences are estimated to be less than those calculated in AP1000 DCD, the applicant has sufficiently shown that the DBA consequences in the TSC meet the requirements of paragraph IV.E.8 of Appendix E to 10 CFR Part 50.

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 above, 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. By the same logic above, 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 the additional FSAR information for LNP 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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP COL application are documented in NUREG-1793 and its supplements.

In addition, the staff concludes that the relevant information presented in the LNP COL FSAR is acceptable and meets the requirements of 10 CFR 52.79(a)(1),10 CFR Part 50, Appendix A, GDC 19, and paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The staff based its conclusion on the following:

- LNP COL 2.3-4 is acceptable because the DBA offsite radiological consequences meet the requirements of 10 CFR 52.79(a)(1), the DBA control room radiological consequences meet the requirements of GDC 19, and the DBA radiological consequences in the TSC meet the requirements of paragraph IV.E.8 of Appendix E to 10 CFR Part 50.

- LNP 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.

Appendix 15B Removal of Airborne Activity from the Containment Atmosphere Following a LOCA

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 LNP COL FSAR, Revision 9, 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 LNP COL FSAR, the applicant provided the following:

Departures

- LNP DEP 6.4-1

The applicant provided additional information in Appendix 15B of the LNP COL FSAR about LNP 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 LNP 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 LNP 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 LNP COL FSAR related to this section. The results of the NRC staff's technical evaluation of the information incorporated by reference in the LNP 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.